Physics of liquid metal fast breeder reactor safety

Richard Wilson*

Department of Physics, Harvard University, Cambridge, Massachusetts 02138

The main physical features of LMFBR safety are summarized with particular reference to the maximum possible mechanical energy release in a hypothetical core disruption accident. It is likely that such a postulated fuel melting accident would lead to a nonenergetic disassembly of the core. With pessimistic estimates of reactivity increaases, but not derived from any mechanistic point of view, mechanical energy impacts on the reactor vessel of 100 MJ have been calculated for the Clinch River Breeder Reactor. These are containable with the present design, and the increase should vary only linearly with increased power for future designs of similar concept. Various possible steps are discussed to increase the certainty of these conclusions.

CONTENTS

*This paper is based on work performed for the Electric Power Research Institute, Palo Alto, California.

I. INTRODUCTION

How should society address issues (e.g., breeder reactor or earthquake) where there is a small, but finite, probability of an accident involving many thousands of people?

There have been several suggested procedures, and in the case of the breeder reactor, three are being pursued simultaneously. First, extensive design work, and in the future careful operation, can reduce the probability of a major accident to a low level. Second, the design of the reactor and its containment vessel can be arranged so that the energy released in a major accident will be contained and not released to the public. Third, a careful choice of site can be arranged so that the maximum number of people involved in an uncontrolled radioactivity release is bounded, and society can recover from an accident if it occurs, as society has recovered from accidents so many times in the past.

There tends to be a division of labor in these approaches between physicists and engineers. It is convenient to make a distinction between (1) engineered safeguards, where any malfunction is corrected by some engineered, either active or passive, device designed for the purpose and (2) inherent physical principles, which can limit the rate of progression of an accident and perhaps set an upper bound to energy release and subsequent radioactivity release. However, there is a considerable interplay between these two approaches; careful use of physical principles in engineering design can prevent malfunctions leading to an accident, and careful engineering can help limit an accident.

There is also a distinction, not very rigid, between the work done by physicists and engineers. Engineers tend to concentrate on the prevention of accidents; physicists tend to concentrate on understanding the details of accident progression and, in the case of the liquid metal fast breeder reactor, on the hypothetical core disruptive accident (HCDA). Therefore physicists tend to be blinded to the fact that when an LMFBR is properly designed, built, and operated fuel melting is very unlikely indeed.

The first experimental sodium-cooled fast reactors were small, and it was possible to build strong enough vessels to withstand all postulated accidents and containment vessels to contain radioactivity. As larger reactors are built, and the cost of reactor vessels and

Copyright © 1977 American Physical Society

containments becomes of commercial significance, it is important to reassess this question. We could ensure safety by preventing any malfunction leading to a core meltdown, an achievement dependent upon reliable engineered safety devices; or we could try to ensure that a core meltdown leads to no energy release that can violate containment, and this usually demands an understanding of complex physical situations; or we could try to ensure that the radioactivity releases in the event of containment violation cause only a limited hazard.

Each of these steps seems to be independent of the others, so that even if there is not absolute certainty of any one of these safety procedures, use of all of them can lead to a very low probability of extreme consequences and, therefore, an acceptable risk. This is the philosophy usually called defense in depth.

The original intention of the author in undertaking this work was to attempt to place an upper limit, on physical grounds, to the mechanical energy release in an accident involving a liquid metal fast breeder reactor. However, the concept of placing an upper limit was found impractical; energetics should be related to a probability of occurrence. In this survey, therefore, I discuss the physical concerns and possible ways in which our physical understanding can be improved so as to reduce the uncertainty in energetics prediction.

In the interest of brevity I will only briefly discuss these other engineered safeguards although it is necessary to realize that they exist.

In discussion of the hypothetical core disruptive accident itself, much attention has been given to detailed mechanistic calculations of fuel melting and movement using complex computer codes (PMC, 1975). Computer codes are hard to understand, and for light-water reactor safety an American Physical Society study group (Lewis, 1975) was not convinced that existing codes are adequate and have been adequately tested. Accordingly, Fauske (1976) has made a plea for an assessment of the core meltdown process on "general behavior principles" —which I here call physical principles.

Although reliance on sound physical principles may be preferable to reliance on an engineered safeguard, there is no rigid distinction. Physical principles are sometimes not as sound as originally perceived, and moreover they must be applied to a technological situation which is not as ideal as the rigid application of the principle demands. This application depends on people, and people are fallible. In many cases, therefore, designers assign a probability to the reliability of a physical principle just as if it were an engineered device.

In this report, therefore, I will follow the development of a hypothetical core disassembly accident placing emphasis upon the inherent physical principles which might prevent an energy release large enough to breach the containment instantly. We will see that such a breach is very unlikely, and I point out various efforts at prevention and mitigation, the reactor design criteria which are important, and the places where further research would be useful.

Since every statement of a possible danger or concern has a short discussion of possible ways of alleviating that concern, it is, of course, important that such statements not be quoted out of context.

This report is in many ways very presumptuous. Many dozens of competent scientists have worked in this field for dozens of years. But very shortly, fast breeder reactor safety must be explained to a wider audience, and eventually to the "general public." In this endeavor a newcomer's explanation of how he perceives the issues may be helpful.

In this way, this review may be an introduction to books on the subject such as those by Graham (1971) or Farmer.

A. The liquid metal fast breeder reactor (LMFBR)

The liquid metal fast breeder reactor, as now conceived, has several parts. The reactor core is usually a squat cylinder about one meter high with a two-meter radius. The fuel is plutonium in the form of plutonium oxide, mixed with uranium-238 oxide in sintered ceramic pellets and encased in stainless steel to form fuel rods about 7 mm in diameter. Liquid sodium, or in some older reactors sodium/potassium mixture, circulates between the rods as a heat transport medium.

The outer rods and the top and bottom of the central rods are arranged to consist mostly of uranium, and form a fertile blanket where the uranium-238 is transmuted to plutonium by neutron capture. In the central part of the reactor core, or main part of the reactor, about 20% of the fuel is plutonium oxide.

The fuel rods are arranged in *subassemblies* of between 200 and 350 rods each, with a subassembly duct surrounding them and acting as a separating partition. Liquid sodium is used as a heat transport medium. It has a high thermal capacity, and its boiling point is high (Table I). Heat exchangers are used first to transfer the heat to a secondary nonradioactive sodium circuit, and then to generate steam for a steam turbine.

Neutrons, atomic mass 1, slow down only a small amount in collision with sodium atoms, atomic mass 23. A *slow* neutron chain reaction cannot therefore take place, and the fuel is arranged to have a high proportion of fissile fuel to enable a fast neutron chain reaction to take place.

The whole primary sodium circuit is placed within the reactor containment building for isolation. In the LOOP-type reactor (Fig. 1), the components are separate; in the POOL-type reactor (Fig. 2), they are all immersed in the sodium POOL and separated by partitions. EBR-I, EBR-II in the U. S., PHENIX in France, PFR in the U. K., and BN 600 in the Soviet Union are POOL-type reactors, whereas the BR5 and BN 500 in the Soviet Union, SNR 300 in Germany, FFTF and the Clinch River Breed-

TABLE I. Some physical parameters.

	Stainless steel	Sodium	Mixed oxide fuel	Mixed carbide fuel
Melting point (°K)	1670	371	3070	2673
Boiling point (°K)	3070	1156		4200
Latent heat of evaporation (J/kg)		3.8×10^6	2.8×10^5	
Specific heat (J/kg °C)	550	1270	637	900

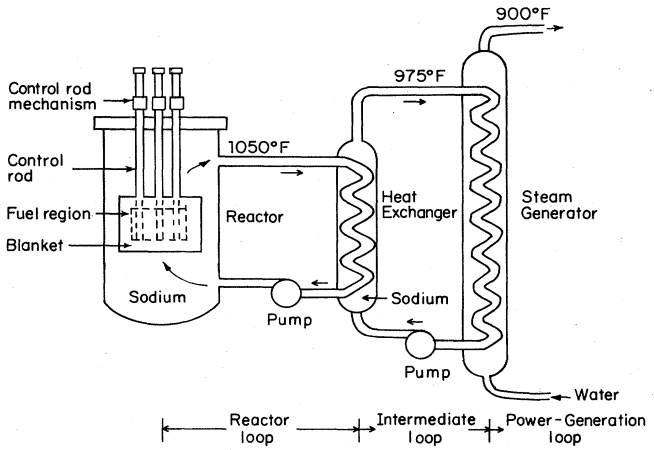

FIG. 1. Schematic for a LOOP-type liquid metal fast breeder reactor.

er Reactor (CRBR) in the U. S. are of the LOOP-type. The POOL-type has a larger sodium inventory, although the amount is very large in both cases. This leads to a slower thermal response and a possibility of a larger sodium fire, although this danger is considered to be of very low probability. Both operate near atmospheric pressure, so that the pipe-break problem which dominates light-water reactor safety is less important, but in the LOOP type the possibility exists of pumping the primary coolant out through a break onto the floor. Such an accident would require a vessel break in the POOL-type. For a LOOP reactor the effects of a coolant leak are minimized by limiting the available volumes onto which sodium could leak. When components fail, it might seem easier to change them with a LOOP-type reactor; but the experience of PHENIX suggests that this argument is not important. Figure 3 shows details of the PHENIX reactor, a POOL-type, which is the most successful large breeder reactor to date, since it has operated regularly as part of the French electricity grid with high reliability.

None of these safety differences between POOL and LOOP reactors are dominant, and all can be handled by good engineering. The choice between the two types rests on engineering and cost considerations outside the scope of this review.

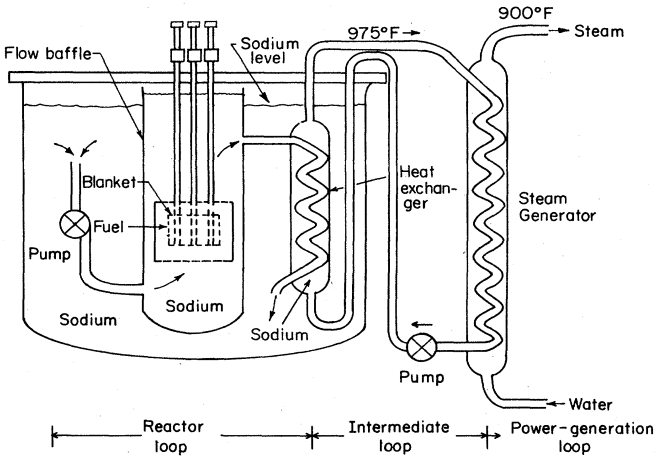

FIG. 2. Schematic for a POOL-type liquid metal fast breeder reactor.

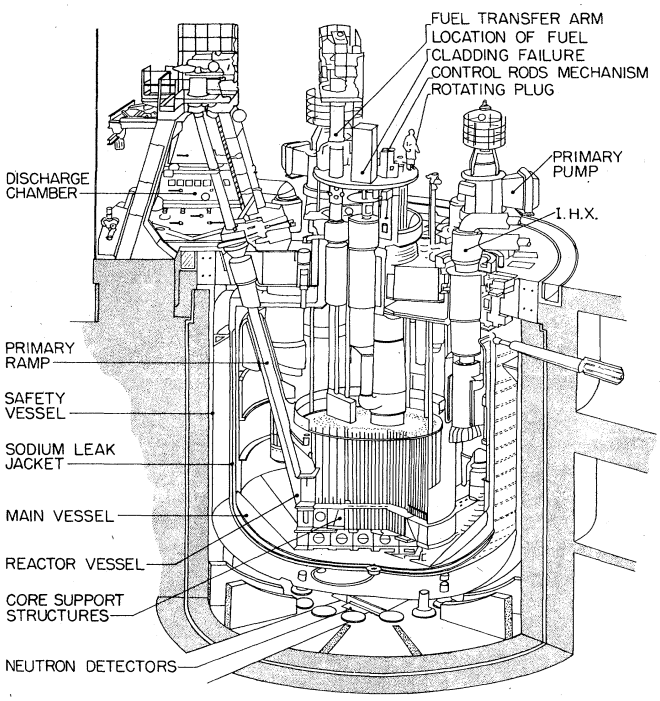

FIG. 3. Details of the PHENIX reactor, Marcoule, France.

II. GENERAL REACTOR SAFETY FEATURES

A. Reactor stability and control

In Sec. II. A–II. E which follow, the areas of reactor theory are outlined which are relevant to a discussion of safety. Dynamic reactor calculations can be made with great accuracy, and there are many good texts on the subject (Hetrick, 1971; Bell, and Glasstone, 1970; Galanin, 1960; Glasstone and Edlund, 1952; Isbin, 1963; Kramerov and Shevelev, 1964; Weinberg and Wigner, 1958; Keepin, 1965), and I only summarize here.

Once the chain reaction starts, it must be kept under control to maintain a constant power level, so the neutrons generated by fission must precisely reproduce themselves leaving none left over for increasing the reaction. One aim of reactor design is to ensure that the reactor remains stable by an inherent feedback mechanism if there is a small change in a reactor parameter. If a large change occurs due to some transient, the reactor must be shut down, and it is important to be sure both that there is time to shut down and that the shutdown mechanism occurs.

In a neutron chain reaction we can define the neutron reproduction factor k as the rate of generation of neutrons, divided by the rate of disappearance of neutrons by absorption or escape.

The reactivity ρ is defined by $\rho = (k-1)/k$. If the reactivity is greater than zero, the power will increase. The basic (simplified) equations for this increase, assuming all neutrons have the same lifetime, are:

$$\frac{1}{P}\frac{dP}{dt} = \frac{1}{n}\frac{dn}{dt} = \frac{k-1}{l_0} = \frac{\rho}{l}, \tag{1}$$

where n is the number of neutrons, P is the reactor power, and l_0 is the neutron lifetime from generation to absorption or capture. Sometimes the neutron generation time $l = kl_0$ is used, which for our purposes is similar; the main variations are in ρ or $k-1$. For rapid increases of power there is no time to allow heat flow from fuel to coolant, so that if Q is the sensible heat of the fuel, C the average specific heat, and T the temperature,

$$P = dQ/dt \quad \Delta T = \Delta Q/C. \tag{2}$$

The reactivity increase leads to a power increase; this leads to a temperature increase. The reactor is designed so that thermal expansion of the fuel moderator if any, spacers between the fuel, and the Doppler effect (to be discussed in detail in Sec. C) lead to a *reduction* in reactivity, and therefore contribute to stability. These equations lead to a time delay between a reactivity increase and the controlling reduction of reactivity due to a (negative) stability coefficient. Time delays also occur because of the necessity of conducting heat from the fuel to the item which is expanding; this leads to complex calculations, often with oscillatory solutions, which we avoid presenting here. For our purposes, it suffices that designs can make the overall stability conditions negative, and these stability coefficients can be measured during reactor startup so that they are well known. However, not all fast reactors will have each individual stability coefficient negative, so a very close examination is necessary in each case.

B. Delayed neutrons

In a fast reactor, the neutrons are captured in less than a microsecond, and it might appear that we should put $t \approx 1$ μsec in the above equations. It is not possible mechanically to insert control rods that quickly. But control is possible because of delayed neutrons. Approximately 0.21% of the neutrons produced in fast fission of plutonium-239 (0.65% for uranium-235 and 1.6% for uranium-238) come from radioactive daughter products with various half-lives, with the most abundant ones of 2.1, 5.6, and 23 sec., respectively. Therefore, if the changes in reactivity are less than the fractional number of delayed neutrons, the effective value of the lifetime will be of the order of seconds and there will then be time for operation of various control devices.

The importance of the delayed neutrons is so great that reactivity changes are expressed in a special jargon: a fractional change in reactivity equal to the fraction of delayed neutrons is one dollar's worth ($1). Thus a $1 reactivity insertion is an insertion of reactivity of 0.21% for ^{239}Pu (or 0.65% for ^{235}U). With faster neutrons, fission of the fertile isotope ^{238}U (with 1.6% delayed neutrons) leads to an average delayed neutron fraction of about 0.33% in a typical fast reactor. The fraction of delayed neutrons then increases a little with neutron energy. The "effective" number of delayed neutrons is also a little greater than this; the delayed neutrons have a lower energy than fission neutrons and produce relatively more fissions. This small effect depends upon reactor details.

Another piece of common jargon is the use of the term prompt critical. A reactor is "critical" when k equals 1. It is "prompt critical" when k exceeds 1 by the fraction of delayed neutrons; (= 1.0033 for a typical LMFBR). If a reactor is made prompt critical, the chain reaction can be sustained on prompt neutrons ($t \approx 0.1$ μsec) alone without waiting for the appearance of delayed neutrons. Accordingly very rapid rates of increase of power are possible.

A more precise discussion, incorporating delayed neutrons, leads to the equations

$$\frac{dn(t)}{dt} = \frac{(\rho-\beta)}{l}n + \sum_i \lambda_i c_i + q \tag{3a}$$

$$\frac{dc_i(t)}{dt} = \frac{\beta_i n}{l} - \lambda_i c_i \tag{3b}$$

where l is the lifetime of the prompt neutrons, λ_i is the decay constant for the ith group of delayed neutrons, β_i is the effective fraction of the ith group of delayed neutrons, $\beta = \sum \beta_i$, and q is the neutron source term.

For a step change of reactivity after a reactor has been critical for a long time, $dc/dt \approx 0$ and we find by the method of Laplace transforms

$$n(t) = \sum_{n=1}^{7} N_k \exp(t/T_k), \tag{4}$$

where T_k are roots of the 7th-order polynomial

$$\frac{k-1}{k} = \frac{l_0}{kT} + \sum_{i=1}^{6} \frac{\beta i}{1+\lambda_i T} \tag{5}$$

This is the *inhour* (for inverse hour) equation for the inverse reactor period T. For positive reactivity ρ, the dominant term is a *growing* exponential and the period is known as the "reactor period" or "asymptotic period" or even "stable period." Values of the asymptotic period for Pu^{239} are shown in Fig. 4 from Keepin (1965).

For super prompt critical excursions the asymptotic period is small compared with delayed neutron periods, and the equations reduce to

$$T = l_0/[k-(1+\beta)] \text{ and } n(t) = n(0)e^{(t/T)}. \tag{6}$$

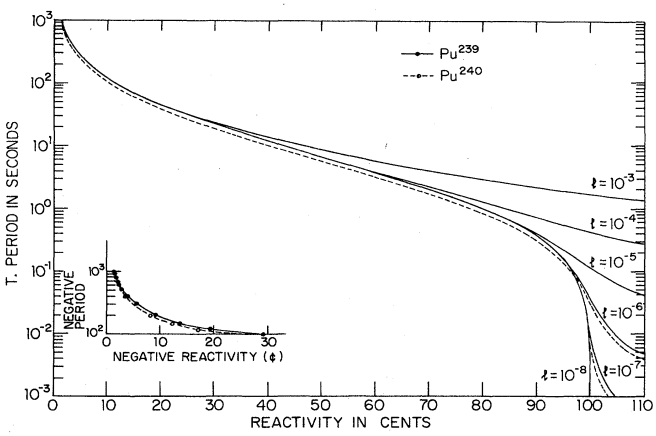

FIG. 4. Relation between the reactivity and the asymptotic reactor period for systems involving pure ^{239}Pu and ^{240}Pu. Here t is the lifetime of the neutrons from generation to capture.

TABLE II. Delayed neutrons from thermal neutrons in ^{239}Pu (Keepin, 1965).

Group	Half-life (secs)	Delay constant (sec^{-1})	Relative yield	Neutrons/ fission
1	54.28	0.0128	0.035	0.000 21
2	23.04	0.0301	0.298	0.001 82
3	5.60	0.1240	0.211	0.001 29
4	2.13	0.3250	0.326	0.001 99
5	0.618	1.1200	0.086	0.000 52
6	0.257	2.6900	0.044	0.000 27
			1.000	0.006 10

For small reactivity increases over critical, T is dominated by the delayed neutron periods ($1/\lambda_i$) and is given by:

$$T = (l_0 + \sum_{i=1}^{6} \beta_i/\lambda_i)/(k-1) \tag{7}$$

C. Doppler coefficient

The most important stability coefficient is the temperature coefficient of reactivity due to the neutronic properties of the elements themselves—usually called the Doppler coefficient because of the particular physical phenomena responsible for this effect.

The description below of the Doppler reactivity coefficient follows closely that of Bethe (1974). Absorption of neutrons in heavy nuclei occurs in resonances. In Fig. 5 we show how the natural cross section varies with energy of the neutrons. It has a sharp maximum and a width (2Γ) which is quite small when a resonance is

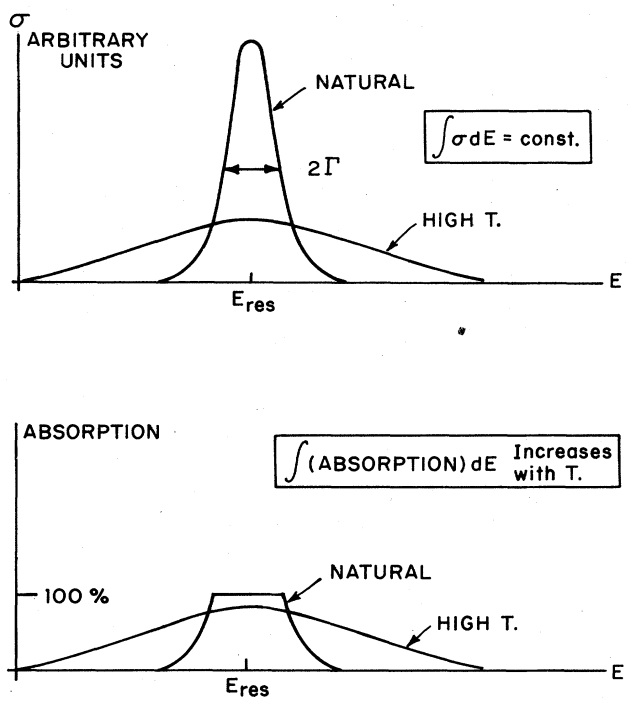

FIG. 5. An illustration showing how Doppler broadening of spectral lines increases absorption.

such that the neutron is captured by a nucleus with the emission of a gamma ray, and is considerably larger when the capture leads predominantly to fission or other nuclear breakup.

When the temperature is low, nuclei are at rest and the natural shape prevails. When the nuclei are in uniform motion the resonance energy shifts due to the Doppler effect (Rayleigh, 1888; Michelson, 1892). When the fuel is hot, the nuclei are in random motion and the resonance broadens; the area ($\int \sigma dE$) remains the same as the temperature varies. In a reactor we have neutrons of all energies, and all the neutrons with an energy within the resonance are absorbed as shown in Fig. 5. Thus the total absorption of neutrons, given by the area $= \int$(absorption)dE, increases with temperature. The fuel shields itself by absorbing most of the neutrons near the resonance capture.

When the resonance leads to fission, temperature increase leads to an increase in reactivity; when it leads to simple capture it leads to a reduction of reactivity. The natural width is smaller for capture resonances than for fission resonances, so that the change in going to the Doppler-broadened width is relatively more important for the capture resonances than the smaller change for the fission resonances. This relative increase in capture probability due to the Doppler effect leads to a net reduction of reactivity with increasing temperature.

The Doppler coefficient can be modified by increasing the amount of fertile material in the core (which captures neutrons and does not give fission) or by slowing down the neutrons somewhat. In typical LMFBR cores (the French reactor PHENIX, the UK reactor PFR, and the U. S. test reactor FFTF), the fissile material is diluted by a much larger amount of fertile material, and the overall temperature coefficient of reactivity (dk/dT) is negative. Moreover, this effect is *prompt*. The fuel UO_2 and PuO_2 are mixed intimately over distances (20μ) comparable to the distances (range of fission products) over which the fission energy is deposited [the acceptable particle size is estimated to be 75μ (Hummel *et al.*, 1966; Fischer and Keller, 1966). This intimate mixing is clearly important in fuel design, but need only be achieved on the average.] PuO_2 and UO_2 are miscible materials, so the intimate mixing will persist even if the fuel melts.

Some calculations have suggested that mixing over larger distances (100μ) is adequate. But this depends on heat transfer from PuO_2 to UO_2, and in a rapid transient (larger than those considered here) this would not be enough. The Doppler coefficient will depend upon neutron energy; for neutron energies above 100 keV resonances overlap and become a continuum and as the temperature goes up, remain a continuum. The Doppler coefficient is therefore reduced.

The importance of the Doppler coefficient is so great that there has been intensive international effort to ensure its value is well determined, and in particular that in reactor designs it is negative (leading to stability) under all circumstances (Beyman, 1974; Nordheim, 1965).

Resonances have been measured, and in the region of several MeV where they overlap, averages have been

measured. Calculational programs have been developed to add up these resonances and to take into account corrections such as overlap of resonances, and shielding of one nucleus by another. These calculations give coefficients which vary less than 20% with different cross-section sets (Russian, English, etc.). First there were experiments on critical assemblies, then a special test reactor SEFOR was built especially for Doppler effect tests. More recently with demonstration LMFBR's (PHENIX, PFR) the Doppler temperature coefficients have been measured and compared with calculation. They agree to about 15%, more than adequate for reactor design and safety analysis. The most important series are the SEFOR tests; the reactor was carefully designed to reduce the reactivity coefficient from expansion. Tests were run at a variety of temperatures, including elevated temperatures reached in accident conditions, and it was verified that the reactor shuts itself down after a pulse. We have, therefore, both a detailed physical theory and specific tests in realistic engineering situations. Unfortunately there is, at present, no good general write-up of this excellent work, although many reports exist (General Electric Company, 1970).

As the temperature increases, the resonances will be broadened and overlap. Then the Doppler coefficient dk/dT will go down with temperature. The Doppler feedback effect is then conveniently described by a number $A = +Tdk/dT$, commonly described as the Doppler constant, since it is insensitive to temperature changes.

For a typical reactor (PHENIX) of 270 MWe, $A = -0.0036$ averaged over the core. For CRBR (350 MWe) it is planned to be $A = -0.005$, and for SUPERPHENIX it is planned to be $A = -0.008$. With mixed Pu/U oxide fuel the magnitude of the Doppler coefficient is reduced as the neutron energy increases. This is because the fission probability for plutonium increases relative to the capture probability for uranium, as the energy increases and the uranium capture resonances overlap and contribute less to the Doppler coefficient. As sodium is removed from the system (as can happen in an accident condition) there is less moderation of the neutrons, and the average energy increases. There is then a greater probability of fission for which there is a positive Doppler coefficient. The average Doppler coefficient is therefore less negative. The Doppler coefficient may also be different in an accident condition where the fuel has slumped. The effect is not likely to be great, but I have not seen such a calculation.

The Doppler coefficient is vital not only for stable operation, but also for limiting and early termination of an accident, as we will later see. It is for this reason that I have gone into detail. The Doppler coefficient is based on well understood physical principles; enough data exist for good calculation and these calculations agree with experiment. Moreover, the average coefficient can be measured for the finished reactor (at operating temperature), so that no surprises can be expected. Therefore, trust can be placed in the Doppler coefficient and its effects.

The Doppler effect can be made more negative by adding more moderator (BeO for example); however, this increases cost and reduces the breeding ratio. As shown in a later section, the Doppler coefficient slows the progression of any hypothetical accident. This will not only reduce the consequences, but make the calculational estimates of them more reliable.

D. Expansion coefficients

The fuel rods are separated by coolant channels. In a fast neutron reaction, the reactivity increases if the fuel compacts; conversely, the reactivity decreases if the fuel expands.

We can therefore design so that thermal expansion of fuel and/or supports, where the core remains intact, leads to a *reduction* in reactivity. If, however, fuel does not remain intact, the fuel can either sweep out, and reduce reactivity, or slump and increase it. If the coefficients have the appropriate sign, the expansion can help to compensate a slow transient, although it is not as prompt in its action as the Doppler coefficient.

In PHENIX at Marcoule there is, and in SUPERPHENIX at Creys-Malville there will be, a flared core. The subassemblies are held at one end but can separate, upon expansion, at the other. This approximately doubles the (negative) coefficient of reactivity with temperature. This becomes about $(dk/dT) = -0.8 \times 10^{-5}/°C$ for the flared core or $(dk/dT) = -1.6 \times 10^{-5}/°C$ for all expansion sources (control rod, expansion, sodium expansion, fuel axial expansion, etc.)

Note that this compares with a Doppler effect for PHENIX $Tdk/dT = -0.0036$ or $dk/dT = -4 \times 10^{-6}/°C$, at 833°K (= 560°C). The reactivity feedback due to geometric effects is more than double those due to the Doppler effects, assuming isothermal heatup. However, during a rapid power transient the fuel would heat up at a higher rate than the coolant and structures, so the effect would not be nearly so great.

The geometric effects here are intrinsic effects and are based on simple physical principles. They can be relied upon, *until the fuel melts*. These coefficients can help in turning around a slow reactivity increase (and all initiating events have slow increases) and thereby act as an important, simple, and passive method of preventing accidents leading to a meltdown. Even if they fail to do that, they can add a time delay enabling operators to manually initiate shutdown mechanisms if the automatic ones which we have postulated have failed.

E. Sodium void coefficients

If the sodium is removed from the coolant channels, there will be a change of reactivity due to three causes:

(1) The sodium will no longer absorb neutrons, so the fraction of neutrons causing fission increases and reactivity will increase. This leads to a positive contribution to the void coefficient.

(2) The sodium will no longer slow down neutrons, so that there will be relatively more fast neutrons. For plutonium the number of neutrons in fission increases with energy as the fast neutron fission in Pu^{240} begins to contribute — one of the reasons for using plutonium in the first place — and the relative cross section for fission versus capture increases. Therefore, increasing the neutron energy increases reactivity, and this leads to a positive contribution to the void coefficient.

(3) There will be more leakage of neutrons out of the reactor, leading to a negative contribution to the void coefficient.

The first of these is small. Of the other two, (3) dominates (2) at the edge of the reactor, but (2) usually dominates (3) in the center and overall in the active core of a large LMFBR.

In a loss-of-flow accident without scram, the flow of coolant sodium is interrupted. Then the reactor will heat up, and eventually the coolant sodium will boil and the coolant channels will be voided of sodium. Then the central part of the core (proceeding radially) is likely to be voided of sodium preferentially — at least initially. The top part of each channel would be voided first, and for this the void coefficient is negative. The central part of the channel would be voided next, making the overall reactivity addition positive. If the surrounding fertile blanket is also voided, the void coefficient is less. The calculated contributions to reactivity for CRBR are:

For voiding central part of core + \$3.5 ($\triangle k/k = 0.011$)

For voiding core + blanket + \$2.5 ($\triangle k/k = 0.008$)

For larger (1200 MWe) reactors it is natural to design a larger core with a smaller ratio of surface area to volume. Then the effect (3) will be less important. Total sodium void coefficients of \$5 (SUPERPHENIX) and \$10 (Combustion, 1970) are found in some designs. If the sodium void coefficient were not positive there would be a more inherent safety. Therefore, it is desirable to see how this coefficient might be reduced and at what cost.

Several schemes have been suggested, two of which are:

(1) Since the void coefficient is negative near the edges, we can design for more edge in the core and thereby increase neutron leakage. This can be by a heterogeneous core structure — by putting some of the blanket in the middle of the core. Care must be taken not to reduce the Doppler coefficient in this process; the leakage hardens the neutron spectrum (produces a larger fraction of fast neutrons), but use of a beryllium oxide moderator can soften it again and prevent adverse effects on the Doppler coefficient. But the larger core can make fuel changing more difficult. More troublesome than any of these, are problems of greater complexity in control for the various core regions.

(2) Various mixed fuel cycles are under discussion using, for example, U^{233} and Th^{232} in part of the core.

Of course, these are not new thoughts and several studies consider possible schemes. For example, Combustion Engineering (1970) claims that increasing neutron leakage by a modular, annular, or extreme pancake design would result in a higher fuel cycle cost and more expensive mechanical design. There might, with some designs, be a higher breeding ratio, but a larger fuel inventory and therefore a longer doubling time. The lower capital cost may not be critical if the first few breeder reactors are subsidized, and the short doubling time is important only to provide a rapid increase in the number of self-fueled breeder reactors. On the other hand, any increased safety margin may be useful, particularly for the first few years of the technology, but the greater inherent (physical) safety advantage has to be compared with the engineering safety disadvantages (these could arise, for example, by larger temperature inhomogeneities and consequent stresses over the core).

The possibility of such a core design is being examined as an alternate fuel management scheme for CRBR. This type of core is colloquially called a radial parfait core.

F. Clad coefficients

The fuel is clad in stainless steel. If an accident occurs such that the sodium coolant boils and the coolant channel voids, then the stainless steel clad will melt and vaporize soon thereafter. If the clad leaves the core, this will also change the reactivity. This could be important if the clad leaves the core ahead of the fuel. The change will come from the same causes as the change from sodium voiding and will be two to three times as big as the change due to the sodium coefficient; since the sodium and the clad occupy similar places in the core, the clad coefficient will change with reactor geometry in a way similar to the change of the sodium void coefficient with reactor geometry, although the greater absorption makes some difference. A reactor core design for a low sodium void coefficient will also be a design for a low clad coefficient. The relative contributions of clad and sodium will, of course, change with thickness of clad and size of coolant channel. If more advanced fuels can have thinner clad and can be shown not to swell with the extensive reactor irradiation, the size of the coolant channels may be reduced somewhat, reducing the clad and void coefficients. This may depend upon development of special clad materials.

Even for an early design a small channel size and thin clad might be possible if the maximum operating temperature were reduced. This would reduce the positive void coefficient, but at a cost in thermodynamic efficiency of the power station, and there would be a *shorter* doubling time.

G. Range of reactivity control

The necessary range of reactivity control by control rods in an LMFBR is less than that required for a light-water reactor, because the absorption of neutrons by fission products (whose number changes with time) is less important for a breeder, and the change in the fraction of fissile fuel is less. This can lead to a safety advantage, in that the maximum accidental reactivity insertion is reduced. However, as already discussed, stability of operation is attained by designing the reactor with negative coefficients of reactivity with temperature. Then there is still an appreciable excess in reactivity which must be corrected as the reactor is started up from a cold condition.

This suggests that the most desirable curve of reactivity vs temperature should be as shown in Fig. 6(a). Unfortunately, this is not achievable, and a more common distribution is shown in Fig. 6(b).

These reactivity coefficients are nonetheless important. In a general balance between safety, high breeding ratio, and cost, we do not want to increase the Doppler

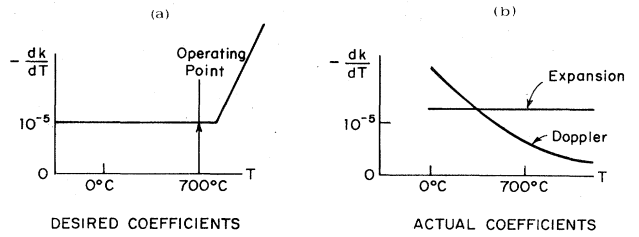

FIG. 6. A schematic showing the desired and actual reactivity coefficients.

coefficient without limit. However, the *mechanical* expansion coefficients are also effective while reactivity changes are relatively slow compared to thermal structural time constants.

The large expansion coefficients obtained in the flared core design of PHENIX and the Creys–Malville design delay certain slower transients even further. Prompt critical reactivity is delayed nine minutes, and these nine minutes might be enough to allow some form of corrective action to be taken.

Unfortunately a design allowing flaring may have other drawbacks; bowing of fuel pins is enhanced and the reduced clamping might allow compaction by radial relocation (and hence a major reactivity increase) in the event of an earthquake.

H. Comparison of LMFBR with light-water reactors

There are several small safety issues, but the one of concern in this paper is the hypothetical possibility of a sequence of events leading to an energetic reaction which, if severe, could lift the lead of the reactor vessel, crack the containment, and spread the radioactivity. At this point it is useful to pause and compare the LMFBR to the light-water reactors now being installed all over the world, and for which we have detailed safety and probability assessments (Lewis *et al.*, 1975; Rasmussen, 1975). This comparison is important because it will enable us to use our accumulated knowledge and experience and the intuition derived therefrom in the best possible way.

In Table III we list some important differences. First, the LWR operates with the moderator and coolant at high pressures. Therefore, a loss-of-coolant accident (LOCA) can result from any failure of the primary coolant system. The reactor shuts down at once, but generation of decay heat continues. If the core remains uncooled, there would be a core meltdown and possible subsequent release of much of the gaseous fission products inventory, but relatively few of the solid ones. All accidents, to be of significance to the environment, must include significant fuel melting or damage.

On the other hand, the primary coolant circuit in an LMFBR is not pressurized. A catastrophic "double-ended guillotine break" seems less probable, and both this and smaller breaks would have less severe consequences since the coolant would not automatically boil and rapidly disappear.

In a light-water reactor, expansion of the coolant tends to shut down the reactor, because the coolant also acts as a moderator, and possible eventual removal of the coolant shuts it down completely. In an LMFBR that is not the case. In most current designs, the reactivity increases as the sodium coolant is removed, making a bad problem worse. Moreover, if the fuel melts, it is possible that it slumps into a more compact lump which is more reactive. In a light-water reactor, shutdown, if not effected by control rod insertion, takes place by removal of moderator (coolant). In an LMFBR, if not effected by control rod insertion, shutdown must take place by removal of fuel. Fuel melting may or may not lead to removal (depending on circumstances), so that vapor pressure may be necessary to ensure fuel removal. This vapor may be sodium vapor, fission gas pressure, or vapor of the stainless steel cladding, all of which are present at the melting point of the fuel. But if these are not adequate, the fuel itself might have to heat up and vaporize to disperse the fuel.

A major difference exists in power density; for an LMFBR a high power density is important to minimize the fuel inventory and hence the investment. Also, a high power density will mean a high rate of production of plutonium for a given amount of fuel. For a given breeding ratio, this leads to a short doubling time and therefore a high possible rate of penetration of the market by breeder reactors. For a light-water reactor, the power density affects primarily the initial investment. The power per unit weight of fissile material is about the same in both reactor types; but the light-water reactors have more nonfissile material in the core, leading to a lower average power density.

Because of the possibility of vaporizing part of the core, an LMFBR has been thought of as potentially the most dangerous reactor type. This possibility is not

TABLE III. Important differences between reactor types, ^{235}U and Pu fuel concentrations.

LWR (Pressurized PWR or Boiling BWR)	LMFBR
Operates with coolant under pressure	Operates with coolant at atmospheric pressure
Removal of coolant shuts down reactor	Removal of coolant may increase reactivity
Compaction of core decreases reactivity	Compaction of core increases reactivity
Power density ≈80 W/cc	Power density ≈400 W/cc
Post accident heat removal needs ECCS	Post accident heat removal from surrounding sodium

unique. It has, for example, been postulated that a control rod could be rapidly expelled from the core in a light-water reactor [as probably happened in the SL-1 accident after an initial manual withdrawal of a few inches (Thompson, 1964a)], and lead to partial core vaporization also. If the possibility of vaporization can be shown to be very remote, the breeder reactor is one of the safest of reactor types (Farmer, 1970).

I. Lines of assurance

It has become conventional in the United States to talk about four *lines of assurance* (LOA) (Hannum *et al.*, 1975) against severe accident consequences. The lines of assurance are lines separating steps in an accident sequence that are more or less distinct, so that the probability of a severe consequence may be approximated by a product of the probabilities of breaching each line of assurance. In the course of this paper, the lines of assurance will become clear. It is desirable to stop an incident as early in the chain of events as possible (CGUS, 1975); as an accident progresses its details become less and less clear and a rigorous proof of safety less and less easy.

The first line of assurance is to prevent any incident leading to a melting of the reactor or its fuel. To do this we must detect any malfunction early, and shut down the reactor promptly. This involves reliable sensing devices, reliable shutdown rod insertion (with an alternate shutdown mechanism of a different type), and also careful design to ensure that any accident initiation proceeds slowly enough for these devices to take effect. We shall see that it is now possible to design a system with a failure rate of 10^{-5} or 10^{-6} per year.

The second line of assurance is to ensure that if there is a meltdown of one subassembly, the damage can be prevented from spreading to the whole core. There was a partial meltdown in the Fermi reactor at Laguna Beach in Monroe, Michigan, but damage was confined to a few subassemblies. I will not dwell heavily on this line of assurance, although the damage in the Fermi reactor accident might have been reduced if adequate instrumentation had existed to detect single-channel flow blockage, because there exist some postulated accident initiators that are believed to proceed to whole core involvement if reactor shutdown does not occur.

The third line of assurance is to ensure that even with whole core involvement (melting) there would be no significant mechanical energy release, so that all fission products would be contained within the reactor vessel itself. A discussion of this is the primary purpose of this paper.

The fourth line of assurance is to ensure that releases of radioactivity from the reactor do not lead to excessive radiation hazards to the public.

J. Capability for shutdown (or scram)

The major safety concern of LMFBR designers is to avoid breaking the first line of assurance and to prevent any transient which leads to even partial core meltdown. If core meltdown can be prevented, the reactor itself will remain in operating condition for the future. Since any meltdown is liable to prevent plant operation for many years, there is a clear economic incentive to prevent meltdowns. Only a small fraction of meltdowns are likely to affect public safety, so that this economic incentive is comparable to the safety incentive even though the maximum consequence to public safety is greater.

A breach of the first line of assurance can be prevented by ensuring that any malfunction can and will be immediately detected, and that shutdown rods will be promptly inserted.

The shutdown mechanisms that have been designed for LMFBR's are fast enough for any transient that has been identified as a possible transient. Moreover, many design features exist to slow transients. Inherent physical features include the existence of delayed neutrons with their long time constant for small transients, and various temperature coefficients of reactivity. Engineered features include mechanical constraints (see next section). Therefore there are some tens of seconds for operation of shutdown devices, which allows a large margin.

Malfunctions can be detected in many different ways — overpower of the reactor as a whole; overtemperature, either of the reactor as a whole or a subassembly; and unusual reactivity or flow fluctuations (noise) which might arise, for example, from flow changes and a local coolant blockage in one subassembly.

Some LMFBR's have operated with considerable failed fuel, that is, fuel where the cladding fails, and this was done deliberately in the BR-5 in the U.S.S.R. for test purposes. But it is important to be sure that if there is a spontaneous failure of a fuel pin, and fuel could be released into the coolant area causing a blockage, the reactor can be promptly stopped before appreciable core damage occurs. The Run–Beyond–Cladding–Breach program is aimed at providing limits on failed fuel operation and detection to avoid this situation. Presently conservative operating limits do not allow such a situation to occur.

Shutdown is accomplished by rapid insertion of control rods; most reactor designs have two separate sets with separate operating mechanisms.

At one time a self-actuating overtemperature device (a fuse) was planned which automatically allowed a few subassemblies to drop out of the reactor if overheating occurred. Difficulties occurred with the reliability of these devices, and this approach is not used in any existing designs. No mechanically automatic shutdown mechanism now exists in any operating reactor.

In the PHENIX reactor at Marcoule it is reported (Zaleski, 1975) that a call for shutdown occurs once or twice a year in operation. Therefore, for an overall accident probability of less than one in 10^5/year, there should be a failure rate of less than one in 2×10^5 calls. The designers of Clinch River Breeder Reactor believe this can be maintained (Graham, 1974). Discussions of the various methods are in Smith (1970), Graham (1971), and American Nuclear Society (1974).

A reliability of this order is estimated for shutdown of light-water reactors. Failure rates of between 10^{-4} and 10^{-5} per challenge have been estimated in AEC (1973) and quoted by Rasmussen (1975). More recent estimates bring this down to between 10^{-6} and 10^{-7} (Fullwood *et al.*, 1976; Vessily, 1977). In addition, the rate of progression of

accident conditions is slow enough that 10 minutes is available in a LWR to shutdown manually by injection of boron poison, leading to an overall probability of failure to shut down less than 10^{-6} per challenge.

It seems probable that this could be improved, both for a LWR and an LMFBR, but since it seems impossible to have an experimental test of 10^6 shutdowns, proof of such a reliability is indirect and may not be acceptable (Yellin, 1976). It appears particularly important to make sure that the initial stages of an accident proceed slowly, so that we may take account of these alternate shutdown probabilities which are disparate enough that common mode failures are less likely.

Since there has been a partial meltdown of one reactor in the U. S. (Fermi) and a complete meltdown of another (EBR-1) (Thompson, 1964b), it is important to demonstrate clearly the improvement over the early situations. The Fermi reactor meltdown was caused by a partial coolant blockage and was tripped manually and not automatically. It seems likely that improved sensing methods will detect these partial blockages, but this needs careful evaluation. On the other hand, both Fermi and EBR-1 used metallic fuel with an expansion coefficient which provided a large negative reactivity coefficient. The shutdown reliability does not depend upon physical principles, but upon engineering principles in which experience plays a major role.

The EBR-1 meltdown was during an experiment with all automatic scrams deactivated. Moreover, EBR-1 had a positive prompt (fuel bowing) reactivity coefficient EBR-1 did not have an appreciable negative Doppler coefficient of reactivity, although it had a negative delayed coefficient of expansion. Fermi had a small Doppler coefficient of $-1.6\times10^{-6}\Delta k/k/°C$.

It is possible that work on a particular reactor design will demonstrate a small enough probability of failure to scram that further lines of assurance become unnecessary. It is noteworthy that the U. S. Nuclear Regulatory Commission (NRC, 1976) admits this possibility by suggesting that a core melt need not be considered to be a design basis accident for the Clinch River Breeder Reactor. Such probability assessments must include the possibility that a plant can be sabotaged and redundant shutdown systems put out of action simultaneously. Probabilities of sabotage are hard to discuss and very uncertain. But diversity of mechanisms and diversity of approaches to safety can much reduce the probability of effective sabotage.

III. ACCIDENT INITIATION LEADING TO POSSIBLE CORE DISRUPTION

A. Transient undercooling

I will now proceed to discuss two major classes of accident sequences with the aid of Figs. 7, 8, and 9, transient undercooling (TUC) or loss of flow (LOF), and transient overpower (TOP). Transient undercooling can occur, for example, if power fails to all the sodium pumps at once. Then the coolant flow will slow. This should be immediately sensed and scram initiated. If there is a failure to scram, the temperature will rise. Various reactivity coefficients will act to reduce reactivity, but the power will, in general, not be reduced

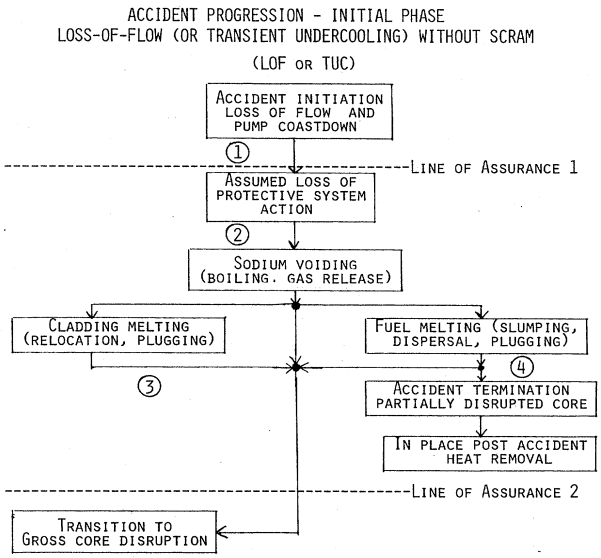

FIG. 7. Accident progression—initial phase; loss of flow (or transient undercooling) without scram.

fast enough to prevent boiling of the hottest sodium channel. This boiling will result in voiding of the channel, increasing reactivity, and a power increase in the reactor. This could cause fuel to fail in cooler channels containing sodium, and the resulting sodium ejection is postulated to initiate a prompt critical excursion of short period leading to a core melt. Such a progression is sometimes called LOF-driven TOP.

Various mitigating steps have been suggested. The pumps in the primary circuit are unlikely to all seize and stop instantaneously and simultaneously. There will be a coastdown and a reduction in flow. This coastdown can be made to be long by the addition of flywheels. A flared core gives a large negative temperature coefficient at this stage, and calculation shows that pump trip without scram will *not* lead to sodium boiling in the PHENIX reactor at Marcoule, and will delay it for nine minutes in SUPERPHENIX at Creys-Malville. This can allow time for manual action in place of the (assumed failed) automatic shutdown mechanism. As noted in the section on scram probability, this could be a major step in the credibility of maintaining the first line of assurance.

There are operational problems with a high inertia pump. In any controlled shutdown the pumps would undercool the reactor and significant thermal gradients would be introduced in various components. This may lead to premature failure and a net reduction in safety.

A variant of the transient undercooling is local coolant blockage in one channel (space between fuel rods) by some obstruction or by failed fuel where the fuel clad-

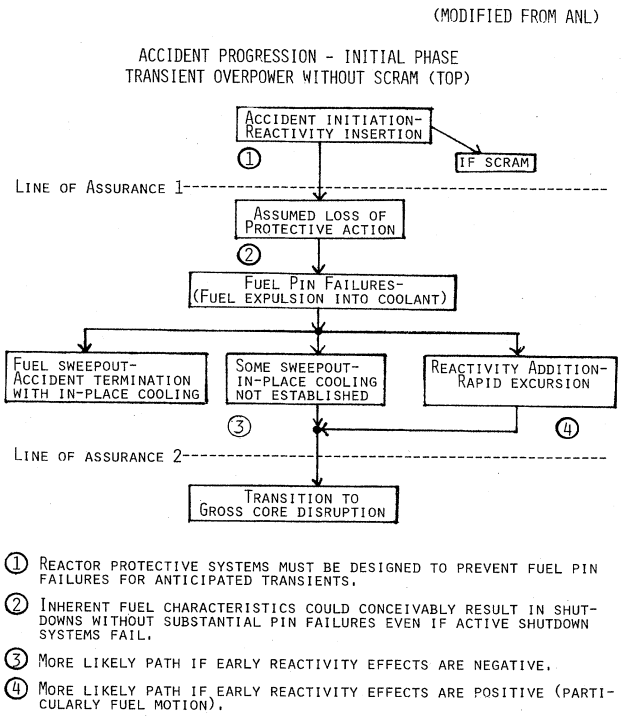

FIG. 8. Accident progression—initial phase; transient overpower without scram.

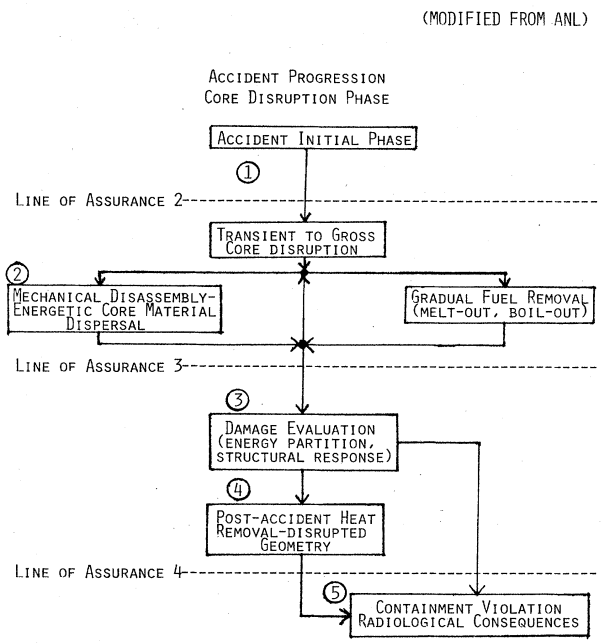

FIG. 9. Accident progression—core disruption.

ding fails and fuel and clad block individual channels. If there is a detector for local flow blockage, the reactor can be shut down. In the Fermi reactor an inlet blockage was not identified even though overtemperature signals were received. This led to local flow starvation, local sodium voiding, and local fuel melting. In the Fermi reactor manual scram occurred before there was any whole core involvement. There are now better designs to prevent obstruction of individual channels.

This type of accident might well be limited to one or two subassemblies if the second line of assurance operates. At the Fermi reactor this was the case, and it is more likely to be so for reactors with a larger negative Doppler coefficient. But the mechanisms involved seem to be a subset of those for whole core involvement, so I will not consider them further here.

B. Transient overpower

The transient overpower (TOP) (Fig. 8) accident could be initiated, for example, by some control rod drive failure which removed one rod. The control rod drives are mechanically designed (ratcheted) so that they can be detached to go in quickly, but can only be removed at a limited rate. Therefore, the rate of change of reactivity is limited to a design rate (for CRBR) of \$0.025/sec. Nonetheless safety engineers have assumed \$0.10/sec in order to be conservative.

If scram does not occur, the power will increase beyond the ability of negative reactivity coefficients to control it, because in CRBR, for example, withdrawal of one rod completely can lead to a reactivity addition of \$3.2 (in the usual jargon we say that the "worth" of this rod is \$3.2).

Naively we might think that at \$0.025/sec the reactivity increase would be \$1 after 40 sec and the reactor would be prompt critical. But for a transient starting from normal power and temperature the Doppler and other negative reactivity coefficients slow down the transient considerably.

C. Core disruption: General

If one of the two major postulated accident events—transient undercooling (TUC) or transient overpower (TOP)—occurs *without* scram, the only ways of shutting down the reactor are by removal of fuel or increasing leakage or rearranging fuel. In the TOP accident the first is most important. Since I have postulated the failure of other shutdown mechanisms, the question arises, how fast will this fuel be removed?

If the fuel could be removed instantly, or at least as fast as the coolant and the clad remove themselves, any reactor transient could be corrected rapidly. But there is inertia in the fuel motion. If reactivity increases much faster than inertia allows it to be removed, there will be a hydrodynamic disassembly.

If reactivity only increases slowly, there can be a benign reactor shutdown by fuel sweepout. It seems intuitively obvious—and calculations support intuition—that the slower the rate of increase of reactivity (ramp rate), the more likely there is to be a benign disassembly. The Doppler coefficient is therefore of great

importance because it can slow the development of the transient.

In the initial stages of a core melt the reactivity increases due to one or another of the following: sodium boiling and voiding; clad melt and motion; and (possibly) fuel falling into the center (slumping); reactivity decreases as the molten fuel is driven out of the center of the core to positions where it contributes less to the reactivity. Will the reactivity decrease fast enough to shut the reaction down? Elaborate computer models (SAS3A) have been constructed to describe this behavior. Calculations for CRBR for example (Bohl, 1975) show that even with the conservative bounding transient assumptions discussed earlier, there will be a low energy release. But these calculations are complex and it is important to check the details. In the next few sections I will describe the physical phenomena. It would be fortunate if there existed a general physical principle in addition to reactor shutdown which could be counted upon to limit the energy release. Such a principle might be sought in the intuition that a self-heated system will push itself apart because of the pressures generated by the heat (Fauske, 1975; 1976). Reactor fuel will remain internally heated for some time because of the fission products (decay heat). In addition, fission product gases, sodium vapor, and stainless steel clad vapor can keep the fuel apart. Unfortunately we do not yet know how to put numbers to this general principle—some more experiments may enable us to do so. The very complexity and size of the SAS3A code make its verification hard. In this connection, it is very important that another computer code (SIMMER) is being written using different calculational principles. They should at least check each other, even though the phenomena they calculate are not entirely the same.

D. Fuel failure models

In a transient undercooling accident (TUC) without scram the fuel continues to heat up until a channel voids and fuel melts, even though reactivity is decreasing due to the temperature coefficients. In a transient overpower accident (TOP) without scram the heat-up is due to, and therefore accompanied by, a reactivity increase. In each case the subsequent events are similar, although the order of channel voiding and fuel melting may be inverted.

After sodium boils and voids and increases reactivity, the cladding on the fuel pins will melt and fuel can come out. The important question arises: *does this pin failure lead to an increase or decrease in reactivity*? If a decrease, it might compensate for the reactivity from the sodium voiding; if an increase, it makes matters worse.

If the fuel pin fails near the midplane of the core, the molten fuel within the pin, possibly propelled by fission product gas pressure, may leave the ends of the pin for the middle before escaping from the clad. The increased fuel concentration will probably cause reactivity to increase. If the pin fails above the midplane, the fuel is likely to be swept out vertically and reactivity will decrease. Gravity has only a second-order effect compared to fission product gas pressure and the normal

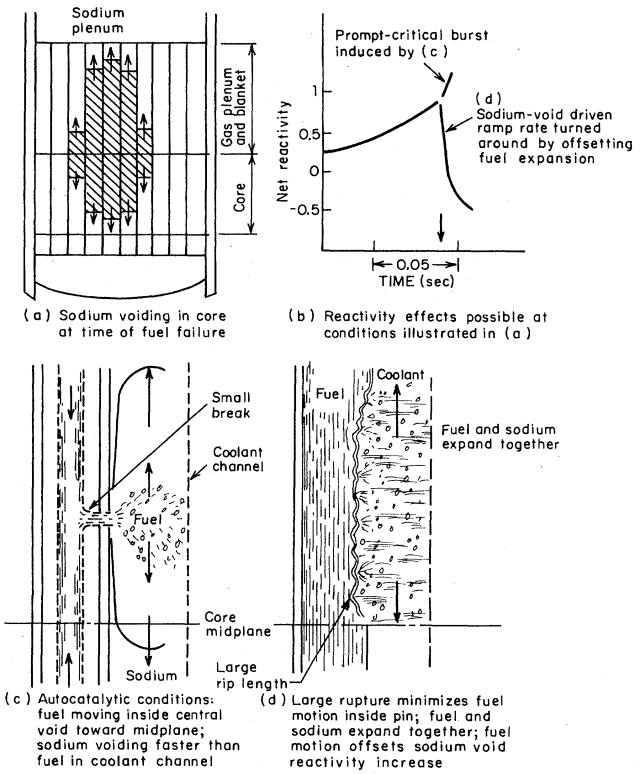

FIG. 10. Illustration of fuel motion possibilities as a consequence of a large positive sodium void coefficient (from Fauske 1976).

direction of flow of the coolant.

The center of the fuel pin is at the point of highest flux, but the hottest part of the fuel is above the center because the coolant flow is upwards. The maximum strain on the fuel pin is at the center, and the hottest part of the cladding is above the center. Therefore we might expect fuel pins to fail above the center, and molten fuel, if any, to proceed out of the median plane to a point of lower reactivity and eventually out of the reactor. This would be a hydraulic (and benign) disassembly.

The vaporization temperature of the stainless steel cladding is close to the melting temperature of the oxide fuel (Table I). This then allows the steel vapor to act as a mechanism for immediate dispersal of the fuel and for prevention of fuel slumping. In the distant future we may have carbide fuel which melts at a lower temperature than the boiling point of the clad; there would then be a greater likelihood of an initial fuel slumping. These various fuel motion possibilities are illustrated by Fauske (1976) and reproduced as Fig. 10.

E. Fuel pin failure: Experiment

Various experiments have been performed on failures of fuel pins. The most important ones are in the TREAT pulsed reactor facility which simulates the power pulse in a postulated LMFBR accident.

Experiments with single pin failures in the TREAT test facility suggest that fuel failures occur above the midplane (Dickerman *et al.*, 1975). However, a model cal-

culation (Kastenberg and Catton, 1974) shows that after there has been some radiation to cause fission products, but not yet at the end of the core life (end of equilibrium core or EOEC), failures will occur near the middle; other models disagree (Bohl *et al.*, 1975). The failure location is clearly determined by many factors—ramp rate, fuel microstructure, fission product distribution, clad structure—about all of which it is almost impossible to be certain. It seems probable, for example, that in a shorter pulse the failure will occur closer to the midplane as the initial temperature of the cladding becomes less important. Since the TREAT pulse is a long one, the experimental results with TREAT may not be representative of postulated accident conditions. Future facilities are planned which may better bound this problem. The best we have now is a range of possibilities, but calculation can sort out the important ones and find the worst case, which at the moment still appears to be a benign disassembly.

The accident analysis will now depend on whether these experiments are valid and calculations are correct. In this case we will get shutdown following fuel sweepout (*hydraulic* disassembly), which is a *benign* or nonenergetic process, and in the other case—that a real accident situation is far worse than experiment or calculation—we will have a large enough ramp rate to initiate the hydrodynamic disassembly. For safety purposes, we make a nonmechanistic assumption (i.e., one for which no consistent mechanism is known) and add the ramp rates from sodium voiding and fuel movement. It obviously takes more fuel motion to compensate a large positive void coefficient than a small one, and until field motion is more certain, a small positive void coefficient may be desirable. But understanding based on further calculation and experiment may allow removal of this highly pessimistic assumption. (I note that the detailed calculations will give step changes and even reversals of reactivity as the sodium void and the fuel reactivities can enter at different times, but this is different from the oscillatory effects noted before.)

I also note that fragmentation of the fuel on leaving the fuel pins may occur as the fuel is swept out. This is one of the necessary (but not sufficient) conditions for explosive fuel/coolant *interactions* to occur (see later). This illustrates the strong couplings among all the mechanisms involved.

F. Maximum rate of reactivity insertion: Fuel slumping

As noted above, fuel sweepout may well occur, but is hard to prove. In the most pessimistic case consistent with the data, the fuel could fail near the midplane and the flow of molten fuel to the center of the fuel pin could increase reactivity. This is unlikely to happen with all fuel pins simultaneously, and as we shall see, it is the *rate* of reactivity insertion, if continued over the duration of the accident, that is important.

G. Maximum rate of reactivity insertion: Recompaction

Another particular mechanism for a large reactivity insertion has been identified over the years. After an initial fuel dispersal we can envisage, although no calculation or experiment shows it, that as the fuel pins fail by core melting, half the fuel drops into a molten pool and the other half is swept upwards and solidifies. The sodium already has been removed. A moment later, the solid fuel falls by gravity onto the molten fuel below. As the fuel hits the pool, the assembly once again becomes critical, with a high rate of reactivity insertion. It is well known that if a subcritical mass of pure ^{239}Pu is imploded rapidly enough, a nuclear explosion occurs.

The only suggested implosion mechanism in a reactor is the uranium/sodium interaction. Qualitatively we expect (and calculation confirms) that the rate of pushing apart in a nuclear reaction is less as the rate of assembly is reduced. Moreover, of course, bombs do not have a large negative Doppler coefficient, they explode before delayed neutrons influence the reaction, and they have fissile material which is more concentrated both chemically and physically than reactors, although there is less of it.

In the case considered here, the rate of reactivity increase can be calculated on an elementary basis as follows: the velocity after free fall over a distance s, assuming, very simplistically, no fluid damping, is given by $v=\sqrt{2sg}$. For a 20 cm fall it is 200 cm/sec. The reactivity for this assembly can be calculated as a function of distance. In an initial disassembly accident, only the central part of the core is likely to be hot enough to melt. For the 19 central subassemblies in the Clinch River Breeder Reactor (CRBR), the maximum ramp rate of reactivity insertion by recompaction is \$30/sec. Such a calculation of course neglects other forces limiting assembly due to fuel dispersal from fission gases or even from thermal energy generated in the fuel.

To be sure, this could be increased if we assume an unrestrained fall larger than 20 cm, or that some unknown driving force gives a greater acceleration than gravity (driving forces such as a hypothetical explosive fuel-coolant interaction would be expected to push the fuel *out*, not *in*, unless a pressure wave is reflected from the pressure vessel). No plausible mechanism has been suggested for this large a reactivity insertion by compaction, even with an explosive fuel-coolant interaction. Further thought and experiment may be able to prove this rigorously and settle the issue.

The French argue for their Creys–Malville analysis that if the core is unrestrained in its fall, which gives the high reactivity insertion of \$30/sec, then there can be no sodium present. Then in the subsequent disassembly, no sodium vapor would be present for the later conversion of thermal to mechanical energy.

It could be argued in principle that the fuel in more than 19 subassemblies separates and recompacts in this way. However, we cannot even find a plausible scenario for these 19 to recompact starting from an initial event. There is considerable power peaking over the reactor core, and the outside of the core, particularly in the fertile blanket, is expected to remain solid. This would apply even if (as the French assume for Creys–Malville) the initiating melting was a disruptive disassembly with a small (\$1/sec) ramp rate; only the central part of the core would have melted before disassembly. It may be possible to rule out the existence of large solid masses of fuel by the dispersal mechanisms above, and also

rule out the possibility of large reactivity increases as they come together.

Increases in reactivity due to collapse of a dispersed sodium fuel mixture have been suggested (Boudreau and Jackson, 1974). These could be enhanced by an explosive fuel-coolant interaction as suggested later. Detailed calculations have not yet been performed on this mechanism from which we might derive a larger ramp increase than described below. This is one of many innovative suggestions that have been made, and most have been disproved. If more years pass and, in spite of diligent search, no mechanism has been suggested for a recompaction which cannot be disproved, even the more imaginative critics will regard as more likely the hypothesis that recompaction is impossible.

H. Coolant voiding

As noted above, in most reactor designs, the reactivity increases as each channel boils in turn, unless fuel is swept out with the sodium. At first, only the hottest channel will boil; in a TUC accident the voiding will start from the top of the channel, and initially the reactivity insertion will be negative, followed by voiding from the middle of the channel with a positive reactivity insertion which will more than compensate the negative contribution from the top boiling. The next hottest channels will boil soon, and then the next within an even shorter time. Then the reactor may become prompt critical and heat up rapidly. Since the reactor has cylindrical symmetry, there are more subassemblies in each ring of increasing radius. Therefore, a rapid increase in reactivity can occur. If the sodium boils in many channels simultaneously, boiling is said to be coherent, and the increase in reactivity insertion is very rapid. The situation is similar for a transient overpower accident without scram (TOP with a large ramp rate); here, however, the coolant voiding might start in the middle of the fuel pin.

The existing mechanistic calculations of coolant voiding show that these voiding rates are not very coherent. These calculations also include modelling of the clad and fuel motion—which is occurring at the same time in the accident progression.

Critics argue that after an initial voiding of the central channels the ramp rate may be so increased that perhaps the remaining half of the channels are voided coherently. At the same time, internal pressures may decrease the natural time for voiding one channel from 1/10 sec to 1/40 sec. For CRBR, voiding one-half of the core (but none of the blanket) in 1/40 sec would give a ramp of \$70/sec. If the fuel were slumping at the same time, there would be an *additional* ramp of \$30/sec for a total of \$100/sec. This total is containable with the CRBR design (Bohl, 1975) (with no large fuel coolant interaction).

For the Creys–Malville design, the sodium void coefficient for the whole core is about twice as big as for CRBR. For some 1000 MWe designs the total sodium void contribution to reactivity is as high as +\$10. At the same rate of voiding, a ramp rate of excursion of \$250/sec is derived, which would lead to energy releases (2000 MJ mechanical) which may be impractical to contain.

Ramp rates due to sodium voiding alone of \$250/sec for a 4000 MWe LMFBR were calculated from a postulated TUC accident initiator by Bleiweis *et al.* (1974). This calculation using the SAS code is in the direction of the very crude argument above which causes concern. These high ramp rates were calculated before the postulated fuel failure added clad relocation and fuel relocation.

Using a more recent version of the SAS code, Hummel *et al.* (1976a) found a maximum ramp rate of \$30/sec from sodium voiding even for a 4000 MWe reactor design with a \$7 sodium void contribution. This more recent code includes a self-limiting effect of sodium voiding due to pressure buildup in the lower plenum. This seems a well defined physical phenomenon not dependent upon details. But with this self-limiting feature removed from the calculation, Hummel found a maximum ramp rate of \$60/sec in comparison with the \$250/sec of Bleiweis. The calculations of Bleiweis have not been confirmed, and it is probable that they are incorrect. If so, there remains no computational evidence for large ramp rates from sodium void.

Hummel *et al.* (1976a, 1976b) performed similar calculations for CRBR and found a maximum voiding rate of \$25/sec; they suggested that a loss-of-flow accident *would not* lead to prompt critical conditions with a possibility of subsequent large energy releases.

These calculations assume some coherence between sodium voiding in the different channels because the computer code (SAS3A) had only 10 channels into which the various subassemblies must be grouped. It would seem that the calculated reactivity insertions would be smoother if more channels were assumed, and the Doppler and other feedback mechanisms would take effect in a smoother way.

Ideally it is desirable to plot the calculated details (temperature rise, ramp rate of reactivity, or whatever) against the number of channels and to increase the number of channels until further increase makes no difference. This cannot yet be done directly, but in a very ingenious calculational scheme, Massey *et al.* (1976) shuffle the different subassemblies between different code channels as the postulated accident proceeds and obtain the effect of more incoherence. The calculations suggest substantially milder super prompt critical conditions. If confirmed, this also would suggest that present safety analyses are unnecessarily conservative. A more recent version of the SAS code (SAS3A) has 34 channels, so that this confirmation (or possible denial) may be imminent.

In view of the importance of the incoherence of sodium boiling, it would be helpful to have some direct experimental evidence of the incoherence over a large scale to be sure that no important physical effect tending toward coherence has been forgotten. Experiments which might produce such evidence would be very hard to perform with sodium in a real reactor.

It is perhaps worth noting that a reactor with a very inhomogeneous core—such as the radial parfait cores—not only has a reduced void coefficient but also has enough temperature inhomogeneities that there will be

incoherent sodium boiling, further reducing the possible ramp rates of reactivity.

I. Sodium superheat

We noted that if the channels were to void all at once, a very fast increase of reactivity would occur, with a consequent increase in damage potential. One obvious possibility for such rapid voiding is that of sodium superheat. If sodium in the first channel is superheated several hundred degrees, the boiling (and voiding) could be delayed. Then the start of boiling in one channel might trigger simultaneous boiling all over the core.

In laboratory experiments with pure materials (without nucleation sites), superheats of several hundred degrees have been obtained. In real life, there are plenty of nucleation sites such as surface roughness, bubbles, and small impurities. It was stated at the conclusion of a conference on fast reactor safety that: "Currently it does seem to be agreed that, in the conditions and purity and gas entrainment likely to be found in reactors, superheat would be limited to a few tens of degrees at most and could well be less" (Gilby, 1970). Thus an early worry that the rapid vaporization of superheated liquid on a time scale of milliseconds, which could give very high ramp rates of reactivity addition, is now discounted (Fauske, 1972).

Although the amount of superheat may affect the detailed mechanistic calculations, once a core disruption is assumed, superheats up to 100 °C are calculated to have little effect on the energy release (Bleiweis, 1974; Hummel *et al*, 1976b), whereas the superheating is believed to be only 5° or 10 °C. The superheat problem is therefore considered to be unimportant.

IV. CALCULATION OF THE HYPOTHETICAL CORE DISASSEMBLY ACCIDENT (HCDA)

A. Introduction

I now proceed to the hypothetical case in which a large ramp increase of reactivity occurs, in spite of the fact that no mechanistic calculation starting from the initiating event predicts it, and in spite of a commonly held belief that none will be found.

In these circumstances, can the Doppler coefficient by itself stop an accident? In a first calculation to gauge the scope of the problem, we ask the question, "can we compensate for complete sodium voiding *and* for reassembly of the fuel into its most reactive configuration by raising the temperature and no other action?" We see the answer is "no" as follows:

Complete sodium voiding can add \$5($\Delta k = 0.016$) to reactivity, and fuel rearrangements \$20($\Delta k = 0.060$). This corresponds to $+\Delta k = 0.08$.

The Doppler coefficient is perhaps $A = T dk/dT = -0.003$ without Na, and -0.005 with Na. Then the temperatures are given by $\Delta k = -0.005 \ln(T_f/T_i)$, where T_f and T_i are the final and initial temperatures of the fuel. Here $T_f/T_i = \exp(0.08/0.005) = \exp 16$. If $T_i = 800$ °C, $T_f = 7$ billion degrees. This isn't even a proper upper limit because the Doppler effect goes down at these temperatures, since all resonances overlap and the capture cross section becomes independent of the temperature. Long before these temperatures are reached, something else must happen.

B. Calculational procedure for thermal energy release

If there is an excess reactivity, which has not been compensated by shutdown rods, and which is too great to be compensated by reactivity feedback from Doppler effect or core expansion (at any reasonable temperature), then fuel must remove itself. If this does not happen in a slow, benign way, then energy will build up until the core is pushed apart. Even so, this is not an explosion in the formal sense; the times involved are, as we shall see, 10 times longer than in a chemical explosion, and the expansion will be slower than the velocity of sound.

The first calculations of this were by Bethe and Tait (1956) who used a homogeneous hydrodynamic model. Shortly thereafter a rigorous weapons explosion code was used for the problem, which showed a smaller energy release (Stratton, 1972). The Bethe–Tait model has been extended by Nicholson (1962), Wolfe *et al.* (1963), and Hicks and Menzies (1965); various computer codes (Stratton *et al.*, 1972; Jackson and Nicholson, 1972) make the results more precise. The basic calculational procedure is very soundly based from a physical point of view. The sound basis of the calculation is a theme of an excellent review by Meyer and Wolfe (1968). See also McCarthy and Okrent (1964).

Here I will go into a treatment which is very approximate to illustrate the physical process. I will assume a small step reactivity insertion. The Doppler reactivity feedback will tend to compensate the reactivity insertion. I will deduce the temperature increase in this small step excursion as in the previous section. An excursion of $\Delta k/k = 0.0003$ above prompt critical would be matched by Doppler feedback (with a coefficient of -0.003) with $T_f/T_i = \exp(1/10) = 1.1$. But this is not all the increase; there can be an overshoot above this value. This is because the increase in reactivity allows the power to increase immediately—and the energy density and Doppler feedback increase is delayed. To demonstrate this process, we neglect delayed neutrons and cooling because the excusion is very rapid and we have the basic equations:

$$\frac{1}{P}\frac{dP}{dt} = \frac{k-1-\beta}{l},$$

$$C\Delta T = \Delta Q \quad \text{or} \quad C\frac{dT}{dt} = \frac{dQ}{dt},$$

$$P = dQ/dt,$$

$$T(dk/dT) = A, \tag{8}$$

where Q is the energy density in joules/gm, P the power density $= dQ/dt$ (W/gm), C is the heat capacity in gm/°C, l is the neutron lifetime in seconds (about 5×10^{-7} sec).

These equations have a solution that is close to being oscillatory. We see this by setting the Doppler coefficient to be independent of temperature

$$\frac{dk}{dT} = \frac{A}{T_0}.$$

Then

$$\frac{d^2k}{dt^2} \approx \frac{A}{T_0}\frac{d^2T}{dt^2} \approx \frac{AP_0}{CT_0 l}(k-1-\beta). \tag{9}$$

Since A is negative, this has an oscillatory solution: the oscillation frequency $\sqrt{P_0|A|/CT_0 l}$ is about 500 cps for a typical reactor. But this approximation is not valid; a more realistic solution gives an overshoot, as does the oscillatory solution, to about twice the temperature rise of a static equation. After this the reactivity returns to, and below, prompt critical.

The total energy release Q is found as follows: Assuming an initial step Δk_0 and an initial condition at a low energy and temperature

$$\frac{d^2Q}{dt^2} = \left[\frac{\Delta k_0}{\tau} + \frac{AQ}{\tau_0 C\tau}\right]\frac{dQ}{dt}. \tag{10}$$

Integrating we find

$$\left[\frac{dQ}{dt}\right]_{t=0}^{t=\infty} = \int_{t=0}^{t=\infty}\left(\frac{\Delta k_0}{\tau} + \frac{AQ}{\tau_0 C\tau}\right)dQ,$$

$$0 = \frac{\Delta k_0}{\tau}Q + \frac{AQ^2}{2\tau_0 C\tau},$$

$$Q = -2\frac{T_0 C}{A}(\Delta k_0). \tag{11}$$

The spike width is $3.52\,\tau/(\Delta k_0)$ (and is *not* the same as the period of the very crude approximation). These equations have been experimentally confirmed by assembling subcritical masses.

Disassembly has not, so far, been included. Disassembly will be delayed more than the temperature increase, as we can see from the following argument. Vapor pressure for disassembly will increase with temperature (monotonically but not linearly). By Newton's second law, pressure is proportional to the second derivative with time of the displacement. The inertia of the system (which cannot be varied) is large enough that without a negative Doppler coefficient, the reactor power would have a large increase before disassembly terminates the excursion. But the excursion can be turned around by a large enough negative Doppler coefficient. Disassembly may take place early for small Doppler coefficients, or later for larger coefficients.

If reactivity increase continues and becomes a ramp instead of a step, the single spike of the preceding calculation becomes an oscillation once more. This is shown by calculations using the PAD code (Stratton and McLaughlin, 1976). This code, developed at Los Alamos for calculations of bomb explosions and critical assemblies generally, is a code in spherical geometry.

The size of the reactor core taken in this particular problem is that of the proposed Clinch River Breeder Reactor. The initial conditions are: a fully voided core, initial temperature 1500°K, 50 times nominal power, initial reactivity \$0.95, and a ramp increase of reactivity of \$44/sec.

Figure 11 shows power oscillations as a function of time for eight different Doppler coefficients. The figure shows clearly the reduction in the first peak as the Doppler coefficient is increased. However, the reactor disassembly does not take place so soon, for the larger Doppler coefficients and more oscillations appear.

Figure 12 shows that these oscillations with time reflect themselves in an oscillatory behavior of the energy release as a function of the Doppler coefficient.

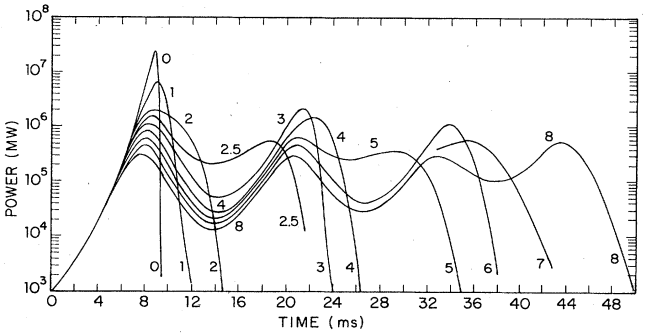

FIG. 11. Reactor power as a function of time for a \$44/sec transient according to the PAD code. The Doppler constant $Tdk/dT = -0.00X$, where X is the number attached to the curve.

It might seem attractive to choose a value of Doppler coefficient for which the energy release is minimized, but this cannot be done because the initial conditions are not well specified. Therefore, for an upper limit calculation, it is necessary to take the upper bound of a graph such as that shown in Fig. 12.

I reiterate the importance of the initial conditions for the start of the disassembly calculation. The thermal energy increase is roughly proportional to the square root of the ramp rate of reactivity increase at prompt critical. This determines the reactivity at the start of disassembly. The power level and temperature are also important. If the power level and temperature are low

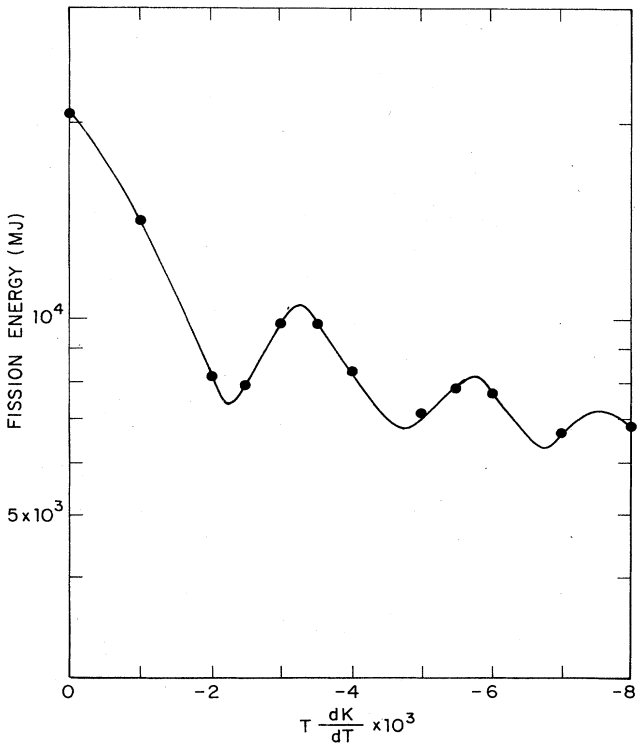

FIG. 12. Thermal, or fission, energy as a function of the Doppler constant according to calculations from the PAD code described in the text.

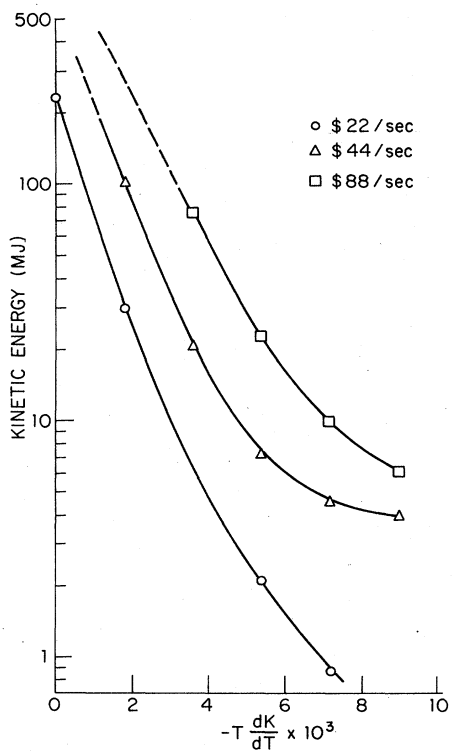

FIG. 13. The kinetic energy release for a similar problem to Figs. 10 and 11 but with a higher initial temperature (2945 °K). Note the absence of oscillations.

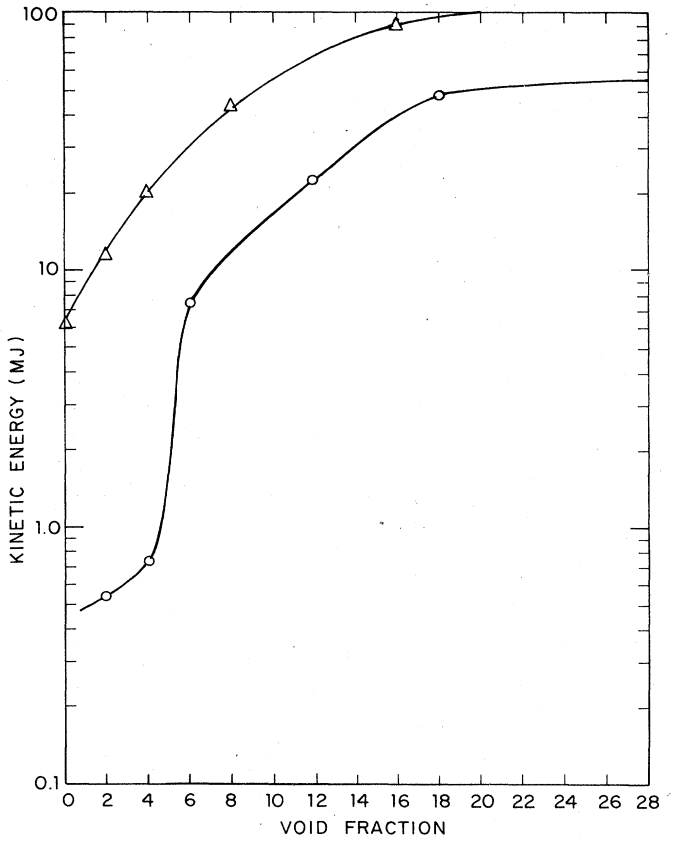

FIG. 14. The kinetic energy release as a function of void fraction in the core. This shows the effect of a void which delays reactivity reduction due to disassembly. Triangles are for the low initial temperature, and circles for a high initial temperature.

at prompt critical, the Doppler coefficient is slower to take effect; this shows itself mathematically in oscillations of large amplitude of the type shown in Figs. 11 and 12.

If the initial condition includes a higher initial temperature, this already produces a negative Doppler reactivity closer to the value about which oscillations would take place, and the amplitude is much reduced. This is shown in Fig. 13, where the *kinetic energy* is plotted against the Doppler coefficient for an initial temperature of 2945°K. The calculated kinetic energy is less than the fission (or thermal) energy shown in Figs. 11 and 12, but would show oscillations with the same period if the initial conditions allowed them.

The uncertainty in initial conditions for the disassembly calculation is referred to in jargon as a dependence on the "switch point" between two computer calculations —the predisassembly phase calculation using the code SAS3A and the disassembly calculation using VENUS. A recent paper (Bleiweis *et al.*, 1974) discussed the uncertainty in these terms, and the problem is implicit in another paper (Kastenberg and Catton, 1975).

Another uncertainty is whether or not the coolant spaces are voided. Disassembly cannot start until the hot fuel and vapor has filled up the coolant spaces, which happens without appreciable reactivity change. Only then can the pressure increase enough to push the fuel apart vertically and reduce reactivity. This is illustrated for the PAD calculations of Figs. 11 and 12, and in Fig. 14, which shows how the kinetic energy varies as the void fraction increases. If the core is voided before disassembly begins the energy is a maximum.

At what point can we begin to envisage that the integrity of the core is so destroyed that the calculation must proceed with a homogeneous disassembly model? Clearly the temperature must be above the melting point of the UO_2 (2767°C) over enough of the core to provide effective disassembly. But is this temperature 3000°C, 4000°C, or 6000°C? How are these initial conditions to be specified? This, together with the question of whether fission product gases are present to cause early disassembly, is the major question about disassembly analyses. This is part of a rule that, if we arbitrarily delay reactivity reduction due to disassembly in an accident calculation, we can calculate larger energy releases but the Doppler effect turns the transient around and therefore dominates the magnitude.

An initial condition for the disassembly calculation with low power level, yet high initial reactivity, is most dangerous (but may be nonphysical), since this delays the Doppler reactivity the most and leads to the largest oscillation about an equilibrium value. For a disassembly calculation the initial reactivity and the ramp rates are clearly connected, so that a search must continue for processes which can cause a high ramp rate of reactivity insertion.

Any disassembly must have some initial cause, e.g.,

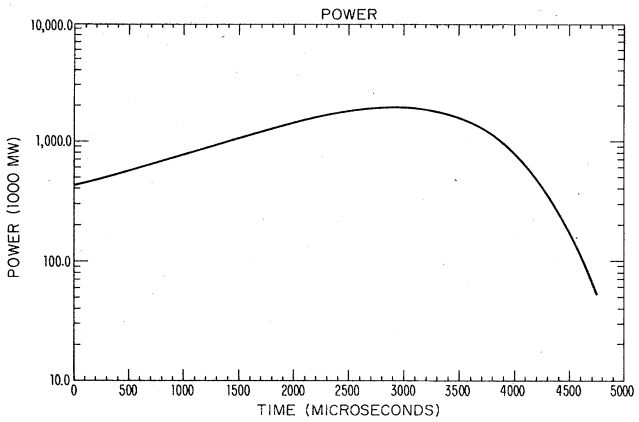

FIG. 15. A calculation from the VENUS code for the Clinch River Breeder Reactor, assuming transient undercooling without scram and limited fuel motion. This shows the power as a function of time. Case 5 from Bohl, 1975.

transient overpower or a loss of flow — accompanied by a failure to scram. This might lead, in a complex series of events, through sodium boiling, voiding of channels, and fuel compaction, to an increase of reactivity, and the reactor becomes prompt critical.

Moreover, the details of the disassembly are important in calculating the mechanical energy release.

In Fig. 15 I present a calculation from Bohl (1975) using the VENUS code (Jackson and Nicholson, 1972), which is for the Clinch River Breeder Reactor and has as a starting point the fuel melting and motion calculated by SAS. The VENUS code calculates in cylindrical geometry, and in this respect is more realistic than PAD.

Although none of the mechanistic calculations directly led to a hydrodynamic disassembly, in this case pessimistic fuel motion was assumed. In Fig. 16 the calculated reactivity transient is turned around by the Doppler coefficient in 1500 μsec and displacement as illustrated in Fig. 17 completes the shutdown *without* subsequent oscillations.

The displacement has only just begun as the reactivity returns to prompt critical at 1500 μsec, and therefore the whole transient is independent of the details of the displacement. The calculation then depends primarily on the Doppler coefficient, the neutronics, and the specific heat, which are well known; other parts of the equation of state are less important. Table IV shows the calculated temperature increase for this initiating event.

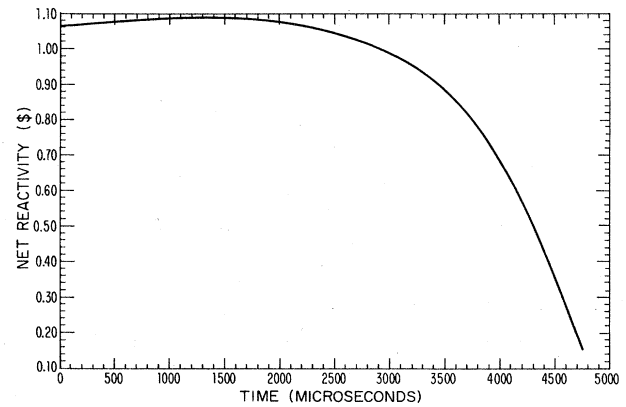

FIG. 16. As Fig. 14; reactivity as a function of time.

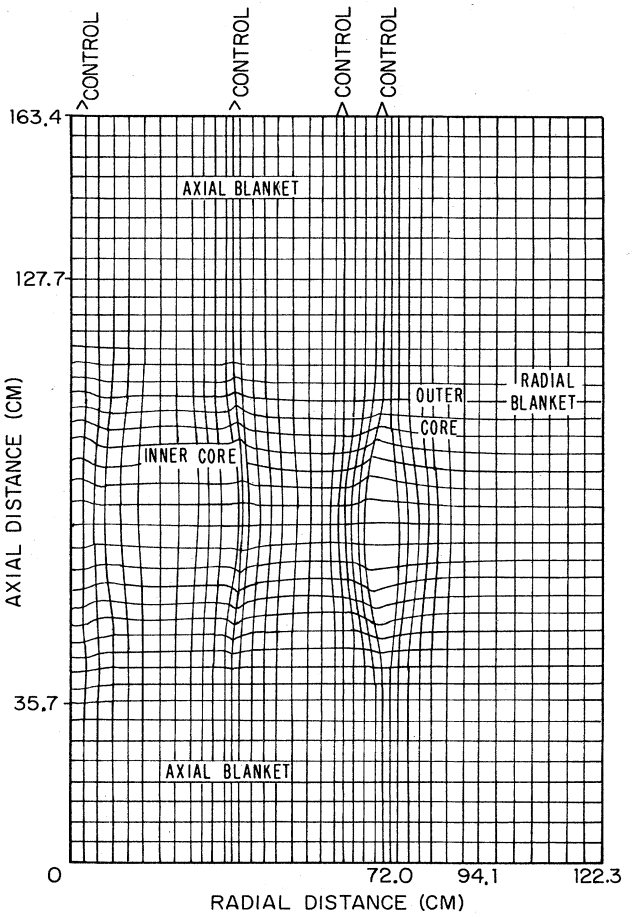

FIG. 17. As Fig. 14; changes of density shown at the end of the calculation.

There are (at least) two approaches: the first is to make a detailed mechanistic calculation of the chain of events leading to a prompt critical excursion; this will define the initial conditions. In his first paper, Bethe

TABLE IV. Results of calculation with limited initial fuel motion summary of disassembly calculation (from Case 5, Bohl, 1975).

Initial conditions	
Core average temperature	2990 °K
Reactivity	1.07 $
Power	4.28×10^{11} W ($440 \times$ nominal)
Ramp	50 $/sec
VENUS II results	
Core average temperature	4150 °K
Maximum temperature	5797 °K
Energy in molten fuel	5560 MJ
Fuel expansion work	
Work-energy to 2.1×10^7 cm^3 "Na-slug impact"	72 MJ
Work-energy to 1 atm	350 MJ

assumed that the disassembly would take place when the fuel had expanded to fill the void space formerly occupied by the sodium coolant. This part of the expansion will not appreciably alter the reactivity. After this point in time, the core can be treated as a single phase liquid. It is important to realize that no plausible mechanistic calculation (using modern LMFBR parameters) has been made which achieve a high enough reactivity or ramp rate to give a large energy release, although with a large positive sodium void coefficient it might be possible. However, it is obviously impossible to explore *all* possibilities in such a mechanistic calculation and there is the concern that a process may have been forgotten even though the treatment is very conservative within the scenario taken. Note that it is not known how to produce a large explosive fuel–coolant interaction, and therefore an explosion-driven recriticality is not a mechanistic model.

Therefore it is conventional to take a second approach and postulate a mechanistic series of events leading to an initial condition, accompanied by a pessimistic assumption of another event leading to a large ramp rate. The other events must not be completely implausible. I have outlined the possible large ramp rates earlier. This second approach is the "nonmechanistic" assumption by which mechanical energy releases may be bounded if not realistically calculated.

C. Calculation of mechanical energy release

The calculation of mechanical energy is less well defined than the total heat energy. What is the mechanism of conversion? The analytic calculations use an *isentropic* expansion of the vapor — fuel vapor, stainless steel vapor, fission product gas pressure, or sodium vapor. The calculated efficiency of conversion is low when the vapor pressure is low, and rises as the temperature (and hence the vapor pressure) rises.

Energy release is sensitive to the pressure and displacements, and the assumptions can make a difference of up to a factor of 2. It is confirmed by parameter studies with the codes that the energy increases, provided they are large, are only dependent on particular parts of the equation of state for the mixed oxide PuO_2/UO_2 (Booth *et al.*, 1974; Fischer *et al.*, 1976a, 1976b). These parts are the specific heat (for this directly determines the Doppler feedback) and the relation between temperature and pressure.

As the temperature of the UO_2 or PuO_2 is raised, the pressure P is at first given by the vapor pressure of the boiling uranium, but after a threshold energy density, Q^*, giving a single mixed phase, it becomes proportional to the energy density, $P=(\gamma-1)(E-Q^*)$, where γ is the ratio of the specific heats at constant pressure and constant volume. For all but the most severe postulated accident the reactor is disassembled by the lower (vapor) pressure. Moreover, disassembly itself plays only a small part in the initial turnaround of the excursion and the energy release. Therefore, the vapor pressure curve is most important.

Associated with the importance of the vapor pressure of the boiling uranium, is the importance of the vapor pressure of the sodium, the clad and of the fission products (particularly Kr and Xe). At low temperatures these pressures will far exceed the vapor pressure of the UO_2/PuO_2 and will dominate the disassembly process. To be pessimistic we might assume that they will have disappeared from the core at the time of disassembly. A problem of consistency arises here which does not seem to have been resolved. If we assume that the sodium coolant, the fission products, and the stainless steel have all disappeared from the fuel mix at the start of the disassembly, should we also assume that they have disappeared for later purposes of calculating the conversion of thermal to mechanical energy? I will return to this later.

It is important to remember that this mechanical energy itself is not the number of interest for any given reactor, rather it is instantaneous forces and pressures which could raise the reactor vessel head and/or violate the containment. The energy available to raise the vessel head is about one-tenth to one-fifth of the energy of expansion to one atmosphere. A direct conversion of the calculated mechanical energy to TNT equivalent is overly pessimistic because all existing reactor accident models give smaller instantaneous forces for the same total mechanical energy than for an equivalent energy release by a TNT explosion (Gilby, 1970). This is especially true of the modern LMFBR's with a negative Doppler coefficient, where the energy is released over a time of the order of 5 msec, whereas TNT explosions occur in hundreds of microseconds. On the other hand, a similar calculation for Fermi I gave energy releases over times of 100 μsec both because of a small Doppler coefficient and because of less accurate, more pessimistic calculations.

D. Experimental tests of the HCDA computer codes

The PAD code has been tested against a number of explosive critical assemblies. These include (Stratton, 1972):

GODIVA — a spherical solid metal system
SNAPTRAN 1, 2, 3
KIWI–TNT
KEWB

For example, the following comparison (Table V) was achieved for the KIWI–TNT experiment. These are releases of the order of some that have been calculated for extreme, but unlikely, reactor accidents. It must be noted that the mechanical energy in this case derives entirely from fuel vapor and therefore this is not a test of the fuel-coolant interaction. The analytic theory shows that a one-dimensional (spherical) model for a core of the same mass should be an excellent approximation even to a flat pancake (Meyer and Wolfe, 1968; Ishikawa *et al.*, 1970). A comparison of VENUS and

TABLE V. Fission and explosive energy for the KIWI–TNT experiment.

	Fission energy release	Explosive energy
Experiment	$10^4 \pm 10^3$ MW sec	345 ± 70 MW sec
Theory	1.85×10^4 MW sec	205 MW sec

PAD has been made (McLaughlin *et al.*, 1976). A direct comparison of VENUS, which is a two-dimensional code and should be more reliable, and experiment SNAPTRAN 2 and 3 and KIWI-TNT has also been made (Bott and Jackson, 1976). This is important, since VENUS is a code used for calculations for real reactor designs. The French SUREX code (Mantamakas, 1974) is similar to VENUS, but there has been no detailed comparison. This is not a complete listing of computer codes. A more complete list is available in a 189 page review, ERDA 75.

If prototypically worthwhile experiments existed to completely bound all possible accident situations, calculations would only be needed to *interpolate* between experimental points, and for this, calculations can be well trusted. But experiments do not bound all possible situations. Therefore this is a place where calculations are used to explore new areas, and though the calculations seem well based physically, we need to be sure that nothing is forgotten and the calculation is indeed as conservative as its proponents claim.

E. Fuel/coolant interaction (FCI)

The nature of the interaction between the fuel and coolant will dominate the conversion of thermal to mechanical energy; the fuel, fission products, and stainless steel have a low vapor pressure and are not enough to be appreciable drivers of mechanical energy. However, if energy can be conveyed promptly from the uranium to the sodium, there could be a transfer of thermal to mechanical energy approaching the thermodynamic maximum of 30% —calculated by Hicks and Menzies (1965). This area is perhaps one where a great margin of safety might be most readily proved. On the one hand, in no experiment, in situations designed to approximate accident situations, has a high conversion efficiency been obtained, and efficiencies of less than 1% are usual. On the other hand, the experimental situations are, of necessity, idealized, and critics correctly claim that large efficiencies cannot yet rigorously be excluded. A theory has been proposed (Fauske, 1972) which excludes these large efficiencies, and it is desirable to understand it in detail.

The discussion of fuel–coolant interactions has tended to be concentrated on whether or not liquid/liquid explosions occur. Explosions are an efficient way of converting thermal energy rapidly to mechanical energy. Apparently similar situations might usually give a low efficiency, but occasionally a large explosion. This would render reliance on (limited) experiments nonconservative.

The experimental situation on fuel–coolant interactions consists of three separate sets of data:

1. There are many examples of vapor explosions as one liquid falls upon another. Many of these examples have been very destructive. These have included molten lava on water (Colgate and Sigurgeirsson, 1973), molten steel on water, molten aluminum on water (Long, 1957; Board *et al.*, 1973), freon 22 on water (Board, Hall, and Brown, 1974; Henry *et al.*, 1974), and liquefied natural gas (LNG, which is mostly CH_4) on water (Burgess *et al.*, 1972). These are summarized by Witte *et al.* (1970, 1973). In some of these a chemical, oxidizing, reaction could, in principle, take place which would liberate energy (and hydrogen gas). But in the LNG/water explosions, for example, the energy release has always been too small to set off a chemical explosion, or even break a pane of glass (Katz, 1975).

2. There have been tests on the direct interaction of uranium oxide and sodium. (It is assumed that mixed oxides will have similar characteristics to uranium oxide.) Some of these, at Argonne National Laboratory (Henry and Fauske, 1975a, 1975b) and at Foulness (Halliwell and Pottmeyer, 1975) have used thermite explosive to raise the uranium quickly to a temperature characteristic of postulated accident situations. These may suffer from the presence of impurities, which can act as an insulating layer to prevent liquid/liquid contact and inhibit explosions or other rapid heat transfer mechanisms. Irradiated reactor fuels will have fission products which are likely to produce an insulating layer also, but to provide a more pessimistic test, other experiments use uranium oxide heated in a crucible and poured or heated by direct Joule heating. These experiments are in progress at ISPRA (Euratom) and Grenoble (CENG, France) (Amblard *et al.*, 1974). In this case there should be no insulating layer except the one caused by fuel vapor.

3. In the TREAT test facility single fuel pins are irradiated with large power pulses; the molten fuel interacts with sodium, and the mechanical energy produced is measured by a piston (Wright *et al.*, 1974; Zin *et al.*, 1975). The efficiency of conversion of thermal to mechanical energy is about 10^{-3}. This geometry is appropriate to a study of the interaction after the failure of a fuel pin, but if a hydrodynamic disassembly takes place, the fuel may be more intimately mixed with the sodium—and more energetic interactions may be possible. For this reason, the experiments of class 2 are important.

In order to convey heat rapidly from uranium fuel to coolant it is necessary to fragment the fuel to obtain a large surface area. We see this for an Al–H_2O explosive interaction in a calculation by Kastenberg and Catton (1975). Explosive interactions have taken place in times of a millisecond or less and therefore we take $t = 1$ msec as a characteristic time. If we have a molten sphere of aluminum, radius 2.5 cm at 200°C above the melting temperature (860.4°), the thermal energy available before solidification is given by $Q = mc\Delta T = 39,000$ J. This can work against outside pressure p for a time t over an area A equal to 79 cm^2, giving an impulse pt. This is relieved by a wave with velocity v which is of the order of the velocity of sound in water— 150 000 cm/sec. Then the impulse $pt = Q/Av = 32\ 900$ dyne-secs/cm^2. The local pressure can be huge, since t can, over a centimeter, be as little as 10 μsec. But at a distance only one millisecond has been observed, and this leads to pressure pulses of 3.29×10^7 dyne/cm^2 or 32 atm. Pressures of this size can cause damage to internal parts of the reactor. But can we release 4×10^4 J to the liquid in one msec? This requires a heat transfer of 5.1×10^5 J/cm^{2}sec, which is an order of magnitude larger than the value obtained by stable film boiling— which is the highest measured. Therefore, to transfer energy fast enough

to enable an explosion to take place — or even an efficient conversion to mechanical energy — it is necessary to fragment the fuel into small pieces. An initial small explosion might achieve this. The explosive nature of the interaction is probably due to superheating of the surface of the cold liquid, releasing energy suddenly. Such arguments as this have been made by Katz (1972) and Colgate and Sigurgeirsson (1973).

In the uranium oxide–sodium interface, similar conditions apply; but sodium has a high thermal conductivity and can quickly remove heat from the point of contact without vaporization that could give a mechanical force. So another condition for an explosive interaction may be that the amount of sodium must not be so great that the heat is carried away. Finally, fission products and fuel vapor (or other impurities) might prevent liquid/liquid contact and inhibit the rapid heat transfer necessary.

For some liquid/liquid interactions, a physical (superheat) explosion might trigger a chemical explosion of greater magnitude, although this does not seem to have been observed. Such chemical reactions might be produced by steel on water, LNG on water, or in a light-water reactor, zirconium on water. With uranium on sodium such a chemical interaction is not possible.

Experiments with UO_2 and sodium have been summarized by Fauske (1972, 1973) with additional comments by this author in italics:

1. True vapor explosions (*have been identified by high pressure pulses and*) have only been observed where liquid sodium (*the colder liquid*) has been ejected into a pool of molten UO_2. (Armstrong *et al.*, 1971).
2. When molten UO_2 and stainless steel have been injected into a pool of liquid sodium, no explosions have been observed, and low thermal to mechanical energy conversion ratios have been achieved (*of the order of* 10^{-3} *in TREAT, but closer to* 1% *for experiments with Na injected into uranium*).
3. Extensive fragmentation of the hot materials occur where they are injected into liquid sodium. The occurrence of extensive fragmentation does not lead to vapor explosions, but rather just to violent noisy boiling.

These explosions have been interpreted as superheat explosions (Katz and Sliepcevitch, 1971, 1972; Nakanishi and Reid, 1971). This limits the energy to that stored in a thin surface layer. The energy can then only be large if the surface area is made large by fragmentation.

Fauske (1972) and (Henry and Fauske, 1975a) have proposed that the rapid vapor generation needed for an explosion can only be provided by *spontaneous nucleation* at the liquid/liquid interface. Spontaneous nucleation is the process whereby, in a pure liquid, bubbles form against the pressure from the surface tension, *without* external nucleation sites. Liquid sodium has a high surface tension and therefore a high spontaneous nucleation temperature. At first sight this seems contrary to the statement that the observed superheat in LMFBR conditions is close to 10°C. This low superheat occurs because nucleation occurs at impurity sites. Fauske and Henry argue that there are not enough impurity sites to allow rapid enough vaporization to cause an explosion.

After an initial drop of liquid has exploded, Fauske envisages a fragmentation of the whole surface with rapid superheat and explosion.

The energy to form a spherical vapor bubble of radius r with surface tension σ is given by $W = 4\pi r^2\sigma + 4/3\pi r^3 + 4/3\pi r^3[p_1 - p_v]$, (where p_1 and p_v are pressures in the liquid and vapor, respectively), and at the limit of stability $p_1 - p_v = -2\sigma/r$ so that, neglecting p_1,

$$W = 16\pi\sigma^3/3p_v^2 . \tag{12}$$

The rate of nucleation becomes $J = A(T)\exp(-W/kT_s)$. The exponential *dominates* this equation, so that rapid nucleation occurs when

$$16\pi\sigma^3/3p_v^2 < kT , \tag{13}$$

when T_s the temperature of spontaneous nucleation is approximately

$$T_s = 16\pi\sigma^3/3p_v^2 k . \tag{14}$$

This temperature is sometimes called the superheat limit temperature (T_{s1}). It is the highest temperature to which a liquid may be superheated, and it is just below the critical temperature.

Fauske's first argument is that the temperature of the hot liquid must be above this spontaneous nucleation temperature or an explosion cannot occur. This condition is satisfied for all well documented cases of liquid/liquid explosions.

The next point is that the temperature of the liquid/liquid interface is less than the temperature of the hot liquid. The two liquids must be in intimate contact for rapid nucleation to occur. The interface temperature for two plane liquid surfaces in *intimate contact* is given by (Carslaw and Jaeger, 1959):

$$T_i = T_H + T_c\psi/(1+\psi) ,$$

$$\psi^2 = k_c\rho_c C_c/k_H\rho_H C_H , \tag{15}$$

where T, k, ρ, and C are the temperature, thermal conductivity, density, and specific heat, respectively, of the hot and cold liquids (subscripts H and c).

Then Fauske proposes that a criterion for the possibility of liquid/liquid explosions is that $T_i > T_s (= T_{s1})$ and for $T_i < T_s$ explosions are impossible.

For liquid hydrocarbons on water $T_H - T_i \approx 5° - 10°$, and although experiments (Porteus, 1975; Porteus and Reid, 1976; Reid, 1976) suggest that these inequalities are correct, the accuracy is not great.

Henry *et al.* (1974, 1975b) and Board (Board and Hall, 1974) have obtained liquid/liquid explosions with water poured on freon (−40°C) if the water temperature was above 76°C ($T_i > 54$°C; T_s for freon is 54°C. Henry *et al.* (1975b) found similar effects for mineral oil poured on propane.

For liquid uranium on liquid sodium, the high thermal conductivity of the liquid sodium can keep the interface temperature well below the homogeneous nucleation temperature, so that

$T_H = 3000$°C (uranium temperature)

$T_c = 400$°C (molten sodium)

then

$$T_i = 1160°C$$

but

$$T_s = 2030°C \quad (16)$$

Thus, according to the first requirement ($T_H > T_s$), an explosion can occur, but according to the second ($T_s > T_i$) it cannot immediately occur. However, if the cold liquid (sodium) is fragmented and entrained in the hot one, the temperature T_i can rise, and when it reaches T_s a vapor explosion can occur as observed (Armstrong *et al.*, 1971).

Board *et al.* (Board, 1974; Board, Hall, and Brown, 1974) do not accept the role of spontaneous nucleation as the sole source for rapid generation of vapor. In freon–water explosions, they found that the pressure pulses were a smooth function of T_i with no sharp discontinuity at T_{sn}. They conclude that Henry and Fauske's relation determines "only the initial conditions for the explosion and is not relevant to the process of explosion development" and that "it is not possible to rule out large-scale fragmentation explosions if there are any other circumstances which can lead to the relevant initial conditions."

From their measurements of the pressure pulses from freon on water, Board, Hall, and Brown (1974) estimate an efficiency of 30% for conversion of thermal to mechanical energy by vapor explosions. Their predictions for UO_2/Na explosions show propagation velocities of 10^5 cm/sec with pressures of 15 kbar. Therefore, in addition to being a mechanism for achieving a high conversion efficiency, explosions could cause energy to cross the core in 1 msec to cause coherent sodium voiding. Although the pressure of 15 cannot act coherently, and over a long enough time, to achieve a major increase in the ramp rate from recompaction, factors of 2 – 3 have been suggested.

Board and Hall (1974) and Catton (1975) have found and studied explosions of molten tin on water with the cine and flash photography. The spontaneous nucleation temperature for this case is far lower than for sodium and the interface temperature is higher, and therefore such conditions are not directly relevant to the LMFBR.

Two further objections have been raised to the Fauske theory. Firstly, although spontaneous time of liquid/liquid contact in the LMFBR core disassembly accident may be a hundredth of a second or more, even if ordinary nucleation sites are not numerous enough to allow vaporization over a millisecond and an explosion, energetic boiling (with a high thermal to mechanical efficiency) may occur (but no mechanism is suggested).

Secondly, the spontaneous nucleation does not explain why at much higher temperatures $T_H \gg T_s$ explosions are suppressed. This is not of *direct* concern to LMFBR safety, but if a theory incorrectly predicts some related facts, it is thrown into general disrepute. This suppression is discussed by Henry and Fauske (1975a) and Reid (1976) and seems to be due to an insulating vapor blanket forming at the interface; Henry shows how this vapor blanket can develop at higher temperatures. Porteus (1975) shows that, for liquid ethane or methane *poured* onto water, explosions do not occur but the vapor blanket may be reduced by bringing the two liquids into contact more forcefully, with a pressure of 1 atmosphere, and an explosion is then inevitable. Moreover, the experiments are affected by a layer of mineral oil on the surface, since explosions are more readily observed with saturated hydrocarbons on mineral oil.

The experiments have been done with easily available liquids and none have simulated the high thermal conductivity and high surface tension of sodium which lead to the low T_i and high T_s.

Even if the Fauske–Henry theory is not exact, high efficiency of conversion to mechanical energy may not occur. The contact temperature in the accident situations envisaged above is *well below* the homogeneous nucleation temperature and far from the regions where higher efficiencies have been observed.

Another possible argument might be that boiling will occur on fission product gases and other nucleation centers, and that superheat will not therefore occur even instantaneously in a reactor environment. If this is so, there will be no superheat explosions regardless of the truth of Fauske's theory. But could there be enough superheat to store energy, but enough nucleation sites to allow an explosion with $T_i < T_s$? This is unlikely but it seems worthwhile to understand all of these problems.

The present experimental tests of fuel–coolant interactions all use only a small amount of material. The Grenoble group note (Puig and Szeless, 1972) "the mechanical work done by the expansion of the vapor might be considerably reduced if condensation in a cold coolant takes place. Because of this effect, which is dependent on the size of the interaction zone, the extrapolation to a large volume of a low efficiency obtained in a small experiment is doubtful." On the other hand, explosion propagation may be more likely in a large system.

If the efficiency of conversion to mechanical work is always of the order of a percent or less, there is a very small mechanical energy release. If it is close to the maximum thermodynamic efficiency, the energy release can be large enough to be too difficult to contain.

Understanding the fuel–coolant interaction seems, therefore, to be one of the most important questions in fast reactor safety at the present time. Imaginative experimental and theoretical work is needed to bound the problems and possibly to suggest reactor geometry modifications that might prove desirable. Experiments are now directed towards understanding the fragmentation explosions that have been shown to occur and discovering the parameters that govern them. Then experiments can be designed to test whether the efficiency of heat transfer to mechanical energy is low even under conditions more conducive to an explosion or large energy release than an LMFBR environment.

This work,is, of course, underway in several places: Berkeley, United Kingdom (Board and Hall, Foulness), UCLA (Catton), ANL (Armstrong, Ziri, and others), ISPRA, Italy (Kottowski), Grenoble, France (Semeria). There are also regular meetings on fuel–coolant interactions held by Organization for Economic Cooperation and Development (OECD), which should be continued until there is further understanding and agreement.

This work may show that even if explosions occur, they will not involve all the uranium and fuel, and the average efficiency may be appreciably less than the 30% of Hicks and Menzies. It must be hoped that this "proof" will be

forthcoming before many LMFBR's are built. A warning is in order: present experiments and present optimism is based on studies with oxide fuels. It is likely that the possibility of a fuel–coolant interaction is greater with advanced carbide fuels, due to the higher heat conductivity.

In reactor safety calculations in France (PHENIX) (Petit, 1975) and the U. S. (for FFTF) (Bohl *et al*, 1975) a mechanistic model is made of the fuel-coolant interaction leading to the expulsion of a sodium slug. This in turn is used, not merely to give a number for the mechanical energy release, but also to examine the way in which the energy distributions vary with time. Loadings on the reactor head and vessel walls are thus calculated.

These deliberately pessimistic models lead to efficiencies of 5–10%, which is less than the theoretical maximum of 30%, but higher than the 0.1%–1% reached in experimental situations so far. More realistic models are being developed by Amblard *et al.* (1974), by Goldhammer and Kottowski (1975), and by Cho *et al.* (1971, 1972).

Until the absence of efficient fuel–coolant interactions is demonstrated by large-scale experiments, these pessimistic models are used by safety analysts. For this report, I assume the model of a sodium slug is correct and therefore take the results from Bohl.

The importance of this efficiency can be seen from Fig. 18 shown by Fauske (1976). The lower line relates the mechanical (work) energy "at slug impact," which is the energy which is available to lift the reactor vessel head, to the average core temperature in the disassembly. The upper line is the thermodynamic maximum (Hicks and Menzies, 1965). The difference between the thermodynamic maximum and the slug impact calculations is a factor of 10. The points above the curve are slightly more pessimistic; the preliminary safety analysis report of the Clinch River Breeder Reactor (PMC, 1975) suggests a maximum of 660 MJ expansion to 1 atm, and a slug impact of 100 MJ; NRC (1976) have suggested taking a maximum figure approximately twice as big.

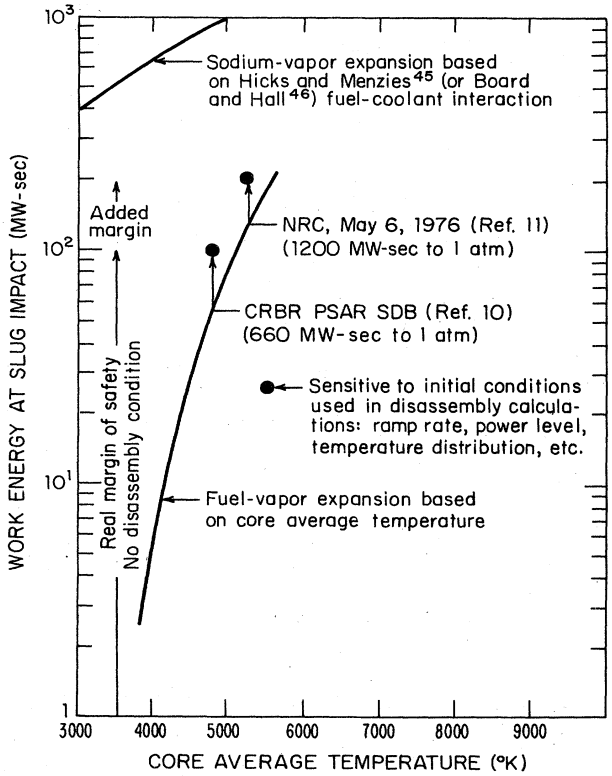

FIG. 18. Illustration of work potential (at slug impact) from two different source terms including fuel vapor and sodium vapor expansions (from Fauske, 1976).

To conclude this section I note that if a hydrodynamic disassembly is postulated to occur by fuel vapor alone, without fission product gases or clad vapor, the temperature must be high, and for a 1000 MW reactor, thermal energies up to 30 MJ can be calculated. Although there was by postulate no sodium present at the start of disassembly, we might postulate that sodium reappears in some unspecified way and then a fuel-coolant interaction occurs. At this time a high kinetic energy can be calculated. But after the first half hour of reactor operation enough fission product gases will have been generated to make it unlikely that fuel vapor alone will cause the disassembly.

V. CONSEQUENCES OF CORE DISASSEMBLY

A. Containment integrity

If a reactor meltdown does not lead to a large energy release, the containment will not immediately be violated. Even if the fuel melts through the reactor vessel at a later time, the fuel vapor will have condensed. In the introduction I noted the distinction between the maximum light-water reactor accident—fuel meltdown accompanied by containment failure—and that for a liquid metal fast breeder reactor—fuel vaporization and release of fuel vapor from a fractured containment. The consequences of the latter may be more severe, and LMFBR acceptability may depend on the size of the maximum possible accident. "The probabilities of various scenarios should be factored into the discussion to ensure the LMFBR risks are comparable to those of a LWR." (NRC, 1976).

It should not be assumed that a release of mechanical energy that exceeds the design maximum would always violate containment in this way. The vessel head bolts would lift, but the rest of the system is designed not to yield.

However, there is little reliable discussion on the margin available here, and what energy would be necessary to give, for example, a 50% chance of immediately violating containment. This seems to be a place where safety margins in addition to those usually considered might well be sought.

B. Post accident heat removal (PAHR)

It is important to ensure that, after an accident, decay heat can be removed. In the light-water reactor safety study (Rasmussen, 1975) a major conclusion was that the failure of post accident heat removal is an important contributor to the overall accident risk. The discussion below is cursory and not intended to be complete. A more complete review is available in Kazimi and Chess (1974).

In a light-water reactor, the accident that has been of principal concern has been the loss-of-coolant accident. The cooling fluid, water, disappears and if the protective devices do not work, the reactor will be uncooled. Pipe breaks can occur, and protective devices (ECCS) can fail. However, in a breeder the liquid sodium coolant has a high heat conductivity, heat transfer, and boiling point, and is, therefore, only dispersed as and when the uranium is dispersed. Simple calculations show that if the uranium and sodium remain in contact, there is enough sodium to take away the heat from fission product decay by boiling for many hours, even if other forms of heat removal cease. These suggest that adequately reliable heat removal systems can be devised. The POOL-type reactors are superior to the LOOP-type here; there is more sodium to absorb the heat. But even in a typical LOOP-type it takes 25 hrs for decay heat to raise the sodium to the boiling point.

The design must ensure that when the fuel is dispersed at the end of an accident it stays with the sodium and there is no further recriticality. Details of these points are outside the scope of this report, but indicate that an acceptable shutdown condition can be achieved more readily in a breeder reactor than after a loss of coolant in a light-water reactor. But "rigorous mechanistic evaluations of the fuel distribution patterns following an HCDA are lacking. The sequence of an HCDA accident is dependent on a large number of physical phenomena that are not completely understood. The PAHR analyses should therefore be prepared to cover a range of possible fuel distribution patterns" (Kazimi and Chess, 1974). This is not likely to change even after a lot of study, so that we must depend on "conservatively bounding analyses."

C. Radioactivity release

For the FFTF the HCDA calculations showed that a core meltdown would stay within the reactor vessel. But it was postulated that the molten core might melt through the reactor in spite of the calculations. As noted above there are two physically distinct times at which radioactivity can be released. Even if the containment building holds during the accident, sodium may chemically interact with the concrete and the containment may have to be bled to avoid buildup of hydrogen. Under this circumstance a fraction of sodium iodide particulates and plutonium particulates would be vented. In discussions of this problem for the FFTF within the NRG it became clear that release of radioactivity by containment melting might be a problem for meeting the regulation (10 CRF 100) of radiation at the site boundary for a design basis accident. This would not be true if core melting is less probable (Class 9 accident). It was suggested that venting through an appropriate filter, large but conceptually simple, avoids this problem (Stratton, 1975).

There is a further possibility that the containment might break instantly. Although early containment violation for light-water reactors is called *incredible* (a Class 9 accident) for licensing purposes, the light-water reactor safety report (Rasmussen, 1975) estimated small but nonzero probabilities of its occurrence. The effect of the containment violations could on rare occasions be large, but according to the report, not as serious as some other equally hypothetical catastrophes to society.

The effects of instant containment violation in the LMFBR might be greater than in an LWR. In the light-water reactor, only the volatile fission products, I, Kr, Xe, and to some extent Cs, would be released. But plutonium and uranium vapor cause the accident I am discussing and can be released with the others. However, with a large negative Doppler coefficient and with fission product gases causing the disassembly, the vapor pressure of uranium or plutonium is small, and therefore the quantity is small, and the vapor would soon condense.

Early accident studies had suggested the liberation of a large quantity of plutonium vapor—not present to such a large extent in a light-water reactor accident. The light-water reactor safety study (Rasmussen, 1975) assumes *some* release of plutonium from a reactor accident, but other radionuclides dominate the accident consequences. The plutonium released would have to be about 100 times greater before the consequences are doubled (see Table VI, 13-1 of Rasmussen, 1975).

VI. SUMMARY AND CONCLUSIONS

A. Reactor safety issues

The Division of Reactor Safety of ERDA has summarized its view of the open reactor safety issues. These are generally grouped according to the Lines of Assurance.

LOA1: Prevent accidents
- Slow pin-to-pin failure propagation
- Reliability of critical components
- Coolant blockage in large bundles
- Demonstration of inherent safety features

LOA2: Limit core damage
- Fuel failure mechanisms
- Fuel dispersal mechanisms
- Short period fuel failure location
- Short period fuel–coolant interaction (FCI) in large intact geometry
- Subassembly to subassembly propagation
- Subassembly coolability

LOA3: Maintain primary system integrity
- Recriticality energetics
- Post accident heat removal (PAHR)
- Short period fuel–coolant interaction (FCI)

LOA4: Attenuate radiological products
- Pu and fission product release
- Attentuation mechanisms
- Engineered safeguards

This list includes some items (in LOA1) which contribute directly to *reliability*, the principal method of assuring real safety.

However, in one way or another, all the items on the list have been mentioned, if not discussed.

Also prepared for ERDA by Argonne National Laboratory was a chart of the present status of the various fast reactor safety issues and their possible resolution. I present this, with which I am in general agreement, as Table VI.

TABLE VI. Key issue and required activities matrix.

Key safety issues and problems		A. Brief description	B. Impact on design/ licensing	C. Current status	D. Major remaining uncertainties	E. Summary/probable resolution of issue
LOA 2 ↑	Pin-to-pin propagation (1)	What are the limits to which we can operate long-term with failed fuel in the reactor? Can local faults (blockages, etc.) propagate, and if so, what are the limits of detectability?	Is individual subassembly instrumentation required? Are whole core sensors required? If pin-to-pin propagation does not occur, ductless core becomes a viable option from a safety point of view.	Operation with limited failed fuel appears feasible. Rapid pin-to-pin propagation appears unlikely. Slow blockage propagation cannot be ruled out at this time.	Long-term operation with failed fuel needs to be demonstrated. Slow blockage propagation cannot yet be ruled out. Establish means for detection if required.	Long-term operation with failed fuel likely to be demonstrated. If blockage propagation should occur at all, it can be detected by appropriate instrumentation.
	Subassembly-to-subassembly propagation (2)	Can subassembly meltdown propagate to its neighbors? With scram but with pump trip? At nominal power without scram? Does propagation lead to major whole core involvement?	Establish what protection systems are required if propagation should occur. Enabling the consequences of the accident to be determined for risk assessment.	Single subassembly meltdown is probably of lower probability if proper detection is used than assumed previously. Subassembly meltdown with early scram is unlikely to lead to major core involvement. Subassembly meltdown without scram may lead to major core involvement.	Establish source terms for thermal and mechanical loads on adjacent hexcans. Establish the extent of propagation if it should occur.	Subassembly-to-subassembly propagation with scram may be demonstrated to be unlikely. Subassembly-to-subassembly propagation without scram may lead to whole core involvement.
↓	Extent of core damage (whole core initiators) (3)	To what degree do we proceed to whole core involvement with the commonly postulated low probability whole core initiators (e.g., LOF and TOP without scram, LOPI with scram?)	Significant core design changes may be required to limit core damage for some initiators. LOPI with scram accident may require design features to limit the depressurization rate.	LOF with failure to scram tends to bring us to whole core involvement (true for FFTF & more so for larger systems). TOP with failure to scram may only result in limited core damage. LOPI with scram may lead to core disruption depending on design.	Time and location of cladding failure in TOP. Fuel ejection, FCI, coolant voiding, fuel sweepout, plugging, & coolability in TOP accidents. Establish margins for coolability under LOPI conditions. Establish design features required to limit core damage for specific initiators.	Current status of LOF with failure to scram unlikely to change without major design concept changes. TOP with failure to scram may only result in limited core damage, but this may not be a general result. LOPI likely to be resolved by design.

TABLE VI. (*Continued*)

Key safety issues and problems	A. Brief description	B. Impact on design/ licensing	C. Current status	D. Major remaining uncertainties	E. Summary/probable resolution of issue
Accident energetics -Na voiding (4)	Can an overpower transient induced or enhanced by Na voiding with positive reactivity effect lead to prompt criticality with significant ramp rate? Are there any autocatalytic effects? What are the relative importance to various accident paths?	Implication on core design if positive sodium void coefficient needs to be designed out. May determine structural design basis for the plant.	Doppler effect limiting initiating accident energetics demonstrated. Na superheat demonstrated to be essentially zero for LMFBR conditions. Negative or slightly positive Na void worth leads to benign initiating accident energetics. For cores with large positive Na void an energetic initiating phase cannot be precluded.	Incoherence effects in voiding and clad motion both core wide and within subassemblies. FCI with sodium in core leading to autocatalytic effects? Fuel failure mechanisms and early fuel dispersal. Inherent mitigating effects (fission products) in the disassembly process. Probabilistic studies and target plant studies required for program guidance.	For commercial plants with large sodium void worths the voiding induced energetics may approach or exceed containment capabilities. The uncertainties in bounding this condition may remain high. Design options which reduce the sodium void worth may provide greater margin for limited mechanical damage to primary system.
Accident energetics -recriticality (5)	Can high ramp rate recriticalities occur due to: Continuation of noncoherent initiating phenomena? Re-entry of initially dispersed fuel? Collapse of boiling fuel pool? Autocatalytic disassembly?	If energetic recriticality can occur, they may be difficult to contain with currently used containment concepts.	The dispersive nature of disrupted core materials is likely to preclude the occurence of recriticality. Analyses of oxide fuel done with simplified models indicate dispersal will occur down to levels corresponding to decay heat.	Details of transition from intact to disrupted geometries. Flow regimes associated with dispersal and heat transfer in fuel steel systems. Details of fuel-steel blockage formation and pressurization by steel vapor. Details of possible gravity and/or pressure driven recompaction.	Energetic recriticalities due to fuel compaction will be shown to be very unlikely.
Accident energetics -FCI -disassembly (6)	Can an energetic thermal interaction occur between fuel and coolant? With Na in? With Na out? Given the initiating conditions for a disassembly, what is the core energy release?	If energetic excursions and/or vapor explosions occur they will determine the structural design basis for plant.	Fuel vapor expansion used as a source term energetic FCIs not considered. VENUS-II disassembly code with improved equation of state available.	Potential for energetic thermal interaction with Na with prompt burst condition. Effects of gaseous fission products on disassembly energetics. Rate and structural effects and the need for multicomponent material treatment in the disassembly.	Energetic fuel-coolant interactions will be ruled out. For high ramp rates gaseous fission products unlikely to affect the energetics.

LOA 3

TABLE VI. (*Continued*)

Key safety issues and problems		A. Brief description	B. Impact on design/ licensing	C. Current status	D. Major remaining uncertainties	E. Summary/probable resolution of issue
LOA 3	System structural response (7)	What is the structural response of the reactor system to accidents? What is an appropriate structural design basis of a given reactor system?	Structural design basis of reactor must be evaluated for licensing.	Verified codes for structural response of primary vessel now available. Preliminary codes for piping system now available.	3-D codes for asymmetric systems and head response required. 3-D piping codes for analysis of loop systems required. Fracture mechanics must be considered in code application.	Ability to analyze system structural response will not pace forward.
↑ LOA ↓	Post accident heat removal (8)	Are ex-vessel or in-vessel core retention systems required to accomodate postulated core meltdowns, or can this be accomplished without explicit devices for this purpose? (e.g., inherent core retention, either in-vessel or ex-vessel.)	Requirements for systems to add to core retention capability will impact design. Ultimate disposition of core debris must be shown for core disruptive accidents.	Inherent capability for PAHR in FFTF. Requirements for other systems remain to be established.	Can inherent in-vessel coolability be shown? Can inherent ex-vessel coolability be shown by passive means? Will engineering devices be required?	Combination of inherent features and/or simple design features should be adequate.
4	Radiological source term (9)	What radiological source term should be used for evaluating containment systems?	Level of containment/ confinement required is determined by radiological source term.	Models exist for characterizing behavior of radiological materials behavior and transport from secondary containment to site boundary.	Models for transport and attenuation of radiological materials within primary system required. Radiological consequences associated with postaccident heat removal not well defined.	Resolution of this issue depends on resolution of energetics and PAHR issues. Engineered containment and filter systems can provide ultimate reduction in dose at site boundary to acceptable levels.

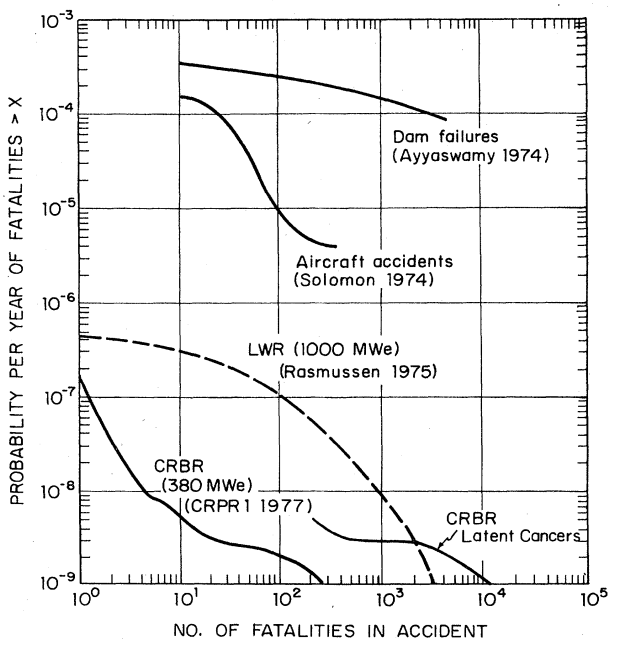

FIG. 19. Comparison of cumulative probability distribution per reactor year for early fatalities due to LWR accidents with early fatalities due to CRBRP accidents.

B. Probability assessment

In this paper I have not made an assessment of the probability of the various events occurring, using what are now conventional methods (Rasmussen, 1975). The study of light-water reactor safety used component reliabilities from operating power plants in many parts of the analysis. Since there are few operating LMFBR's, assessment of component reliabilities is less supportable, and therefore the whole analysis for an LMFBR is less accurate than for an LWR.

However, a preliminary study has been done for CRBR (CRBRP, 1977). The results are shown in Fig. 19, compared with a light-water reactor safety analysis (Rasmussen, 1975). This study supports an implication in much of the present paper. While an accident may be as probable for an LMFBR as for an LWR, in most cases the energy release may be small enough to be contained. However, a large release, by recriticality for example, cannot be excluded. This, although of low probability, might be quite large. These considerations, and use of a single non-urban site, give a different shape to the curve than the reactor safety study. Both give accidents smaller or comparable to the maximum calculated for dam failures (Ayyaswamy *et al.*, 1974; Sherard *et al*, 1963), aircraft accidents (Solomon *et al.*, 1974), or toxic chemical spills (Simmons *et al.*, 1974). It should be noted that only *prompt* effects have been included and latent cancers can be 30 times greater (CRBRP, 1977).

C. Tasks for the future (major tasks)

(1) The probability that a liquid metal fast breeder reactor (LMFBR) will undergo an accident leading to a whole core meltdown should be no greater than for a light-water reactor (1 in 20 000 reactor years according to Rasmussen). With careful design and attention to detail it can be much less. It may be that present reactors and procedures already preclude a core melt to a lower probability than this. However, this is very hard to demonstrate, and some thought can usefully be given as to how to prove this if it is true. If the inherent safety of present reactors can be demonstrated, all other conclusions become less important.

(2) A great amount of work has been spent on tracing the possible course of a meltdown accident if it is ever initiated. The calculations lead to a meltdown with no appreciable energy release—a fizzle rather than a bang. These calculations depend on complex modelling and computer codes and may not be reliable, although as work proceeds, it seems that the early calculations were overly pessimistic. Work should still proceed on postulating—and hopefully refuting—scenarios where major energy releases occur.

(3) Large energy releases can only follow from large *rates of increase* of reactivity. The calculation of the *thermal* energy release is reasonably well determined once the rate of increase of reactivity is defined. Reactor designers in the last 20 years have found some reliable methods to restrict rates of reactivity increase based on proven physical principles. Such searches must continue, and in some cases the searches can be supported by experiment.

(4) The conversion of thermal to mechanical energy for fuel–coolant interactions depends on the working fluid. A thermodynamic maximum of 30% has been suggested, but all existing experiments and detailed models suggest figures much less than this. The critical parameter is probably the impact on the head of the reactor vessel, and no model yet computed brings this efficiency higher than 5%. Studies should continue.

(5) The conversion of thermal to mechanical energy by means of liquid/liquid explosions has been suggested as a way of obtaining high conversion efficiencies. There now exists a theory which seems to rule out such explosions in LMFBR geometries. This theory should apply to all liquid/liquid interactions and not merely to LMFBR situations. The theory can and should be tested by those qualified, even those who may not be aware of the details of breeder reactors.

(6) One of the ways that has been postulated of obtaining high ramp rates involves high sodium and clad void coefficients. These coefficients might be worse for large breeder reactors than for the present small breeder reactors if the design is not changed. However, at some capital cost and reduction in breeding gain, large coefficients can be designed out of the system. Since a high breeding gain is not helpful if public opinion will not allow breeders, a design with a lower gain may be useful until safety can be otherwise demonstrated.

(7) For the Clinch River Breeder Reactor, some pessimistic assumptions suggest a maximum ramp rate due to recompaction of the core of \$50/second (Bohl *et al.*, 1975). This leads to a theoretical thermal energy increase of 5560 megajoules. The models suggest that the maximum kinetic energy available to a slug of sodium impacting on the pressure vessel head would be 72 megajoules, and a total mechanical energy release by expan-

sion to one atmosphere of 350 MJ. The preliminary safety analysis report (PMC, 1975) is more pessimistic and assumes that the total mechanical energy release by expansion to 1 atm can be as high as 660 MJ. The reactor vessel and head are designed for an energy release of 100 MJ to the point of sodium slug impact with the head, with some engineering margin of safety, so that even if the 72 MJ were to be somewhat exceeded, fracture would not be expected. Calculations give similar energies for other reactors. An "acceptable value" must be decided.

(8) A further concern is recriticality, even if the recriticality occurs with a low rate of reactivity increase and therefore energy release. Will there be several recriticalities, or will the reactor eventually shut down completely? The answer so far is twofold: it probably will not attain criticality and it will not matter if it does, provided decay heat removal is maintained.

D. Tasks for the future (smaller tasks)

(1) A search should continue for passive devices, such as yielding core restraint systems which can slow the development of a transient. This may assist inherent safety by allowing delayed remedial action. It is important not to include devices which may affect operational safety by putting greater demands on the system in a more probable manner such as flywheels on the sodium pumps.

(2) Tests of fuel failure processes so far show fuel sweepout. Conditions should be found, if possible, under which this *does not* occur, so as to be sure we are far from an important threshold.

(3) Sodium coolant boiling studies should continue, both experimentally and theoretically, to find conditions under which coherent sodium boiling and voiding occur to be sure that reactor accident conditions are far from a threshold. Such experiments might artificially vary the power distribution for the purpose.

(4) All the computer programs must continue to be tested against each other and against simple physical ideas.

(5) Any important issues which safety experts believe are disposed of must be properly documented; these would include cases similar to the Doppler coefficient calculation and measurement supported by SEFOR tests. Sodium superheat is a case in point. Until this is done, the disposition of these issues cannot be adequately demonstrated to those with a need to make judgments concerning LMFBR safety.

ACKNOWLEDGMENTS

Various aspects of this report have been discussed with many people. In particular, I would like to thank Dr. R. D. Smith, UKAEA, Risley; Dr. R. Breem and Dr. Raj Sehgal, EPRI; Dr. Hans Fauske, ANL; Dr. R. Semeria, CENG, Grenoble; Dr. H. Kottowski, ISPRA, Italy; Dr. W. Stratton, LASL; Dr. Dieter Smidt, Karlsruhe, Germany; Dr. M. S. Plesset, California Institute of Technology; and Mr. John Graham, Westinghouse.

GLOSSARY OF TERMS

ACPR	Annular Core Pulsed Reactor
ACRS	Advisory Committee on Reactor Safeguards (advisory to the AEC now to NRC)
AEC	Atomic Energy Commission (obsolete; see ERDA)
ATWS	Anticipated Transient Without Scram
BOL	Beginning of Life (of reactor core)
BWR	Boiling Water Reactor (water is boiled directly in the reactor)
CRBR	Clinch River Breeder Reactor (an LMFBR of 350 MWe proposed to be built at Clinch River, Tennessee)
DFR	Dounreay Fast Reactor, Scotland
EBR 1, EBR 2	Experimental Breeder Reactors 1 and 2 (built by Argonne National Laboratory)
ECCS	Emergency Core Cooling System (of importance mainly in light-water reactors)
EOEC	End of Equilibrium Core. EOEC, BOL, and EOL all correspond to various stages of burn-up of the reactor fuel
EOL	End of Life (of reactor core)
ERDA	Energy Research and Development Administration. U. S. agency taking over many activities of the AEC and other agencies
FCI	Fuel/Coolant Interaction. For an LMFBR we are concerned with a uranium-oxide/plutonium-oxide mixture interacting with sodium
FFTF	Fast Flux Test Facility (A fast test reactor being finished at Hanford, Washington to study fuel performance)
FTR	Fast Test Reactor
GCFBR	Gas Cooled Fast Breeder Reactor. The gas is helium
GODIVA	Two hemispheres of ^{235}U metal used for critical mass measurements and for power transient measurements
HCDA	Hypothetical Core Disruptive Accident (sometimes hypothetical core disassembly accident). An accident which is terminated by disassembly of the reactor core
LOA	Line of Assurance; It is supposed that the progression of a fast breeder accident can be stopped by one or another lines of assurance
LOCA	Loss-of-Coolant Accident (of importance mainly in light-water reactors)
LMFBR	Liquid Metal Fast Breeder Reactor. The liquid (metal sodium or sodium/potassium eutectic) is the coolant
LWR	Light-Water Reactor (usually one of two types: BWR and PWR)
NRC	Nuclear Regulatory Commission (taking over the regulatory activities of AEC)
PAD,VENUS, REXCO, SUREX RX2-POOL	Computer codes for studying reactor disassembly
PAHR	Post Accident Heat Removal
PBR	Power Burst Facility (a pulsed test reactor)
PFR	Prototype Fast Reactor (built at Dounreay, Scotland)
PLBR	Prototype Large Breeder Reactor (of about 1000 MWe being designed in the U. S.)
SAREF	Proposed Safety Related Experimental Facilities including a modified EBR-II
SAS 2, SAS 3, SAS 3D	Various stages of a sophisticated computer code for studying the beginning stages of a reactor accident

SCRAM	Rapid emergency shutdown of a reactor by rapid insertion of shutdown control rods
SEFOR	Southern Experimental Fast Oxide Reactor built by General Electric Co., explicitly to study the Doppler effect and its ability to terminate a reactor excursion
SLSF	Sodium Loop Safety Facility (a safety test loop in a reactor)
SIMMER	Reactor accident code to study in detail core melt behavior being prepared for NRC
SL 1	An army operated reactor, water moderated, in Idaho which was destroyed by an accidental disassembly
SNAPTRAN 1,2,3	Explosive uranium disassembly tests. Snap =Space Nuclear Applications
SNR 300	A German LMFBR under construction
TOP	Transient Over Power (accident) which is only serious if accompanied by a failure to SCRAM (unprotected TOP accident)

REFERENCES

Amblard, M., J. Costa, S. Fabrega, C. Lackme, and P. Symtaeious, 1974, "Out of Pile Studies in France on Sodium-Fuel Interactions," in *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, California, April 1974 (American Nuclear Society Conf. 740401-P2).

American Nuclear Society, 1974, *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, California, April, 1974 (Conf. 740401-P2), all Chapter 13.

Armstrong, D. R., F. J. Testa, and D. Haridon, 1971, "Interaction of Sodium with Molten UO_2 and Stainless Steel Using a Dropping Mode of Contact," ANL Report 7890.

Ayyaswamy, P., B. Hauss, T. Hseight, A. Moscati, T. E. Hicks, and D. Okrent, 1974, "Estimates of the Risks Associated with Dam Failure," UCLA–ENG 7423, March 1974.

Bell, G. I., and S. Glasstone, 1970, *Nuclear Reactor Theory—An Advanced Course* (Van Nostrand Reinhold, New York).

Bethe, H. A., 1974, Cornell Workshop on the Major Issues of a National Energy R & D Program, College of Engineering, Cornell University, Ithaca, NY 14850.

Bethe, H. A., and J. H. Tait, 1956, "An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses," U.S.-U.K. Reactor Hazard Meeting, RHM (56)/113, April 1956.

Beyman, T. D., 1974, "The Physics of Fast Reactors", Rep. Prog. Phys. **37**, 951.

Bleiweis, P. B., D. Okrent, and W. E. Kastenberg, 1974, "A Comparison Study and Analysis of Models for 1000 MWe and 4000 MWe LMFBR Hypothetical Accidents," in *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, California, April 1974 (American Nuclear Society Conf. 740401-P2), p. 1322.

Board, S. J., 1974, "A Mechanism for UO_2/Sodium Interaction," Nucl. Sci. Eng. **54**, 233.

Board, S. J., R. B. Duffey, C. L. Farmer, and D. H. Poole, 1973, Nucl. Sci. Eng. **52**, 433.

Board, S. J., and R. W. Hall, 1974, "Propagation in Thermal Explosions: I Tin/Water Experiments," Central Electricity Generating Board (UK) Report, RD/B/N2850.

Board, S. J., R. W. Hall, and G. E. Brown, 1974, "The Role of Spontaneous Nucleation in Thermal Explosions: Freon/Water Experiment," Research Dept., Berkeley Nuclear Laboratories (UK) RD/B/N3007.

Bohl, W. R., H. K. Fauske, D. R. Ferguson, K. J. Henniger, and R. W. Ostenson, 1975, "Analysis of Transient Undercooling and Transient Overpower Accidents without Scram in the CRBR," Preliminary Report ANL/RAS 75–29.

Booth, D. L., D. C. Menzies, and M. G. Chasanov, 1974, "Reactor Development Progress," Rept. ANL–RDP–22, p. 743.

Bott, T. F., and J. F. Jackson, 1976, "Experimental Comparison Studies with the VENUS-II Disassembly Code," International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, October 1976 (American Nuclear Society).

Boudreau, X., and J. F. Jackson, 1974, "Recriticality Consideration in LMFBR Accidents," in *Proceedings of the Fast Reactor Meeting*, Beverly Hills, California, April 1974 (American Nuclear Society Conf.-740401-P2), p. 1265.

Burgess, D. S., J. Biordi, and J. Murphy, 1972, "Hazards of Spillage of LNG into Water," U.S. Bureau of Mines, PMSRC Rept. No. 4177.

Carslaw, H. S., and J. C. Jaeger, 1959, *Conduction of Heat in Solids* (Clarendon, Oxford).

Catton, H., 1975, Dept. of Engineering, University of California at Los Angeles, private communication.

CGUS, 1975, "The Liquid Metal Fast Breeder Program, Past, Present and Future," Comptroller General of the U. S.

Cho, D. M., *et al.*, 1971, "Pressure Generation by Molten Fuel Coolant Interactions under LMFBR Accident Conditions," in *Proceedings of Conference on New Developments in Reactor Mathematics and Applications*, Idaho Falls, Idaho, March 1971.

Cho, D. M., *et al.*, 1972, "A Rate-Limited Model of Molten Fuel/Sodium Interactions: Model Development and Preliminary Calculations," ANL Rept. 7919, March 1972.

Colgate, S. A., and T. Sigurgeirsson, 1973, "Dynamic Mixing of Water and Lava," Nature **244**, 552.

Combustion, 1970, "1000 MWe LMFBR Follow-on Study, Report of Tasks II and III. A Reference Conceptual Design" Combustion Engineering, Inc., Rept. CEND-377, Vol. 3.

CRBRP, 1977, "Clinch River Breeder Reactor Risk Analysis," Produced for U. S. Energy Research and Development Administration by Westinghouse Electric Corp.

Dickerman, C. E., E. Q. Barts, A. DeVolpi, R. E. Holtz, W. F. Murphy, and L. H. B. Rothman, 1975, "Recent Results from TREAT Tests on Fuel, Cladding and Coolant Motion," First European Nuclear Conference, Paris, April 1975, Annals of Nuclear Energy **3**, 315.

Farmer, F. R. 1970, Preface to *An Application of Fast Reactor Safety*, UK Atomic Energy Authority (S) R188.

Farmer, F. R., 1977, Editor, *Reactor Safety* (Academic, New York).

Fauske, H. K., 1972, "The Role of Nucleation in Vapor Explosions," Trans. Am. Nucl. Soc. **15**, 813.

Fauske, H. K., 1973, "On the Mechanism of Uranium Dioxide-Sodium Explosive Interactions," Nucl. Sci. Eng. **51**, 95.

Fauske, H. K., 1975, Argonne National Laboratory, private communication.

Fauske, H. K., 1976, "The Role of Core Disruptive Accidents in Design and Licensing of LMFBR's," Nucl. Safety **17**, 550.

Fischer, E. A., and H. Keller, 1966, "Influence of the Segregation of Oxide Fuel on the Course of Power Excursions," Nucleonik **8**, 471.

Fischer, E. A., P. R. Kinsman, and R. W. Ohse, 1976a, "Critical Assessment of Equation of State Data for UO_2," J. Nucl. Mater. **59**.

Fischer, E. A., P. R. Kinsman, and R. W. Ohse, 1976b, "Extension of Vapor Pressure Measurements of Nuclear Fuels to 7000°K for Reactor Safety Analysis," J. Nucl. Mater. **59**.

Fullwood, R. R., *et al.*, 1976, "ATWAS, A Reappraisal," EPRI Rept. NP 251.

Galanin, A. D., 1960, *Thermal Reactor Theory* (Pergamon, Oxford and New York).

GEC, 1970, General Electric Company Reports, GEAP-13702, GEAP-13837, and GEAP-13838.

Gilby, 1970.

Glasstone, S., and M. C. Edlund, 1952, *The Elements of Nuclear Reactor Theory* (Van Nostrand, New York).

Goldhammer, H., and H. Kottowski, 1975, "Calculational Model and Code for Fuel Coolant Interaction," Working paper for

OECD Meeting on Calculational Models, April 28-29, 1975.
Graham, J., 1971, *Fast Reactor Safety* (Academic, New York).
Graham, J., 1974, "Nuclear Safety Design of the CRBR," Nucl. Safety **16**, 564.
Halliwell, N. A., and R. C. Pottmeyer, 1975, "Thermal Interaction Experiments with UO_2 and Sodium Using Pyrotechnic Heating," Nucl. Eng. and Des. **32**, 311.
Hannum, W. H., J. D. Griffith, and A. C. Millunzi, 1975, "Reactors–Safe At Any Speed,"Trans. Amer. Nucl. Soc. **22**, 483.
Henry, R. E., J. D. Gabor, I. O. Winsch, E. A. Spleha, D. J. Quinn, E. G. Erickson, J. J. Heiberger, and G. T. Goldfuss, 1974, "Large-scale Vapor Explosions," in *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, California, April 1974 (American Nuclear Society, Conf.–740401-P2), 922.
Henry, R. E., and H. Fauske, 1975a, "Energetics of Vapor Explosions," American Society of Mechanical Engineers Heat Transfer Conference, California, August 1975.
Henry, R. E., *et al.*, 1975b, "Experiments on Pressure Driven Fuel Compaction," Trans. Am. Nucl. Soc. **22**.
Hetrick, D. L., 1971, *Dynamics of Nuclear Reactors* (University of Chicago, Chicago).
Hicks, E. P., and D. C. Menzies, 1965, "Theoretical Studies on the Fast Reactor Maximum Accident," in *Proceedings of the Conference on Safety Fuels and Core Design in Large Fast Power Reactors*, ANL Rept. 7120, p. 654.
Hummel, H. H., P. Graebler, W. B. Loewenstein, J. B. Nims, A. B. Reynolds, and T. A. P. Herle, 1966, "Recent Theoretical Work in the U. S. on Fast Breeder Reactors," in *Proceedings of Conference on Fast Breeder Reactors* (London), p. 493.
Hummel, H. H., Kalimullah, and P. A. Pizzica, 1976a, "Limits on Sodium Voiding Reactivity Addition Rate for Very Large LMFBR's," Nucl. Sci. Eng. **59**, 440, also ANL Rept. 76-77.
Hummel, H. H., P. A. Pizzica, and Kalimullah, 1976b, "Studies of Unprotected LOF Accidents for CRBR," ANL Rept. 76–51.
Isbin, H. S., 1963, *Introductory Nuclear Reactor Theory* (Reinhold, New York).
Ishikawa, M., N. Hirakawa, and K. Tasaka, 1970, "A Study of Destructive Energy Released at Fast Breeder Reactor Maximum Hypothetical Accident," Japan Atomic Energy Research Institute, JAERI-1188, 470.
Jackson, J. F., *et al.*, 1970, "Pressurization Rate Effects in Irradiated Core Disassembly Calculations," Trans. Am. Nucl. Soc. **22**, No. 370.
Jackson, J. F., and R. B. Nicholson, 1972, "VENUS-II and LMFBR Disassembly Program," ANL Rept. 7951.
Kastenberg, W. E., and I. Catton, 1975, "Progress Report of the U.S.A.E.C. On Safety Consideration of Commercial LMFBR's," December 26, 1974.
Katz, D. L., 1972, "Superheat-Limit Explosions," Chem. Eng. Prog. **68**, 68.
Katz, D. L., 1975, talk at Asilomar Conference on Risk/Benefit Analysis, September 1975.
Katz, D. L., and C. M. Sliepcevitch, 1971, "LNG/Water Explosions: Cause and Effect," Hydro. Proc. **50**, 240.
Kazimi, M. S., and J. C. Chess, 1974, "A Condensed Review of the Technology of Post Accident Heat Removal for the LMFBR," Brookhaven National Laboratory Report BNL/FRS 74–3.
Keepin, G. R., 1965, *Physics of Nuclear Kinetics* (Addison-Wesley, Reading, Mass.), Section 9-3.
Kramerov, A. Y., and Y. V. Shevelev, 1964, *Engineering Analysis of Nuclear Reactors*.
Lewis, H. W., *et al.*, 1975, "Report on Light-Water Reactor Safety," Rev. Mod. Phys. **47**, Suppl. 1.
Long, G., 1957, "Explosions of Molten Aluminum in Water: Cause and Prevention," Met. Prog. **71**, 5, 107.
Mantamakas, M., M. Fortunator, A. Meyer Heine, and P. Francois, 1974, "Some Problems with Fast Reactor Safety," in *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, California, April 1974 (American Nuclear Society Conf. 740401-P2), p. 1290.
Massey, J. V., L. Lois, T. R. Speis, and T. G. Theofanous, 1976, "A Method for Assessing Core Wide Incoherency Effects on LMFBR Whole Core Accident Evolution," Report to the Fast Breeder Safety Conference, Chicago 1976, and Purdue University Report PNE–111–76.
McCarthy, W., and D. Okrent, 1964, "Fast Reactor Transients," in *Technology of Nuclear Reactor Safety*, edited by T. J. Thompson and J. G. Beckersly (MIT, Cambridge, Mass.).
McLaughlin, T. P., 1975, "Preliminary Topical Report on Comparison Reactor Disassembly Calculations," Los Alamos Report, LA 6106-MS, September 1975.
McLaughlin, T. P., J. F. Jackson, H. M. Forehand, J. J. Koelling, and W. R. Stratton, 1976, "LMFBR Disassembly Analyses," International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, October 1976 (American Nuclear Society).
Meyer, R. A., and B. Wolfe, 1968, "Fast Breeder Meltdown Accident," Adv. Nucl. Sci. Technol. **4**, 197.
Michelson, A. A. 1892, Phil. Mag. **34**, 280.
Nakanishi, E., and R. C. Reid, 1971, "Liquid Natural Gas-Water Reactions," Chem. Eng. Prog. **67**, 36.
Nicholson, R. B., 1962, "Methods for Determining the Energy Release in Hypothetical Reactor Meltdown Accidents," Atomic Power Development Associate, APDA 150, December 1962.
Nordheim, N. W., 1965, in *Technology of Nuclear Reactor Safety*, edited by T. J. Thompson and J. G. Beckersly (MIT, Cambridge, Mass.), Chap. IV.
NRC, 1976, Letter from R. Denise of Nuclear Regulatory Commision Staff to Project Management Corporation, May 6.
Petit, J. F., 1975, "Creys-Malville Plant (SUPERPHENIX) Main Inherent and Design Safety Characteristics," Trans. Am. Nucl. Soc. **22**, No. 479.
Porteous, W., 1975, "Superheating and Cryogenic Vapor Explosions," Ph.D. Thesis, MIT, Cambridge, MA 02139.
Porteus, W., and R. C. Reid, 1976, "Liquid Hydrocarbon Vapor Explosions," Chemical Engineering.
PMC, 1975, Project Management Corporation, Clinch River Breeder Reactor Project, Preliminary Safety Analysis Report, Appendix F, Core Disruptive Accident Accommodation.
Puig, I., and A. Szeless, 1972, "Evaluation de la dispersion du combustible chaud du sodium," First CREST Meeting on Sodium Fuel Interaction, Grenoble, France, January 1972.
Rasmussen, N. C., 1975, *An Assessment of Accident Risks in the U. S. Commercial Nuclear Power Plants*, WASH-1400, NUREG 75, "The Rasmussen Report."
Rayleigh, Lord, 1888, Phil. Mag. **27**, 298.
Reid, R. C., 1976, "Superheating Liquids, Am. Sci. **64**, 146.
Sherard, *et al.*, 1963, *Earth and Earth Rock Dams* (Wiley, New York).
Simmons, J. A., R. C. Erdmann, and B. N. Naft, 1974, "The Risk of Catastrophic Spills of Toxic Chemicals," UCLA-ENG 7425.
Smith, R. D., 1970, "Report on Instrumentation," Karlsruhe Conference on Fast Reactor Safety, Karlsruhe, Germany.
Solomon, K. A., R. C. Erdmann, T. E. Hicks, and D. Okrent, 1974, "Airplane Crash Risk to Ground Population," UCLA-ENG 7424, March (plus additional information from Canary Islands accident, 1977).
Stratton, W. R., L. B. Engle, and D. M. Peterson, 1972, "Reactor Power Excursion Studies," in *Proceedings of Conference on Engineering of Fast Reactors for Safe and Reliable Operation*, Karlsruhe, Germany, p. 1331.
Stratton, W. R., 1975, at April 4 meeting of the Advisory Committee on Reactor Safeguards, Nuclear Regulatory Commission.
Stratton, W. R., and T. P. McLaughlin, 1976, private communication.
Thompson, T. J., 1964a, in Thompson and Beckersley, *Technology of Fast Reactor Safety* (MIT, Cambridge, Mass.), p. 671.

Thompson, T. J., 1964b, in Thompson and Beckersley, *Technology of Fast Reactor Safety* (MIT, Cambridge, Mass.), p. 625.

U. S. Atomic Energy Commission, 1973, "Report on Anticipated Transients Without Scram (ATWS)," Regulatory Staff of the AEC, WASH 1270.

U. S. Energy Research and Development Administration, 1975, "A Compendium of Computer Codes for the Safety Analysis of LMFBR's," ERDA-31.

Vessily, W., 1977, Nuclear Regulatory Commission Report, July.

Weinberg, A. M., and E. P. Wigner, 1958, *The Physical Theory of Nuclear Chain Reactors* (University of Chicago, Chicago).

Witte, L. C., J. E. Cox, and J. E. Bouvier, 1970, "The Vapor Explosion," J. Met. **22**, 39.

Witte, L. C., T. J. Vyas, and A. A. Gelabert, 1973, "Heat Transfer and Fragmentation during Molten-Metal/Water Interactions," J. Heat Transfer **95**, 521.

Wolfe, B., N. Friedman, and D. Riely, 1963, "Influence of the Doppler Effect on the Meltdown Accident," in *Proceedings of the Conference on Breeding Economics and Safety in Large Power Reactors*, October 1963, ANL Rept. 6792, p. 171.

Wright, R. W., J. J. Barghusen, S. M. Ziri, M. Epstein, R. O. Ivins, and R. W. Mouring, 1974, "Summary of Autoclave TREAT Tests on Molten Fuel Coolant Interactions," in *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, California, April 1974 (American Nuclear Society Conf. 740401-P2).

Yellin, J., 1976, Reply to Comment, Bell Journal of Economics 7, 711.

Zaleski, P., 1975, private communication, Electricité de France and Technicatome, France.

Zin, S. M., M. Epstein, R. W. Wright, J. J. Barghusen, D. H. Cho, F. J. Testa, G. T. Goldfuss, and R. W. Mouring, 1975, "An In-Pile Study of Thermal Interactions between High Energy Molten UO_2 Fuel and Liquid Sodium," Nucl. Sci. Eng. **56**, 229.

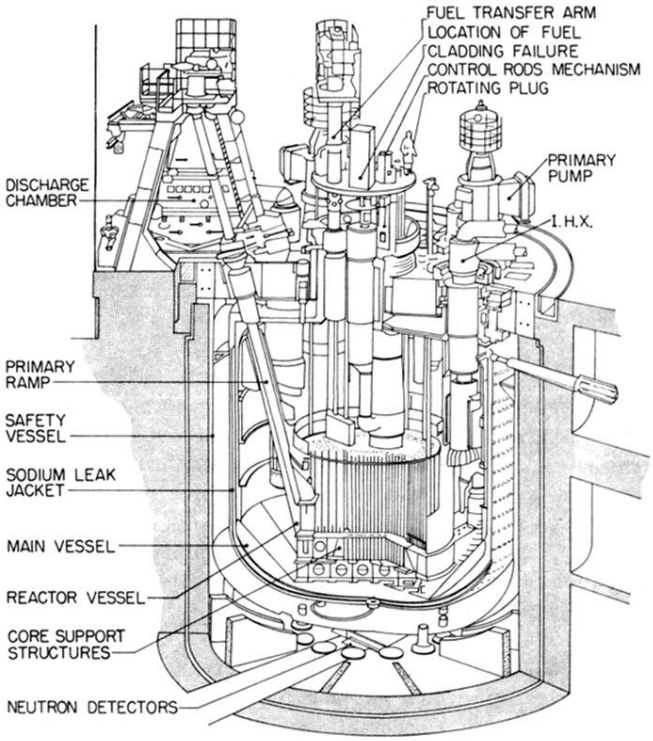

FIG. 3. Details of the PHENIX reactor, Marcoule, France.