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SAFETY PROBLEMS OF LIQUID-METAL-COOLED FAST BREEDER REACTORS

by

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R. O. Ivins, W. C. Lipinski, D. R. MacFarlane,
J. F. Marchaterre, D. Miller,
R. B. Nicholson, and R. M. Singer

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Reactor Technology

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ABSTRACT

This report reviews the safety problems related to the design of cores for liquid-metal-cooled fast breeder reactors. A phenomenological representation is related to potential accidents, and this representation is used to describe the program of work needed to develop adequate design principles. Some of the program requires massive in-pile installations for execution, and some is typical of work carried out in many university and industrial laboratories. In the summary, emphasis is placed on the latter work program.

INTRODUCTION

It is our intent that this collection of articles be used as a source of reference material for further research in liquid-metal-cooled fast-breeder-reactor (LMFBR) safety by university students and staff, as well as industrial and laboratory workers. The material in this report is based on work prepared for the project "Studies of LMFBR Safety Facilities" initiated at Argonne National Laboratory in 1968. The purpose of that Project was to define and justify a program of in-core tests and the facilities needed to carry out a program to establish the safety properties of LMFBR cores. Dr. Robert Duffield, Director of Argonne National Laboratory, pointed out that much of the material used in that project would be of general utility and inspired this report.

There is no recognized discipline called "Safety," let alone specialties in reactor safety or even LMFBR safety. There is no standard format for safety discussions. Judgments of relative priority frequently reflect no more than a consensus of estimates as to which problems create the most anxiety at any one time. We therefore claim no special role for this collection of documents, nor do we assert that it has any primacy as opposed to other, more scattered, work. Whatever value resides in this report is in the fact that it represents a collection of work centering on one major area of reactor safety and attacking major problems in that area from a number [illegible] f view simultaneously. It has long been the history of

work in reactor safety to approach problems in piecemeal fashion, so that at some moment it is the Doppler effect, at another the void coefficient, and at still another the power coefficient that is uppermost in peoples' minds. Although there is no question that each of these problems is significant, the uninitiated person may easily come to regard the problem posed at the moment as being the whole of reactor safety rather than just one significant part. Safety is a study of the interrelated effects of different malfunctions, and the emphasis should therefore be placed on the interrelatedness of problems rather than on the particular nature of any single problem.

We have divided the material in this report largely along phenomenological lines--partly, because this is the way it was done in the original report for which the material was first prepared; partly, because we are interested in inspiring attacks on outstanding problems rather than inspiring design changes that will circumvent problems. The latter are necessary at various stages of any reactor design, but the attack on the problems themselves is needed to uncover sound basic design principles. There are, however, many useful ways to organize the work of reactor safety, and the one presented here may not necessarily be well suited for other purposes.

We start with a chapter on the relationship of LMFBR safety problems to reactor safety in both the historical context and relative to work currently being done. Next we treat the particular problems of LMFBR safety and attempt to categorize them as those of which the solution is essential to the design of large economical plants and will play a significant role in determining the optimum design, or as those problems that are either well on their way to solution or which, while they are significant to the safe design of a reactor plant, are not liable to affect the design in a qualitative way.

Because it is so closely related to traditional engineering studies and to the steady-state design problems frequently encountered in course work in nuclear engineering, we then introduce the topic of coolant dynamics. The problems associated with unstable boiling and possible superheat lead not only to a discussion of the possible reactivity effects of voids in the sodium, but also to the possible failure of the fuel through insufficient heating. This leads naturally to the next chapter, which is a discussion of the propagation of failure of fuel from one fuel pin to its nearest neighbors.

If such autocatalytic propagation were to occur, it could then lead to a situation where extremely hot, essentially uncooled fuel were eventually to mix with relatively cool sodium, and the results could be a violent boiling process akin to an explosion. Such violent boiling can occur on several scales, and it is not restricted in principle to the single-fuel-pin problem.

Indeed, the one case observed in an accurate in-pile simulation, the SPERT-1D test, involved the melting of a major portion of the core. The accident phenomenon associated with the melting of fuel on a subassembly scale or larger is discussed next.

We complete the sequence by a discussion of the most severe accidents, where the entire reactor core participates in liberating energy at an extremely high rate until its energy content causes the reaction to cease by disassembly. Following such an accident, there would be a need for containment of the debris, which will be heated by retained fission products. This problem is reviewed next, and we close with a discussion of problems centering on the development and testing of instrumentation.

CHAPTER I

PERSPECTIVE VIEW OF LMFBR SAFETY

Safety in large fast power reactors should be seen against the background of reactor safety, in general, and the influences of LMFBR design.[1] From such a perspective, we try to determine those problems we expect to solve by means similar to those employed in the past, we expect to determine those problems that are essentially new, and we expect to determine those problems that are highly sensitive to design limitations.

A. Comparison of LMFBR's and Other Reactor Types

The technical feasibility of building and operating nuclear power reactors of various types and size without endangering public safety has been demonstrated many times. Reactor safety as a topic would not loom large were it not for the continuing pressure of the need for more economical power sources which give rise to new types of reactor design and design modifications. Each power-reactor type brings characteristic safety questions into consideration. For example, the high-pressure gas-cooled reactor is sensitive to primary system rupture with loss of coolant and depressurization. Should such a loss occur, it is important to minimize and control any reaction between air and the graphite. Similarly, leaks from the steam system into the primary system coolant pose significant questions. The boiling and pressurized water reactors (BWR's and PWR's) operate with high-pressure coolant and thus also are potentially subject to accidents involving depressurization and loss of coolant. The sodium-cooled fast power reactors employ a low-pressure coolant, which will not flash normally in the event of a primary system rupture. The sodium air reaction has received some emphasis, although primary systems are usually surrounded by an oxygen-depleted atmosphere.

In common with pressurized water reactors using chemical shim control, the liquid-metal-cooled fast breeder reactors (LMFBR's) may

frequently have a positive void coefficient. In PWR's, the local temperature may, under conditions where power and flow are no longer properly matched, become so high as to cause a departure from nucleate boiling (DNB), and this will lead to film boiling and, if the condition is allowed to continue, overheating and melting of the fuel. In LMFBR's, current design conditions are such that boiling is generally unstable, and if there is void formation, the void is expected to propagate rapidly through the channel with subsequent overheating of the fuel. In distinction to the PWR's, however, the region of positive void coefficient in the large LMFBR's can be very extensive, and the amount of reactivity added through such voiding can be very large. Moreover, the small prompt neutron lifetime emphasizes the effects of the rate of reactivity insertion. In addition, whereas the thermal reactors operate with low-enrichment fuel, the enrichment of fuel in the LMFBR's is high enough that there is a potential for large reactivity insertion rates from coherent collapse of the core.

These considerations have frequently led in the past to the belief that a reactivity accident presents the greatest hazard. Because the coolant system pressure in LMFBR's is low, the reactivity insertion rates from malfunctions such as control-rod ejection are not likely to be extremely high, and such accidents are not normally considered to be likely. More reasonable is the assumption that if precautions are not taken, either through design methods or the installation of proper protective systems, local faults on the subassembly or individual fuel-element scale may propagate. If this propagation can be autocatalytic, large-scale fuel failure and sodium boiling with consequent large reactivity insertion rates may result. Therefore the emphasis in LMFBR systems is on the questions of local anomalies, which means detecting such anomalies and determining the potential of propagation of damage within each subassembly and from subassembly to subassembly.

B. Safety Problems Specific to LMFBR's

In the following chapters we will restrict our discussion by and large to LMFBR's of the type currently being considered as likely candidates for central station power production--that is, 1000-MWe designs. Many of the considerations reported in this document apply to designs with greater potential but with less active development at this time. But, if past experience is any guide, we should be prepared to expect that such future developments will bring their own characteristic problems.

The typical large fast reactor under active design today employs a mixed-oxide (PuO_2, UO_2) fuel, helium bonded to stainless steel cladding having a diameter of about 0.250 in. The core volume is a little more than one-third fuel, the coolant is a little less than a half of the total volume, and the remainder is taken up by cladding and structural material. The enrichment of the fuel tends to be about 20% plutonium or somewhat less.

Typical peak oxide fuel-element ratings are 15 kW/ft, with a peak power density of about 600 kW/liter of core.

Such reactors typically have a substantially negative Doppler coefficient of reactivity coming from the fertile ^{238}U. This Doppler coefficient can be remarkably reduced in size if the sodium should be voided.

Generally speaking, in these large sodium-cooled fast reactors, loss of sodium gives an appreciable positive reactivity change. This effect comes from the lack of balance between two competing factors. First, the loss of sodium tends to harden the neutron spectrum, and in the hardened spectrum, the ratio of capture to fission in the plutonium decreases, the fast fission yield from the fertile atoms increases, and the capture in the fertile nuclei and the fission products decreases, relative to the increase in the fission cross section of the fissile nuclides. Thus, this effect adds reactivity. In competition, the leakage increases with the loss of sodium, and this decreases reactivity. As the size of the reactor increases, the neutron leakage tends to become less important and the reactivity increase from spectral hardening tends to dominate.

This competition causes the change in reactivity with loss of sodium to have a strong spatial dependence. Generally, the negative leakage component is small near the core center, so that even for a reactor having an overall zero sodium void coefficient there will be a positive component from the central regions of the core and a negative contribution from the periphery. If sodium were to be expelled beginning at the center of the core, there would be an initial reactivity gain, even though the total loss of sodium led to a loss of reactivity.

Other reactivity coefficients arise from changes in the average fuel concentration in the core such as can arise from the bowing of fuel elements or subassemblies induced by radial and axial temperature gradients. Axial fuel movement from thermal expansion, particularly in the course of a transient power excursion, can also lead to significant reactivity feedback.

Thus, there are a variety of sources of reactivity feedback inherent in LMFBR design. Some of these, like the Doppler coefficient and fuel movement from axial expansion, are prompt; others, like those from sodium voiding, may be prompt or may be delayed slightly, depending on the details of the excursion. Small delays in reactivity feedback can have an important effect on the course of a transient. A direct consequence of the short prompt-neutron lifetime of LMFBR's is the fact that a super-prompt-critical burst may take its course before mechanical control mechanisms could have time to act to produce any significant effect on the outcome of the burst, even though they were triggered at its onset. During such a burst, there may be insufficient time for large-scale conduction of heat from fuel

to coolant, unless the fuel fragments and mixes with the coolant. The fuel in solid form may tend to expand so rapidly that inertial effects will be significant. If fuel displacement is the primary source of reactivity loss to terminate the power burst, very high pressures may result before sufficient expansion takes place. As we have mentioned, an important characteristic of LMFBR's is that they gain reactivity when their fuel is rearranged in a more dense configuration displacing coolant. Large amounts of reactivity are potentially available from anything that moves the fuel into geometrical arrangements of higher average fuel density. Because the short prompt-neutron lifetime limits to a fraction of a dollar the amount of reactivity that can be inserted before the temperature rises sharply and the burst is terminated, it is the rate of reactivity insertion at prompt critical, rather than the total amount of excess reactivity available, that primarily determines the course of events. Following such a burst, there is a considerable amount of energy available for conversion into mechanical work, even though the nuclear reaction has to all intents and purposes ceased.

C. Summary--The Main Problem Areas

We have set the stage for examining three types of LMFBR safety problems. The first type is: local anomalies in the core which may propagate; such anomalies include local sodium boiling induced by disturbances within an individual flow channel or channels, pin failure, and failures (such as coolant flow blockages) that may affect a significant portion of a single subassembly. The second type of safety problem is associated with sources of large rates of reactivity insertion; such sources include coherent motions of fuel and coolant which might arise as the result of large-scale autocatalytic propagation of local core anomalies. The energy generated in such a burst will eventually cause the nuclear reaction to cease by virtue of core disassembly. However, a considerable amount of energy in the expanded core is still available for conversion into mechanical work. Dissipation of this energy and removal of the fission-product decay heat from the dispersed core constitutes the third class of problems, containment and cooling of postburst core debris.

In the succeeding chapters, these three classes of problems are discussed roughly in the order named. There is a considerable amount of interaction between the various problems, however, and strict separation is not always possible. First we discuss specific problems in relation to their importance. We take up next the problems of local anomalies, such as sodium boiling and fuel-failure propagation, at which point it is natural to introduce the topic of molten fuel-coolant interaction. This topic is applicable on all three problem levels, however, since it is applicable to (1) the propagation of pin failure, (2) the propagation of subassembly damage, and (3) the conversion of stored heat to mechanical energy following a severe excursion. Large fuel and coolant motions and severe excursions are treated next, including a discussion of the effects of severe excursions on containment. We close with a discussion of postburst phenomena and instrumentation problems.

CHAPTER II

IDENTIFICATION OF THE IMPORTANT SAFETY QUESTIONS

A. Introduction--Basic Safety Requirements

The first concern of this chapter is to identify the basic requirements on the behavior of a reactor under accident conditions that have to be established to make a judgment about the safety of the reactor.

1. Accident Probabilities

Any accident that may happen to a reactor can be characterized by its probability and its consequences. The safety of the reactor is then determined by a judgment as to whether the probabilities of these consequences, for the entire range of possible accidents, are acceptably low. Thus, to discuss reactor safety, we have to use terms to describe both accident probabilities and accident consequences.

Probabilities can be measured numerically, but at present it is more usual to use terms like "credible" and "incredible," or "design basis" and "hypothetical," without giving these terms explicit numerical meanings. The problem here is that such terms are not always understood in the same way by everyone who uses them.

To avoid this confusion, we shall use the terms "likely" and "unlikely," defining them in the following way:

A likely accident is one that has more than a small probability of occurring during the lifetime of a reactor. It normally involves a single failure (e.g., coolant pump failure without coincident failure of the shutdown system). Clearly, a likely accident must not cause serious damage to the reactor or harm to the public.

An unlikely accident is one we do not expect. It may involve two or more coincident independent failures (e.g., pump failure together with failure of the shutdown system), or a single improbable event (e.g., a major pipe rupture).

This use of "unlikely" has two difficulties. Firstly, it is hard to be sure that apparently independent failures are always independent, and that there are no unsuspected events (e.g., fire or earthquake) which can cause coincident failures.

Secondly, as defined here "unlikely" includes the entire range of very improbable accidents. In practice, however, we have to distinguish

between those unlikely accidents that are nevertheless probable enough that we have to design for them, and those that are so improbable that they can be ignored, however severe their consequences. We can call the latter "very unlikely" accidents.

As the term is currently used, the "design basis accident" is the worst accident with a high enough probability that its consequences have to be designed against. This concept is useful, but is subjected to the disadvantage that it is tempting to fix it in terms of what can be designed against, rather than by judging whether the probability of a worse accident is acceptably low.

Another term sometimes used is the "maximum credible accident." This is less serious and more likely than a "design basis accident." The term is misleading, however, because it is never the worst possible accident. The use of this term implies a hidden judgment about what accidents are so probable that they have to be taken seriously.

2. Accident Consequences

The main concern of reactor safety analysis is with dangers to public health, and the greatest danger is the release of radioactive materials to the environment. (Even in the case of a nuclear explosion, the mechanical damage is likely to be of negligible concern compared with that done by radioactivity.) In most cases, three levels of radioactivity release can be distinguished:

a. Negligible release, such as is caused by an accident that does not breach the primary coolant circuit.

b. Controlled release, due to breach of the primary circuit while the rest of the containment is intact. (In this case, the function of the containment is to keep the radioactivity release rate down to a level that is acceptable in the particular environment.)

c. Uncontrolled release, due to breach of all containment, so that the accident is a major disaster.

A secondary concern of safety analysis is with the economic consequences of accidents, in terms of the cost of repairs and of failure to utilize the plant.

3. The Safety Standard

A reactor safety standard (from either the design or the licensing point of view) is a relationship between consequences and probability, stating what are the acceptable probabilities of various consequences arising. In the simple terms we use above, for example, it must state a

judgment of what are acceptable probabilities for controlled or uncontrolled radioactivity release.

In fact, of course, such a judgment would go into more detail about the acceptability of different accidents, and would be expressed in other terms. (For example, it may be necessary to say that uncontrolled radioactivity release must be "impossible," glossing over the fact that this is an impossible ideal.)

If we are to judge that a given reactor is safe, certain requirements about its behavior in accidents have to be met. These requirements can be stated in different ways, but however expressed, they can be reduced to the following four. We require that:

a. No likely accident can have more than negligible consequences (i.e., it must not breach the primary coolant system).

b. No unlikely accident can have more than limited consequences (i.e., it must not breach the secondary containment).

c. Accidents with serious consequences must be "impossible" (i.e., breach of all containment must be very unlikely).

d. The damage done by a likely accident must be easily repairable (i.e., repairs can be made at reasonable cost).

The argument leading to these requirements can be refined, using, for example, better terms than "likely" and "unlikely," or even going to the detail of numerical probabilities; but however it is expressed, these requirements or their equivalents emerge as basic to reactor safety.

We now have to determine what technical questions these requirements pose for an LMFBR, by considering the entire range of possible accidents.

For likely accidents, our main concern is with the consequences that could give rise to minor radioactivity releases; for unlikely accidents, our main concern is with more serious consequences. For this reason, it is convenient to consider likely and unlikely accidents separately, because the different concerns merit different types of treatment in the two cases.

B. Likely Accidents

Likely accidents can be conveniently divided into accidents due to an event whose initial effects are localized to a small part of the core, and those due to an event that starts by affecting the entire core. Our main concern at present is with reactors whose fuel is in the form of pins grouped in subassemblies, but we must remember the possibility of other fuel configurations, such as coated particles, being important in the future.

1. Local Failures

Local failures are relatively hard to detect, but the damage done is in general acceptable (from both safety and economic standpoints) if it can be confined to one subassembly. Thus, the main problems are the detection of incipient accidents, and the prevention of the propagation of damage.

a. Accidents Initiated by Fuel-pin Failure. As there may be 10^5 fuel pins in a single reactor core, failure of the cladding of individual pins (due to manufacturing defects or hot spots) should, for economic reasons, only very rarely require the reactor to be shut down. Thus, the effects of a single pin failure should be small, and propagation of this failure to cause failure of more than a few other pins must be improbable, even if the fuel is near the end of its irradiation life, or has been subject to some previous minor maloperation (such as a minor reactor power or coolant flow transient).

The initial fuel failure may be influenced by such things as manufacturing defects, the mechanics of the fuel and cladding materials, fission-product gas, or hot spots caused by coolant-flow irregularities, foreign matter in the coolant, or local power peaks. Propagation could be caused by:

(1) Fission-product gas release (which could cause gas blanketing and overheating of adjacent pins).

(2) Fuel released from the cladding failure (which might lodge against or stick to adjacent pins causing overheating, or might cause a vapor explosion--see Chapter V--thereby damaging adjacent pins by a pressure pulse).

(3) Distortion of the broken cladding or fuel-pin spacer to touch adjacent pins and cause overheating.

Failure and the possibility of propagation will depend on such things as whether the fuel is vented or not and whether the vent is working properly, and whether the failure occurs while the reactor is at steady power or is caused by a power or coolant-flow transient. These points are discussed in more detail in Chapter IV, but without going into detail we can identify the main areas of interest. We need to know:

(1) How a fuel pin behaves as it fails. (How big a hole is made in the cladding? Where is it likely to be? How quickly is gas released? Is it propelled by gas? Does the pin bend? What are the effects of sodium logging, if it can occur?)

(2) What happens to gas or fuel released from the pin. (Can the gas cause overheating of adjacent pins? Where does the fuel go? Can it cause a vapor explosion? If so, is it violent enough to break other pins?)

(3) How the answers to 1 and 2 depend on burnup, prior maloperation of the fuel, the cause of failure or the pin design.

Instrumentation to detect failure of a single fuel pin is important as a guide to a reactor operator, and also as a means of initiating action to prevent the spread of damage beyond one subassembly.

b. Accidents Initiated by Flow Blockage in a Single Subassembly. If a partial blockage of a subassembly is big enough, it may lead to coolant boiling and voiding, and fuel melting. This sequence may take place so quickly that, with the difficulty of detecting such a blockage, it may be impossible to prevent melting of most of the fuel in the subassembly, even if every subassembly has a flowmeter. This would be acceptable, however, if the probability of such subassembly blockage were low enough, and if the damage due to melting could be confined to the one subassembly. The problem is to be sure that blockage and melting of one subassembly, when the reactor is not shut down immediately by a protective system, cannot damage any of the rest of the core (which might be economically unacceptable), and that such damage cannot initiate a damaging power excursion before it is detected (which would be an unacceptable safety risk). It is also desirable to know how big a partial blockage is needed to cause boiling and melting; and the less likely event of a complete subassembly blockage, which raises similar problems, has to be considered as well.

Propagation beyond one subassembly could be caused by:

(1) Pressure pulses from a vapor explosion. (This is a crucial question. Can explosive boiling of the coolant break the subassembly walls and damage the rest of the core? Can it distort the core to increase its reactivity or prevent the insertion of control and safety rods?)

(2) Fuel-movement reactivity effects. (If the fuel can compact at the center, the size of the subassembly must be limited so that the reactivity increase is less than a dollar. Even so, it could cause a power transient, which could damage the rest of the core. This damage might involve widespread cladding failure, releasing gas that might cause a prompt-critical excursion, even though the initial reactivity transient was far from prompt critical.)

(3) Fuel melting through the subassembly wall. (This seems to take place slowly, so that propagation before the failure is detected is unlikely.)

(4) Coolant reactivity effects. (This would always be prevented by making the subassemblies small enough so that the maximum coolant void reactivity is much less than a dollar, and a significant power excursion is impossible.)

(5) Pressure pulses due to collapsing vapor bubbles. (This seems unlikely because the pulses carry little energy.)

Instrumentation to detect boiling or fuel failure in order to shut the reactor down before propagation beyond one subassembly can take place may be important, particularly if a damaging vapor explosion is possible.

In relation to subassembly blockage accidents, we need to know:

(1) How the coolant moves when boiling occurs. (Once ejected, can it reenter the subassembly and come into contact with hot fuel? Is the reentry accompanied by violent condensation? If cold liquid cannot reenter the subassembly, a vapor explosion is unlikely.)

(2) Whether a vapor explosion is possible. (What conditions are needed to cause it? How much damage can it do?)

(3) How the disintegrated fuel moves. (Is it ejected from the subassembly? Can it compact at the center? Does it freeze on the subassembly walls? How is its motion affected by the presence of boiling coolant?)

(4) Whether analysis predicts the melting of hot fuel through the subassembly walls correctly. (If it can melt through, how quickly does it do so?)

(5) What the effects of a partial blockage are. (How big has a blockage to be to cause boiling or fuel damage? What are the effects of fuel support grids or wrapper wires?)

(6) Whether boiling can be detected quickly by acoustic or other means.

c. Accidents Initiated by Simultaneous Blockage of Several Subassemblies. These accidents might be regarded as unlikely were it not for the Fermi accident. Careful design of the coolant inlets to the subassemblies can reduce the probability of simultaneous blockage so that the fact that the consequences are more severe than those of single subassembly blockage is offset by the lower probability. Analysis of this accident presents no additional type of problem not discussed under single-subassembly blockage in b above.

d. Accidents Initiated by Blockages in a Reactor without Solid Subassembly Walls. Reactors have been proposed in which the fuel subassemblies have either no walls or walls with holes.

advantage that if one subassembly is blocked, coolant can enter it from adjacent ones to help prevent serious overheating. A disadvantage is that a strong barrier to the spread of damage is no longer present.

The problems of this type of design are the effects of blockages between the fuel pins, and fuel-pin failure propagation (as failure could now spread through the whole core). We need to know:

(1) The effects of a blockage on coolant flow and temperature distributions. (How big a blockage is needed to cause boiling? Or fuel-pin failure? How does the boiling coolant behave?)

(2) What happens when fuel pins fail. (How does the fuel move? How far can the damage spread?)

2. Whole-core Failures

Here we are concerned with accidents initiated by, for example, coolant circulating pump failure or control-system malfunction. Since such accidents affect the whole core, they can usually be detected easily, and damage is prevented or minimized by shutting down the reactor.

Coincident failure to shut down is normally unlikely, and as such is dealt with in C below. It could, however, be likely if some event (such as a fire or an earthquake) could cause one of the failures mentioned above and a failure of the shut-down system. Avoidance of such coincidences is normally a design problem and beyond the scope of this study.

Nevertheless, the development of design principles to minimize the possibility of unforeseen coincidences, or the evaluation of the effects of earthquakes, for example, on reactor structures, are important reactor safety questions.

C. Unlikely Accidents

Unlikely accidents are those with a low probability of occurring and so are of concern only when the consequences are serious, involving at least breach of the primary coolant system. If the reactor is shut down successfully and the primary containment stays intact indefinitely, the amount of damage done to the core is not of primary importance.

In practice, this means that our interest is confined to super-prompt-critical excursions, arising, for example, from one of the whole-core accidents mentioned in B.2 above, but now with coincident failure to shut down. In such a case, the problem is to predict:

1. Whether prompt critical is reached, and the rate of change of reactivity at prompt critical.

2. The amount of energy released (as work and heat) and the amount of damage done by the super-prompt-critical excursion or excursions.

3. The behavior of the hot debris after the super-prompt-critical burst, and the possibility of more heat being turned into work to do more damage.

4. The amount of radioactivity (fission products, plutonium, radioactive sodium, or other materials) released from the primary containment.

It is convenient to organize this discussion of unlikely accidents differently from that of likely accidents, considering successive phases of a super-prompt-critical excursion in turn.

1. Before Prompt Critical

It is important to predict the rate of change of reactivity at prompt critical accurately (i.e., to within about 30%), because the magnitude of the subsequent excursion is strongly dependent on it. Prompt critical may be reached by, for example, control-rod movement or the movement of a large gas bubble, but the most difficult cases are when coolant ejection due to boiling, or the motion of fuel as it melts or breaks up, contribute to the reactivity changes.

In the case of an accident initiated by a coolant-flow failure with failure to shut down, the coolant will boil and be ejected from at least part of the core before the fuel collapses. For one of the present 1000-MWe designs, for example, this may cause the reactor to go prompt critical, with the rate of change of reactivity depending on the velocity and void distribution in the transient two-phase coolant flow alone.

If, on the other hand, the sodium void reactivity is less positive, or if coolant boiling causes fuel failure to follow very quickly, or if the accident is initiated by a rapid power transient, the cladding may fail and the fuel melt so that movement of fuel contributes to reaching prompt critical. In such a case, the rate of change of reactivity will depend mainly on the motion of the fuel as it disintegrates in the presence of boiling coolant. This motion may be complicated by vapor explosions if hot fuel and liquid coolant come into contact.

In either case, however, there is a lot of inherent randomness: in the location of hot spots and vapor nucleation sites, in the void distribution in two-phase flow, in the behavior of the cladding as it fails and of the fuel as it cracks and melts, and in the fuel movement. This randomness may always prevent an accurate prediction of the rate of change of reactivity at prompt critical.

We need to know:

a. How to characterize the transient hydrodynamics of coolant ejection from a subassembly by boiling (particularly the velocity and void distribution).

b. How to predict the motion of broken and melting fuel, either with or without the presence of boiling coolant. (This is a crucial question, about which we know very little. The first need is for some qualitative information on the sort of motion which may be expected. This could provide a basic model for quantitative approach.)

c. The range of uncertainty to be expected in reactivity rate calculations based on such a model.

2. The Super-prompt-critical Excursion

After the reactor goes prompt critical, energy is generated very rapidly until the reactivity transient is ended by the Doppler effect or by material motion. If the Doppler effect is large, the reactor may pass through prompt critical more than once before the excursion is finally terminated.

We have to predict the amount of energy released in the excursion, which appears either as heat or as kinetic energy of the core materials and their immediate surroundings; and the amount of damage this kinetic energy can do to the reactor containment, both as functions of the rate of change of reactivity at prompt critical. Ideally, we would like to know two important prompt-critical reactivity rates: that which just results in breach of the primary containment, and that which just breaches the secondary. If the prompt-critical reactivity rate for a certain unlikely, but not "very unlikely," accident (see above, p. 15) is higher than the latter, the reactor is unsafe; if it is lower than the former, the reactor is reassuringly safe for that accident.

Calculation of the energy generated in the excursion involves coupled neutronics and hydrodynamics, allowing for feedback from the Doppler effect and the disassembly under the influence of the very high pressures caused by the rapid nuclear heating. Difficulties arise from the lack of equation-of-state data for the core materials, and in the treatment of the hydrodynamics. The latter is particularly difficult in the case of relatively mild excursions when the strength of the core structure is important.

Damage to the containment and structure by the excursion itself is caused by shock waves generated during the high-pressure disassembly of the core, and by the pressure of the expanding vaporized fuel. The effects

of these on the structure (including effects such as sodium hammer, etc.) can be studied experimentally using model tests and chemical explosives, provided we understand the equivalence of chemical and nuclear explosives in this respect. We need to know:

a. Whether our excursion calculations are adequate (i.e., in the treatment of transient neutronics, three-dimensional hydrodynamics, the effects of restraints, etc., both for cores that are intact and cores that are damaged at the start of the excursion.)

b. Equation-of-state and Doppler-coefficient data.

c. How much damage is done to the containment . (Is the vessel broken? Is the plug lifted? Are model experiments using chemical explosives adequate?)

3. After the Excursion

After the super-prompt-critical excursion, the core debris is very hot, and we wish to know whether it can do more damage. For example, it might cause a vapor explosion in the cold coolant that surrounds it, and this might breach the primary containment even if the latter survives the excursion intact, or missiles might be produced that could penetrate the secondary containment. Alternatively, ordinary boiling of the coolant may be enough to pressurize and breach an otherwise intact primary containment.

If the primary containment is breached, either during or after the excursion, we wish to know how much radioactivity escapes. Thus, we need to understand the behavior of fuel or fission products released into the coolant or blanket gas, the behavior of the coolant and gas within the residual core and vessel structure, and especially any dispersal mechanisms such as sodium fires.

Lastly, we wish to know the final disposition of the core material so that appropriate disposal measures can be prepared, and to assure ourselves that it can be contained and cooled indefinitely.

We need to know:

a. How the core debris behaves. (Can it boil the coolant and pressurize the containment? Can it cause a vapor explosion? Can such an explosion produce damaging missiles? Can coolant or blanket gas be forced out of the primary containment?)

b. How much radioactive material can be released from the fuel debris. (How much fuel is suspended or dispersed in the coolant? How many fission products are retained in the fuel, dissolved in the coolant, or released to the blanket gas?)

c. What the effects of a sodium fire are. (Does it damage the secondary containment? Can the radioactivity it releases be handled? How can it be prevented or minimized?)

d. What happens finally to the debris. (Can it be cooled and contained indefinitely? Is a critical reassembly possible?)

D. Summary of the Main Safety Questions

We now take all the areas mentioned above where more information is needed (introduced by the phrase, "we need to know:" at various points in Sections B and C above) and summarize them in a list of all the questions about which more knowledge (to be gained experimentally or theoretically, in or out of pile) will materially improve our understanding of LMFBR safety.

This is not intended to be a detailed account of all the information needed, but a survey of the entire field. The questions mentioned here are discussed at length in Chapters III-IX below.

The questions about which information is needed are listed, in what appears at the present time to be their order of importance.

1. The Most Important Questions

The four questions discussed below are of outstanding importance in that they touch on areas where there is concern that an LMFBR may be unacceptable for safety reasons. If these questions are answered, our major worries about LMFBR safety will be removed.

a. Vapor Explosions. We need to understand the violent-boiling or "vapor-explosion" phenomenon in order to make realistic predictions as to the conditions under which it can occur with hot fuel or steel and sodium. (Is it possible on the scale of a single fuel pin, a single subassembly, or the whole core?) This understanding should also allow us to make realistic, as opposed to very pessimistic, estimates of the efficiency of conversion of heat to work and thus of the damage capability.

This question needs to be answered to remove the concern that a likely accident (single subassembly blockage) may damage the whole core (which would be economically unacceptable) and possibly even cause a super-prompt-critical excursion (which would make the reactor unacceptable from the safety standpoint). It also has an important bearing on the possibility of fuel-pin failure propagation, and on the ability of the containment to withstand a mild super-prompt-critical excursion (if a vapor explosion can be caused by the core debris).

If the question of vapor explosions is answered, it may be possible to eliminate the need (which pessimistic predictions indicate) for certain

engineered safety features, such as very fast acting and sensitive flowmeters on every subassembly, for example. Such features would be very hard (perhaps impossible) to design and expensive to install.

The initial understanding of this phenomenon will come from out-of-pile tests, where qualitative observations can be made easily. In-pile tests in TREAT will probably then provide a quantitative understanding, and data on the efficiency of the process, on the scale of a few fuel pins. Larger-scale tests, on up to an entire subassembly, would require a new facility.

This is first on our list of outstanding problems, because at the time of writing it is the point of greatest weakness in LMFBR safety analysis. The pessimistic assumptions to which designers are forced in the absence of a good understanding can even threaten the viability of some designs. This importance, however, is not permanent in that once an understanding has been reached, vapor explosions will cease to be of great interest, unlike fuel behavior on failure, for example, which will always be a live issue.

b. Fuel-pin Failure and Failure Propagation. We need to understand the behavior of a fuel pin as it fails, and whether the failure can propagate.

This requires an understanding of the "metallurgy" of fuel and cladding including topics such as fuel swelling, cracking, fission-product gas release, sodium logging, and cladding swelling, ductility, and hardening. This must enable us to predict the effect of burnup or previous maloperation on the mode of failure, and the behavior of unvented fuel or vented fuel in which the vent has become blocked. Typically, we need to know how big a hole is made in the cladding, how much fuel and gas are released, whether the pin is bent, and what damage is done to other pins. We must know the mode of pin failure caused either at steady power by manufacturing defects or slowly-developing temperature perturbations, or in power or coolant-flow transient situations. This information must ultimately be provided for carbide as well as oxide fuel.

This question must be answered to eliminate the possibility that a very likely event (pin failure) may cause significant local damage (needing replacement of a subassembly, which would be economically unacceptable), or even damage to the whole core (if failure can spread beyond one subassembly).

In-pile work is required, with support from out-of-pile experiments (on the behavior of gas in a coolant stream, the mechanical aspects of fuel behavior, etc.). Provision of information on many aspects of fuel behavior is covered by Element 7 of the LMFBR Program Plan,

but it is an important and by no means trivial task to ensure that this information is available in a suitable form for safety studies. In-pile work specifically on the safety aspects of fuel-pin failure and failure propagation is currently being carried out in TREAT and GETR.

The question may change in nature, or disappear, if future reactors use other fuel configurations than bundles of pins.

c. Motion of Molten Fuel. We need to know how the fuel moves as it breaks up or melts in the presence of boiling coolant. We want to know whether it can collect at the core center, and if so, how fast, or whether it is inevitably or probably swept out of the core. We are seeking a model of a process of which at present we have no understanding. If we can make such a model, we need to know how reliable it is and whether we can be certain of conclusions based on it. We wish to understand the process for reactors, either with or without solid subassembly walls.

This problem needs to be solved to enable us to predict whether a super-prompt-critical excursion is possible in the case of, for example, a coolant-flow failure with coincident failure to shut down, and if it is, to predict the rate of change of reactivity, which determines the ability of the containment to handle the accident. In addition, if whole-core accidents with coincident trip failure (which we have called "unlikely" here) come to be thought of as likely in the future, this problem becomes more important, because it then determines the possibility of a likely accident leading to a super-prompt-critical excursion.

Although out-of-pile model tests may provide some help, this is mainly an in-pile problem. TREAT may assist, but a large-scale study requires a new facility.

d. Severe Excursions. We need to know whether our super-prompt-critical excursion calculation methods are adequate to predict the amount of energy released, and whether they are overpessimistic. We wish to be certain that our treatment of the transient neutronics and hydrodynamics is good enough, and particularly in the case of mild excursions whether we can calculate the effects of the strength of the reactor structure well enough.

This problem needs to be solved to give a less pessimistic basis for containment design. Coupled with c above, it may be possible to show that some accidents on the border between likely and unlikely can be contained within the primary system.

This is largely a theoretical problem, in which some out-of-pile model tests may be useful. An in-pile check may be required.

2. Other Important Questions

The three questions below are not as important as the first four, because they do not have a strong bearing on the behavior of the core in an accident, or because (in the case of coolant boiling) significant progress toward solution has been made. Nevertheless, if these questions remain unanswered, undesirable distortions may be forced on LMFBR designs.

a. Coolant Boiling. We wish to understand superheating in sodium, the hydrodynamics of ejection of coolant by boiling from a subassembly (other than by a vapor explosion), the two-phase flow that ensues, and the reentry of liquid into a voided subassembly.

The ejection phase is of great importance in predicting prompt criticality in a reactor with a large positive coolant-void coefficient of reactivity. It is largely understood, at least for single-tube geometry.

If reentry while the fuel is still hot proves impossible, the danger of a vapor explosion in a subassembly blockage accident is reduced.

This is out-of-pile work, unless there is difficulty in developing suitable high-flux heaters. An in-pile check may be needed to determine what superheating is possible in a reactor.

b. Ultimate Containment. The final problem of a severe excursion or core meltdown is whether the core debris can be cooled and contained indefinitely, or whether the decay heat will cause it to melt through every containment and allow the release of radioactivity. Current design efforts suggest possible solutions, but a significant program of work is required to ensure their feasibility.

c. Instrumentation. An acoustic or flowmeter boiling detector, or some other way of detecting local abnormalities, would be useful if it could act fast enough to prevent either pin-to-pin or subassembly-to-subassembly propagation of damage. A fast-acting fission-product detector would also be of interest.

Development of such instrumentation has to be done out-of-pile, but in-pile tests are needed.

3. Other Questions

The questions posed above are all in some way critical to the development of LMFBR's. There are other questions that are not critical in this way, but on which a significant amount of work remains to be done. These we summarize briefly here, without attempting to place them in order of importance.

a. Doppler Effect. This has to be understood in order to predict behavior in a super-prompt-critical excursion.

The problem is partly answered; such work as remains is mainly theoretical, possibly requiring in-pile checks.

b. Sodium Void Reactivity. This has to be calculated accurately to predict behavior in loss-of-coolant accidents. An important problem is the complex geometry the coolant void may adopt.

This is a theoretical problem, with in-pile checks in a ZPR.

c. Explosion Damage. Given the amount of energy generated in a super-prompt-critical excursion, we need to be able to predict the amount of damage such an excursion can do to the containment. This requires knowledge of the response of the core and blanket structure, the reactor vessel, the plug, and the biological shield, to the shock waves produced.

This problem needs theoretical analysis and out-of-pile model tests using chemical explosives, provided these can be made equivalent to the nuclear explosion. An in-pile test could provide a check, but is not essential.

d. Postburst Behavior. The behavior of the debris after a super-prompt-critical excursion is not certain. There may be a possibility of a large vapor explosion which could damage the containment, and there may be other problems. The main need is for a model of what might happen.

e. Subassembly Wall Melting. Propagation of failure from one subassembly to another by means of fuel melting through the walls may be possible. It is necessary to check whether this is the case.

This question can probably be answered analytically.

f. Equation of State. We need to know the pressure and temperature of the core materials--especially the fuel--as functions of volume and energy content during super-prompt-critical excursions. These data are not known accurately at high energy contents; better data may improve the accuracy of the explosion calculations. Perhaps, however, the results are not in fact very sensitive to equation of state data.

This is mainly an out-of-pile problem, but some specialized in-pile experiments may be required.

g. Fission-product Behavior. After a super-prompt-critical excursion, fission products and fuel material may be dispersed from the collapsed fuel pins. Since they may then be able to escape from the

containment, we wish to understand their behavior, and especially the extent to which they may be dissolved and trapped in the coolant.

This is an out-of-pile problem.

h. Sodium Fires. If the primary containment is breached, a coolant fire may be possible. This may damage the secondary containment and will disperse radioactive sodium and possibly fuel and fission products. We must understand the behavior of such a fire, occurring as either a spray fire or a pool fire, to predict the stress to which it subjects the secondary containment, and the ability of the filtration system to handle the combustion products.

This again is entirely an out-of-pile problem.

CHAPTER III

SODIUM BOILING

A. Introduction

The boiling process in liquid metals is of major interest in the analysis of abnormal reactor behavior, because of the need to assess rates of sodium voiding and expulsion to derive the proper feedback reactivity. Many of the basic phenomena involved can be studied experimentally out-of-pile using high-heat-flux heaters that are being developed to simulate reactor conditions. The following questions need to be satisfactorily answered to ensure accurate simulation of reactor conditions:

1. How many fuel pins must be used to characterize adequately coolant expulsion from, and reentry into, a single subassembly during an accident?

2. Does a reactor environment significantly affect the nucleation conditions? (That is, what role does radiation or the presence of fission gas play in nucleation?)

Further, since out-of-pile experiments tend to be much cheaper, faster, more flexible, and easier to instrument than in-pile experiments, there is great incentive to do as much as possible of the sodium expulsion and reentry experimentation out-of-pile. However, if out-of-pile experiments indicate that relatively large superheats are required to initiate boiling in experiments that accurately simulate reactor conditions (high heat fluxes, similar coolant purity and fuel-element surfaces, similar flowrates and temperatures) in all aspects except the radiation environment, then in-pile experiments are required to see if the large superheats are also observed in the reactor. In any case, it would appear that some limited in-pile testing would be required to attempt to verify out-of-pile results.

Even in out-of-pile sodium boiling experiments, however, the available instrumentation for transient void measurement is not entirely satisfactory, and therefore the use of fluids other than sodium may provide useful information for model development.

In the following discussion, analytical and experimental efforts to date are reviewed briefly and a suggested work plan broken down into out-of-pile and in-pile experiments is presented.

B. Analytical Problems and Data Needs

At present, all the sodium-expulsion data have been obtained in single-channel experiments, which indicate that boiling of sodium is

followed by rapid voiding of the channel, and that the rate of expulsion depends strongly on the amount of superheat attained by the liquid before flashing occurs.[2] Most of these experiments have simulated complete flow blockage or stagnant sodium.

1. Superheat

The large superheat required to initiate boiling in liquid metals, as evidenced by recent experiments,[3-8] has caused considerable concern in fast-reactor safety analysis. This is because large superheat represents a significant amount of energy stored in the coolant, which could be released extremely rapidly at the onset of boiling, thus producing large rates of reactivity change from voiding and large pressure pulses. The reported superheats have ranged from a few degrees to approximately 800°C, and the spread in these data is indicative of the many variables that can affect the superheat behavior of the alkali metals, e.g., pressure-temperature history, heat flux, heating surface, fluid velocity, pressure, aging, purity, dissolved gases, heating method, radiation, and perhaps many more.

The pressure-temperature history has been shown to have an effect on superheat[3,8] and may explain part of the variation found in published data where this variable is not accounted for. Surface condition is used to explain the spread in the data reported in Ref. 4, but more recent results[9] indicate that surface roughness is of little or no importance, in agreement with the equivalent cavity model,[10] which describes the effect of the pressure-temperature history. The results of Ref. 10 are based on the supposition that a wide spectrum of surface cavities exists. A smooth surface, or one with a limited range of cavity sizes, such as might be produced by corrosive attack in a reactor environment, might provide an overriding effect in some operating conditions. Decreasing total pressure generally increases superheat.[3,4,11,12] Data have been reported to indicate that the superheat may be unaffected,[4,5] increased,[3] or decreased[13] by an increase in the heat flux.

Differences in fluid purity may be the cause of much of the data variation,[5] and dissolved oxide and dissolved gases may influence the superheat. These are perhaps the least well-defined quantities in present experiments. The type of heating method is also indicated as a variable, since experiments using direct-heating techniques[4,5] give generally higher superheats than those using indirect methods.[3,6] Hence the results of experiments using indirect-heating methods must be used with care. Some investigations have reported superheats to be lower under forced circulation than under stagnant conditions.[14,15] The detailed discussion of the effect of inert gas on superheat presented in Ref. 16 indicates that the effects of many variables (heat flux, heating method, fluid velocity, etc.)

appear to be related to gas-transport phenomena. According to a recent theoretical study,[17] nuclear radiation may affect the incipient superheat, but only at relatively high superheats (>200°C). So far, experimental data have shown no effect of reactor environment on superheats for thermal-neutron fluxes ranging up to approximately 2×10^{11} neutrons/cm^2-sec.[9]

The foregoing discussion indicates that the superheat phenomenon is extremely complex and depends on a number of variables. The importance of these variables has generally been recognized in boiling research with other fluids, but much of our understanding of boiling is still empirical. Because existing data have been obtained in systems in which the variables that influence superheat were not carefully controlled, the data are of limited use for reactor-safety design criteria at the present time. It is very likely, however, that if gas bubbles are present in the regions of the core where boiling is likely to start, essentially no superheating will result;[18] but if the bubbles are not present, an analysis of nucleation from surface cavities assumes importance in predicting the liquid superheat. Hence the conditions under which gas bubbles will be carried through the core with coolant must be determined. This required an understanding of the flow of coolant through the reactor system, and especially of the conditions under which the cover gas can be entrained.

In addition to basic studies of the liquid-superheat phenomena, simulation tests under typical reactor conditions are needed to be sure that in-core superheats can be predicted accurately. Particular emphasis should be on the following:

a. Heating surfaces that simulate cladding surfaces after long periods of reactor operation.

b. Heat fluxes of 1-2 kW/cm^2 from indirect heating techniques (i.e., heat not generated in the coolant).

c. Coolant of defined (and monitored) composition, including the oxide and gas content, the amount of entrained bubbles and gases leaving solution during typical pressure and temperature variations.

d. Geometry and velocity that closely resemble reactor conditions.

e. The effect of ionizing radiation, as discussed above.

2. Void Growth and Sodium Expulsion

The problem of theoretically determining the growth rate and associated pressure rise in an isolated vapor bubble in a superheated fluid of infinite extent has been extensively studied, especially for

nonmetallic fluids (e.g., Refs. 18-22). A limited amount of work has also been done on the equivalent problem for metallic fluids.[23] However, the simplified physical models used in these analyses may be inapplicable to reactor geometries, except for extremely short times after the start of growth. The interaction of surrounding fuel elements with the growing bubble can be expected to cause a deviation from the predictions of an infinite fluid model of the void growth and pressure transients. Therefore, it is necessary, at this time, to rely upon experimental data.

Some suitable data exist and are in general agreement with the prediction that the vapor pressure pulses are directly related to the amount of superheat. Some dependence on geometry is expected, but it may be weak. For example, the pressure pulse resulting from the growth of a vapor bubble in a large single tube containing sodium superheated by 105°C was measured as less than 0.6 atm.[7] Similar experiments using a small tube reported pressures of 0.7 atm.[24]

If extremely large superheats were attained in a reactor, the initial pressure pulse could be large; for example, for sodium initially superheated by 500°C, pressure peaks of 7-13 atm were measured.[24]

Two basically different approaches have been adopted in theoretical studies of transient void formation rates in reactor coolant channels: (1) vapor growth according to thermodynamic equilibrium,[25-28] with zero liquid superheat before as well as during the void formation; and (2) vapor growth based on thermal nonequilibrium[29,30] with liquid superheating. The various two-phase flow patterns assumed in these analyses range from homogeneous flow to a pattern consisting of liquid slugs separated by vapor slugs--either with or without liquid film present on the heated surfaces. The TRANSFUGUE[25] and MacFarlane[27] models are designed for analyzing boiling in either power-transient or flow-coastdown accidents, although experimental verification of their predictions is lacking. These two models do not have any provision for treating liquid superheat. For the analyses of blocked-channel accidents, the BLOW,[30] VOID,[29] and BURP[26] codes have been proposed. The BLOW code is perhaps the most attractive, because of its simplicity and because it has been experimentally verified to some extent in single-channel tests. Several experiments[4,31,32] have verified that void growth occurs as a single bubble, which expands and eventually occupies the entire cross section, resulting in a piston-type expulsion. This void-growth mode is consistent with the BURP and BLOW models. However, the experiments have indicated that a thin liquid film remains on the heated wall, which gives rise to an increased expulsion velocity. This is consistent with the BLOW model, but the liquid film is not present in the BURP model. The most recent expulsion model developed is that by Cronenberg, Fauske, and Bankoff.[33] This is similar to BLOW in that it assumes the formation of a single bubble

across the entire channel, with the exception of a thin wall film, but differs from it in that it considers not only vaporization from the wall film, but also from the upper and lower interfaces.

The basic requirement for making improvements to existing voiding models is to obtain suitable experimental information about the flow regimes throughout the voiding period, so that the effects of the critical heat-flux limit (when vapor comes into contact with the heated surface), nonequilibrium, and compressibility[34,35] can be evaluated. All these effects will tend to decrease the voiding rate, assuming the reactor coolant channels remain intact. The importance of considering the two-phase compressibility effects was pointed out in a paper by Le Gonidec et al.,[4] where sodium-ejection experiments verified the single-bubble expulsion theory (with a liquid film on the heated surface). It was also shown, however, that condensation occurs as the bubble or vapor slug moves into colder regions resulting in two-phase choking at the top of the void controlling the downward motion of the lower liquid-vapor interface.

Reference 4 also discussed water experiments carried out to simulate sodium expulsion and concluded that there are basic similarities between water and sodium vaporization if similar initial conditions are chosen. Thus, it appears that a significant portion of the detailed information on flow regimes needed to refine existing single-channel voiding models (with the exception of the incipient superheat phenomenon) can be obtained by using nonmetallic fluids which are considerably easier and less expensive to work with. The usefulness of this approach is indicated by the similar observed behavior during rapid boiling of nonmetallic fluids with low subcooling prior to heating[36] and experiments with sodium.[5] Recent experiments on flashing flow of sodium and water[37] also give similar results and, indeed, it has been shown that the same theory developed for predicting maximum discharge rates of nonmetallic fluids can be used for metallic systems.

At present, all the sodium-boiling data have been obtained in single-channel experiments, and this has provided a basis for developing one-dimensional expulsion models. The determination of whether these models can adequately describe the coolant motion in a blocked-fuel subassembly depends upon results of experiments with multipin clusters. At the present level of high-temperature heater-technology development, a seven-pin assembly can be built with an adequate power density, and a 19-pin assembly appears to be possible. Larger multipin assemblies probably can be built, but the cost of the power supply may become prohibitive. Thus, if a seven-pin or 19-pin out-of-pile mockup appears to simulate adequately the behavior of a complete reactor subassembly (several hundred pins), experiments can be performed more profitably out-of-pile. However, if it is determined, based on analysis or out-of-pile experiments with fluids other than sodium, that larger multipin

systems are required for satisfactory simulation (because of wall effects or radial power-gradient effects), it is an open question as to whether these experiments should be conducted in-pile or out-of-pile. In this connection, it would probably be worthwhile to perform a reasonably detailed technical and economic study to obtain a definitive answer to this question. Also, tests using nonmetallic fluids in multipin systems[38] are of considerable value in the understanding of two-dimensional effects in void-growth patterns. Such tests will provide an answer to the question of determining the necessary number of fuel pins to characterize a fuel subassembly accurately.

3. Void Collapse and Sodium Reentry

As stated, the pressures involved in voiding a channel (disregarding fuel meltdown and dispersion) are probably not high enough to damage the core structure, unless in-core superheats are very large. On the other hand, higher pressures may arise when vapor bubbles condense in the subcooled liquid, as may occur during the expulsion of vapor from a coolant channel into colder regions of liquid. Such pressure pulses have long been known to occur for nonmetallic fluids and have been reported for metallic fluids.[5] Reference 5 reported measured recondensation pressure peaks as high as 50 atm, but made no attempt to predict the results by analysis. A preliminary analysis of the collapse of a sodium-vapor bubble has been recently made available,[39] but its results are quite limited. It was pointed out, however, that although a considerable amount of acoustic energy can result from vapor collapse, it is unlikely that much mechanical damage can occur. This is in agreement with the limited amount of experimental information now available. It is clear from Refs. 18, 24, 40, and 41 that large collapse pressures can be reached (up to about 25 atm), but the duration of these pressures is in the range of 0.1 to 1 msec. As suggested in Ref. 39, the initial subcooling appears to correlate with the collapse pressure pulse data,[40] whereas the initial superheat does not.[24,41]

An attempt was made to calculate bubble-collapse pressure peaks using a model based on the well-known Rayleigh equation.[30] This calculation predicts peak pressures up to 50 atm in a saturated sodium pool at 500°C while the bubble radius oscillates between 1 and 0.15 mm with a period of 300 μsec. In more recent studies,[42,43] the effect of the temperature discontinuity at the bubble surface was introduced and shown to be important in evaluating collapse rate and pressure generation. Although the available information indicates that the pressure pulses caused by sudden vapor growth or collapse are unlikely to cause mechanical damage at the expected reactor conditions, the magnitude of these pulses increases markedly with increased superheat or subcooling, and further study, both experimental and theoretical, is required to clarify any possible deleterious results. Furthermore, any analysis of mechanical damage

resulting from these pressure pulses must involve the coupled problem of pressure rise and structure deformation. This problem has seen only a very limited amount of attention[44] and deserves much more if limiting conditions on the types and magnitudes of "tolerable" pressure pulses are to be determined.

A related problem, also of much importance in determining possible mechanical damage caused by pressure pulses, is the study of the propagation of pressure disturbances in liquid-vapor mixtures. To determine the time sequence of events in an accident, we must have information on the propagation velocity and attenuation of the pulses in a reactor core. If the velocity is very low or the attenuation is very strong, the pressure disturbance would tend to be quite localized. This problem has received some study,[35,45] and it was shown that the phenomenon is strongly dependent upon the flow regime.

Little or no attention has been given to calculational methods concerning the reentry of liquid sodium after the coolant channel is voided. Such calculations are, however, in progress.[46] If rapid reentry of liquid sodium occurs, it may give rise to large pressure buildup (sodium hammer effect) and result in damage. Here the actual geometry and reactor conditions appear to be of great importance in determining the strength of shocks and the probability of propagation of damage over the core. Also, the amount of molten fuel present when reentry occurs has a sizable influence on the severity of a possible vapor explosion (molten fuel-coolant interaction). In view of the possible damage and effects on latter events, the reentry problem deserves considerable attention in the future for work on fast-reactor safety--both experimental and analytical.

In addition to the studies outlined above, which are aimed at obtaining a basic understanding of sodium nucleation and the various two-phase flow phenomena that may occur, a series of reactor-simulation experiments is required as outlined in Sections B and C below.

C. Out-of-pile Program

The out-of-pile experiments are aimed at providing data that apply to reactor situations with the fuel and coolant channels intact. The sequence in which these experiments are discussed is not meant to imply priority.

1. Power Transients

The importance of power transients at full flow with intact geometry depends upon whether the claddings fail before or after coolant boiling. These experiments are designed to simulate situations resulting from modest reactivity insertions where coolant boiling and expulsion occurs before the cladding or fuel fails.

To perform these out-of-pile simulations, we must develop heaters that are capable of giving peak transient fluxes in the range of 1-2 kW/cm^2. In general, it is desirable to be able to protect the heaters from burnout at high vapor-void fractions, since it is a great advantage to be able to use a set of heaters for as many experiments as possible. However, since we wish to obtain some indication of the threshold of fuel-cladding failure due to vapor blanketing during sodium expulsion (even though this is not specifically required to calculate reactivity feedback due to sodium void), some heaters will undoubtedly be run in experiments that lead to burnout. Such information is required specifically in modeling when one desires to change from a model describing the intact coolant channel to one describing the meltdown and fuel-coolant interaction after the fuel-cladding failure threshold has been exceeded. This does not mean that these out-of-pile coolant-boiling simulations are designed to provide all the information on fuel-cladding failure thresholds. Threshold determination is a complex problem dependent on a variety of parameters, and therefore in-pile simulation is necessary.

Some comments on the number of pins that can be heated electrically in simulation experiments, from a practical standpoint, have already been made in Section B.2 of this chapter. However, the requirement as to the number of fuel pins needed to represent accurately subassembly or whole-core behavior must at present still be considered an open question. Results of the out-of-pile experiments may give unexpected results and indicate that very much larger pin clusters are required to represent certain phenomena than is now believed to be the case. If such results should occur, the possibility still exists of making coolant-voiding measurements in-pile as part of the program of experiments for studying fuel-failure mechanisms. That is, the coolant-dynamics measurements could be made during the initial phases of a complete subassembly failure-propagation experiment. But at the same time, one must remember that such experiments will be much more difficult to interpret and analyze because of the handicap imposed by making the measurements in-pile. At present, it is anticipated that the geometry (within the limitation of 19 pin clusters), flow rates, pressure drops, liquid legs above and below the core section etc., can be accurately mocked up in the out-of-pile experiments. An early check on the validity of this assumption is needed.

Although there are certainly problems with electric-heating simulation, the disadvantages are greatly outweighed by the experimental advantage of performing sodium boiling experiments out-of-pile rather than in-pile.

The important parameters to be investigated in the out-of-pile experiments are incipient superheat, point of nucleation, voiding rate, reentry rate, and pressure generation. These require measurements of

temperature, pressure, flowrate, and void-fraction distribution and growth rate. In the case of the void fraction, it is desirable, because of the sensitivity of reactivity effects to the location of void, to have a fairly detailed description spatially of the void distribution during the period of void growth. However, it is not completely known at this time how accurate spatial-distribution measurements can be within the confines of available time and money to design and build experiments. Although most experimental techniques for making the measurements are relatively well developed, a considerable amount of work remains to be done on developing a technique for measuring the transient vapor void fractions.

2. Flow-coastdown Transients

The same general considerations discussed above for power transients apply to the flow-coastdown transients. The major difference is that the heaters supply lower heat fluxes, comparable to those during steady-state reactor operation at full power. A means will be required to decrease the flow in the test section in accordance with that which would be found in the case of a pump failure in a large LMFBR cooled reactor system when the pump fails. Typically in such a case the time for boiling to start after the initiation of the coastdown is in the range of 5-10 sec. In general, the quantities to be measured in these experiments are the same as those outlined for the power-transient simulation experiments. Therefore, the experimental apparatus developed to run the power-transient simulations will be completely acceptable for the flow-coastdown transients.

3. Local Flow Blockages

Basically, the same type of heated subassembly and flow loop can be used for these experiments as for the flow-coastdown and power-transient simulations; however, in addition we must have the following additional features:

a. A method of introducing a local blockage of the flow passage between pins, once steady operating conditions are established as a means of simulating a foreign body becoming lodged in a subassembly and restricting flow in a small area.

b. A method of introducing a blockage or partial blockage in the inlet of a subassembly, such as might be incurred when a large foreign object lodges in the inlet orifice or plenum. This case is relatively easy to simulate, compared to the previous situation.

c. A method of studying the effects of local design variations in pitch-to-diameter ratio, flow-path length, spacer type, etc. This requires a facility with an adaptable test section.

As for power transients and flow-coastdown simulation, the parameters to be measured are temperature, pressure, flowrates, and void-fraction growth rate, and, here again, the measuring equipment and test loop would be identical to that required for the previously described simulation experiments. Thus, the same basic type of experimental equipment used for the simulations of power and flow-coastdown transients can be used for local-flow-blockage experiments.

4. Electric Heating

The foregoing discussion indicates that a major item in the out-of-pile experimental program is the development of suitable heaters for simulating relatively large clusters of fuel pins in a simulated subassembly, and, indeed, the success of the out-of-pile program proposed depends strongly on such heaters and their associated power supplies and hardware. Considerable effort is currently being expended in developing electron-bombardment (EB) heaters and resistance heaters for out-of-pile simulation experiments. Currently, the EB heaters[47] appear to be the best means of achieving the high heat fluxes necessary to simulate power transients. The major difficulties with electric-heating simulation using EB heaters arise in the areas of: (1) accurate mockup of the axial heat flux profile and (2) the dissimilar thermal behavior of a heater pin and an actual reactor fuel pin, due to differences in heat capacity between the two. Although resistance heaters may be limited to lower heat fluxes (perhaps ~0.5 kW/cm^2), they have the advantage that they more nearly simulate (in terms of heat capacity) an actual fuel rod. The heat fluxes that appear to be achievable with the resistance heaters would be satisfactory for the constant-power, flow-coastdown transients. All indications are that the required heater development is within the realm of possibility, and therefore the question of the being able to perform the out-of-pile simulation studies with boiling sodium does not appear to be seriously in doubt. In any event, it appears that the uncertainty here is less than that associated with being able to develop suitable in-pile instrumentation for performing the experiments in-pile.

D. In-pile Program

A major fraction of the in-pile experiments in which the sodium boiling and voiding phenomenon would be studied are those experiments with a core segment consisting of a whole subassembly or more, in which fuel-element failure thresholds have been exceeded and the coolant channels are no longer intact. In this class of experiments, we are no longer studying boiling and expulsion as a separate phenomenon, since boiling is now produced by the interaction of molten fuel with coolant. Therefore, these in-pile experiments would not be oriented specifically toward the measurement of coolant motion due to boiling and expulsion, but rather to the motion of fuel material once the failure threshold has been exceeded.

Naturally, the instrumentation of such experiments will include measurements of temperatures, pressures, and coolant flowrates (probably at the upper and lower end of the fuel subassembly), in addition to the fuel motion measured with a fast-neutron hodoscope. Such data will make it possible to determine something about the gross motion of coolant, even though specific details about the spatial distribution of coolant vapor voids will be lacking. Therefore, it is possible that some data on coolant expulsion during the in-pile meltdown experiments, in the form of auxiliary measurements, can be obtained. This type of experiment is discussed in detail in Chapters VI and VII.

In addition, however, it is inevitable that a limited number of in-pile experiments designed just to study boiling with coolant channels intact will be necessary. The purpose of such experiments will be to attempt to verify the coolant boiling models which have been derived from the out-of-pile work.

CHAPTER IV

FUEL-ELEMENT FAILURE AND FAILURE PROPAGATION

A. Introduction

In Chapter II we pointed out that the most important single safety question about fuel pin failure is that of propagation. Under what conditions can the failure of one pin precipitate the failure of others and set up a chain reaction affecting so many of the pins in a subassembly that it has to be replaced? How long can operation with failed fuel proceed before the risk of propagation becomes important? These questions are important because single pins will probably fail frequently, but the need to replace a whole subassembly as frequently would be unacceptable. More important, the possibility of damage spreading beyond the one subassembly raises a significant safety risk.

However, fuel failure has a wider importance than this. Cladding failure is the main limit on the irradiation life of the fuel. There is inevitably economic pressure to extend this limit and improve the fuel so that it can be irradiated longer. Our ability to do this depends on our understanding of the mechanisms that cause fuel failure. Thus, we have to ask both the general questions of how the fuel behaves and what causes it to fail, as well as the specific one of whether such a failure can propagate dangerously.

These questions illustrate the fact that a general understanding of fuel failure will always be important in a system where fuel failure is the main performance limit. This will involve knowledge of the behavior of the fuel and cladding materials under irradiation, of the interaction between them, and, in particular, of the mechanisms that stress the cladding and cause it to fail, both during normal reactor operation, and during abnormal and transient conditions. We must also know the influence of design variables, such as fuel density, cladding thickness, or whether the fuel pin is vented or not, and of operating conditions, such as temperature and power generation rate.

To answer the propagation question, we have to know first whether there is more than a low probability that the failure of one pin in "normal" operation can precipitate the failure of "many" others. Here, "normal" operation includes steady-power operation, routine operating transients such as power changes, and minor incidents such as accidental reactor trips or periods of mild overpower. Failure of "many" fuel pins means so many that the subassembly has to be replaced immediately because the amount of fuel or fission products released into the coolant is unacceptable, or because propagation of damage to other subassemblies is likely.

Secondly, we must also answer the propagation question for fuel that has suffered prior maloperation. For example, fuel that for a period has been operated above the design temperature or power level may then be in a sensitive condition, so that, although the maloperation itself does not cause failure, a subsequent failure caused some other way can propagate. This would set a limit on the degree of maloperation that could be tolerated before the fuel had to be replaced, or it might indicate that fuel misused in this way could afterwards be safely operated only at a reduced power level.

B. Ways of Answering the Questions

There are two ways of answering questions about the behavior of a complex system like a fuel pin. The analytic approach involves understanding each mechanism at work, and then constructing an account of the behavior of the system. The empirical approach involves setting up a physical model of the system, and subjecting it to the conditions of interest to see what happens.

For a fuel pin, however, neither approach provides all the answers. The analytic method breaks down, because the system is too complicated to analyze without experimental guidance; the empirical method is misleading, because it is impossible to observe the experiment closely enough and because the conditions of interest cannot be reproduced accurately; therefore an analytic understanding is needed to interpret the observations.

Thus an understanding of fuel-pin behavior and failure requires both basic studies of the individual mechanisms at work, and empirical data derived from experiments that simulate the situation of interest. In addition, reactor operating experience can provide some information.

The steady-state aspects of fuel behavior (1, a and b below) are covered by Element 7 of the LMFBR Program Plan. These steady-state data should be interpreted to yield as much information as possible relevant to safety questions.

1. Basic Studies

We wish to understand the important aspects of fuel-pin behavior as functions of power, temperature, and burnup. These include:

a. The Fuel. How does it deform due to thermal expansion, cracking, irradiation swelling, and creep under the influence of temperature changes and burnup? How fast is fission-product gas released in normal operation or in transients? What is the extent and nature of sodium logging if sodium should get into the pin? If the fuel melts while the cladding remains intact, can the fuel be redistributed within the pin, and if so, how fast, and to what extent?

b. The Cladding. How does it swell and creep under the influence of irradiation and the stress due to swelling fuel and gas pressure? What are the effects of stress concentrations due to hot spots and cracks in the fuel? What are the time and position of failure, and what kind and size of hole is formed in either "natural" failure or one caused by an accident?

c. Pin Behavior after Failure. How much gas and fuel are released from the pin, and how quickly are they released? Is the fuel solid or liquid? Is it propelled by high-pressure gas? Is the pin bent?

d. Behavior of the Gas Released. Does the released gas cause overheating or damaging mechanical stress of adjacent pins?

e. Behavior of the Fuel Released. Where does the released fuel go? Can it stick to or lodge against adjacent pins and cause overheating, or is it ejected from the subassembly? Can it cause a vapor explosion?

Even if all these data were available, however, the interactions between the various effects are so complex that it would be impossible to put them together to make a convincing prediction of pin behavior on failure without experimental support.

2. Empirical Studies

It is probably necessary to do experiments in which one pin of a group is made to fail, to see whether there are conditions under which this failure causes other pins to fail as well. The object of such a test would be to simulate LMFBR conditions as closely as possible. This would require the experiment to be done in-pile, in flowing sodium. It would also require that the state of the fuel and the cladding be as closely as possible the same as in an operating LMFBR. In particular, attention would be given to the state of:

a. Fuel. The extent of cracking, central void formation and swelling, and the quantities of dissolved and released fission-product gas must be correct.

b. Cladding. The cladding must have the correct hardness (induced by irradiation, strain or time). Also, if an artificial defect is used to initiate the failure, its effect must be the same as that of the "natural" failure we wish to study.

If all these conditions are to be met, we must have a clear idea of (1) the state of a fuel pin in an operating reactor, and (2) what influences its behavior when it fails. These demand a good analytic understanding before a worthwhile experiment can be done.

3. Experience

Some useful information about fuel-pin failure and failure propagation may be gained from operating experience of oxide-fueled fast reactors such as BR-5, RAPSODIE, and the experimental oxide subassemblies in DFR and EBR-II. Conditions in these reactors are not the same as in a typical LMFBR, but it is of interest to see what can be learned. Prototype reactors now under construction, such as BN350, PFR, and PHENIX, will also provide some data.

C. Ways of Assembling the Information

We now discuss what has to be done to enable us to answer the safety questions in the ways indicated above.

1. Basic Studies

Much information on the individual mechanisms affecting fuel-pin behavior exists already. In many cases, the need for new safety experiments cannot be defined until existing experimental results have been understood in the context of accidents. This can be illustrated by looking in turn at each of the five areas listed in Section B.1 above.

a. The Fuel. Much information on the behavior of oxide fuel (cracking, swelling, creep, fission-product gas release, etc.) under normal operating conditions exists as a result of test irradiations and has been analyzed to give some understanding of the response to steady operation. More information of this type should come from the operation of prototype reactors. Tests in TREAT and other facilities give some information on behavior in transients, mainly from postmortem examinations.[472-53]

Further tests, on fission-product gas release in long transients (typical of flow-failure accidents), on the effects of sodium logging, on highly irradiated fuel, and possibly on slumping of fuel within the pin, are clearly needed, but it is not possible at this stage to specify these in detail. The first need is to digest existing data thoroughly, so that new safety experiments, in TREAT, for example, can be specified properly.

b. The Cladding. The situation here is similar to that for fuel. The metallurgy of cladding materials at normal temperatures is probably understood well enough (see, for example, Refs. 54-56), and even if it is not, the most urgent need is for analysis of existing data, so that further experiments can be specified. There may be a need for experiments at higher temperatures, which are only reached in accidents, and where the behavior of the cladding may be quite different from that at normal temperatures. High fluence tests are needed to elucidate the problem of cladding swelling.

c. Pin Behavior after Failure. Information on the behavior of fuel and cladding after the cladding has ruptured should come from a and b above. TREAT tests, or tests in a loop in FFTF for some other test reactor, would probably provide data on the mode of failure and pin distortion.

d. Behavior of the Gas Released. This can probably be studied best by out-of-pile tests.

e. Behavior of the Fuel Released. The most important area here is that of vapor explosions, which is discussed in detail in Chapter V. We are also concerned with the motion of fuel released from a broken pin, and whether it can melt adjacent pins. It might be possible to get some information out-of-pile, but in-pile tests seem most likely to be useful. TREAT test, especially of the "flat-topped" type, will provide useful data, from hodoscope or postmortem observations. If wall effects turn out to be significant, however, and tests on a larger scale are needed, these will require the use of a large facility. This is related to the general problem of the motion of fuel in the presence of coolant, which is discussed in Chapter VI.

The most pressing need in obtaining a good analytic understanding of the behavior of a fuel pin as it fails is to digest and codify existing experimental data on fuel and cladding materials: i.e., data from TREAT, etc.[47a-53] When this is done, it will be possible to define in detail future experiments in TREAT or other existing facilities, and to establish the extent to which data from new facilities are needed. The other main analytic need is for an understanding of vapor explosions.

2. Empirical Studies

Out-of-pile tests on the behavior of gas and vapor in the coolant stream between clusters containing various numbers of pins are currently planned. In addition to yielding data needed for hydrodynamic analysis, these tests should play an important role in establishing the adequacy of the small-scale in-pile tests, which are likely to be the main source of information.

Failure-propagation experiments in flowing sodium could possibly be done in the GETR natural-convection capsule (at present being used under the PA-10 program), but the only existing facilities where such experiments could be done in forced convection are Mark-I or Mark-II loops in TREAT. Experiments have been done using clusters of seven EBR-II pins (metal fuel) in a Mark-I loop,[57] but in this case all seven pins were subjected to roughly the same conditions, so the tests study mode of failure rather than failure propagation. In addition, the TREAT power burst in these experiments was about half a second long, and so is hardly representative of the failure at steady power we are mainly concerned with.

TREAT could be used for a propagation experiment with oxide fuel if it were operated in the "flat-topped" mode to give a prolonged power pulse of such an intensity as to make one pin (with some sort of artificial defect) fail while the others experience conditions that would not normally cause failure. One could then observe under what conditions failure of the first pin would cause the otherwise sound pins to fail.

The main disadvantages of such a test would come from the limitation of the length of a TREAT pulse to about 10 sec. Thus, the fuel would probably not be generating heat long enough for the restructuring (by cracking and creep) typical of high-temperature operation to take place. Also, any propagation mechanism other than a very rapid one would not be observed. In addition, such an experiment can probably accommodate only seven pins, so that if wall effects (such as the freezing of fuel on the loop walls) are important, the results could be misleading. There may also be problems with flux depression in the central pin, or with the use of pre-irradiated fuel.

An alternative is to use a flowing-sodium loop in a test reactor such as ETR. The main advantage of such tests would be in the long time scale which would allow slow fuel restructuring and slow propagation of failure, if it is possible, to take place.

In addition, such a loop might accommodate 19 pins or more, provided such an experiment can be proved safe, showing an important improvement over TREAT's seven pins. Even a 19-pin cluster, however, may not be representative of a subassembly containing 200 or 300 pins, so that such tests may tell us whether propagation occurs, and hence establish operational limits beyond which propagation becomes a risk; but they may not indicate directly the extent of the propagation if it does occur, because this may depend on gas or coolant movement, which may be affected by the size of the cluster.

A disadvantage of such tests would be the difficulty of observing what happens in detail. In particular, there may be no way of telling how the fuel moves. This in turn would make more difficult the problem of extrapolating from 19 to 300 pins.

Another problem is that of specifying the tests correctly. For example, the fuel would probably be operated at steady power for a period to restructure it to represent fuel in an operating LMFBR. One pin would then be made to fail artificially to represent a "natural" pin failure. But to determine how long it is necessary to operate to get the correct restructuring, we have to understand the behavior of the fuel material. And to be sure that our artificial failure represents the "natural" failure correctly, we have to understand the behavior of the cladding material, and to know what the "natural" failure we wish to study is. All of these require a good

basic understanding of fuel and cladding behavior, which does not exist at present and is essential before any such tests can be specified properly, let alone performed. It must be emphasized that the analytic and empirical approaches to the propagation problem are interdependent, and cannot proceed separately.

Given this analytic support, it seems probable that a series of tests of this type can solve the fuel-failure propagation problem, provided the limitation to a small number of pins is shown to be unimportant.

D. Summary--Outline Plan of Work

The interaction between analytic and empirical work described above can be summarized in the form of a suggested program of experimental and theoretical work as follows:

1. Existing information on the behavior of oxide fuel and cladding under irradiation (both steady state and transient) should be analyzed to construct as far as possible a coherent account of the behavior of an LMFBR fuel pin in an accident. The principal sources of this information will be TREAT tests and experimental fast-reactor fuel irradiations, but thermal-reactor data, from SPERT, for example, may also be useful.

2. Similarly, the existing data on vapor explosions should be analyzed to make the best prediction possible in our present state of knowledge of the maximum violence of such an explosion in-pile. This is discussed in more detail in Chapter V.

3. These results (fuel-behavior and vapor-explosion predictions) should then be used to formulate in-pile tests on failure propagation, provided a suitable facility is available.

4. The account of fuel behavior produced by 1 above should make it clear where our analytic understanding is most lacking. Additional tests in TREAT, a steady-state irradiation facility such as EBR-II, or a new facility, should be initiated to improve this understanding. Also, slumping tests, using TREAT or possibly PBF, may be indicated.

5. Out-of-pile tests on the behavior of gas and vapor in the coolant stream between clusters containing various numbers of pins, which are planned at present, should be completed and analyzed.

6. From the results of in-pile propagation tests on clusters of up to 19 pins, and the out-of-pile tests described in 5 above, it should be possible to deduce whether the results can be extrapolated to full-size subassemblies, or whether larger-scale tests are needed.

7. A similar program to the above but for carbide fuel should be initiated. The main need at present is for basic irradiation data (steady state and transient) to be used to obtain an analytic understanding of the material. When this is available, propagation tests, etc., will suggest themselves if they are needed.

8. Metal-clad ceramic pins may not always be the preferred fuel configuration for a fast reactor, and, for example, metal fuel or even coated particles may become popular. In such an event, the failure-propagation work proposed above might be irrelevant. But whatever form the fuel takes, propagation of a localized failure to cause damage to a significant amount of the fuel will be serious, and the propagation problem, in some form or other, will require attention.

CHAPTER V

VAPOR EXPLOSIONS

A. Introduction

If a hot material comes rapidly into contact with a cold liquid, there are circumstances in which boiling and expansion are so violent that the effects are similar to those of a chemical explosion. This is what we call a vapor explosion. Although it is not a precise definition, it is adequate, because in practice we are interested in any conditions under which boiling is so violent that it can be damaging.

The importance of this phenomenon for LMFBR safety has been documented by Hicks and Menzies.[58] Although a vapor explosion has never been observed in an LMFBR, the concern is not based on theoretical considerations alone, but on evidence from accidents of various kinds, all of which were characterized by explosive boiling.

This chapter reviews this evidence together with the results of experiments done, mainly at ANL, in an attempt to understand the phenomenon. We then discuss the importance of vapor explosions in LMFBR's, and the experimental and theoretical work still to be done.

B. Explosions from Molten Materials and Water ("Steam Explosions")

1. Evidence from Accidents

Various reports have been made of the explosions that occur when molten materials come into contact with water. Loosely known as "steam explosions," these have been attributed to the sudden conversion of water to steam--a mechanical process, rather than a chemical reaction.

Serious steam explosions are not uncommon in metal industries. Lipsett[59] reported a foundry incident that resulted in the death of one man, injuries to many others, and building damage to the amount of $150,000. A severe explosion appears to have been caused by the sudden generation of steam when about 100 lb of molten steel dropped accidentally into a trough of water. Lipsett concluded that the incident involved the conversion of about 30 lb of water to steam and that it created an air shock wave equivalent to that produced by 12 lb of TNT.

Several explosions have also been observed in the processing of molten aluminum. As a result, Long conducted a series of tests in which 50-lb batches of molten aluminum were poured into a tank of water. He observed explosions which were characterized by a violent bursting action, closely followed by a loud noise and a shock wave. He argued that some

triggering action dispersed the molten aluminum so that the heat transfer was fast enough to generate steam in an explosive manner. More recently, Hesse and Brondyke[61] reported very violent explosions from molten aluminum and water, but they gave evidence that the most violent of these involved chemical reactions as well as a release of thermal energy. The paper-pulp industry has also witnessed similar incidents.[62,63] Large furnaces have been destroyed when water, leaking from cooling pipes in the furnace walls, has contacted a molten mixture of sodium carbonate and sodium sulfide (known as smelt).

The steam-explosion phenomenon seems to give a reasonable explanation of the destructive pressure pulse experienced in the SPERT 1-D test.[64-66] This destructive pressure pulse occurred, not at the time of peak power, but 15 msec later, when about one-third of the aluminum fuel-plate surface was molten. It was probably triggered by the impact of water reentering the reactor core, which dispersed the molten aluminum and caused a steam explosion. This hypothesis has been partly substantiated by Zivi et al.[67,68] in shock-tube experiments, where a column of water was dropped on molten aluminum or silver and produced an explosion.

2. The Fragmentation Process

As part of the thermal-reactor safety program at Argonne, experimental work has been initiated to study such explosive interactions between molten materials and water.[69] Initial experiments have involved the injection of a small amount of water (1-2 cc) into molten metals and salts (100-200 g). Explosions have been observed with molten tin, potassium iodide, and silver chloride. One explosion with tin was sufficiently violent to shatter the upper half of the container, a $1\frac{1}{4}$ in.-ID, 1/4-in.-thick graphite crucible. High-speed motion pictures showed that, following the injection of the water, the molten surface bulged and partially collapsed before the explosion. The water-injection tests are continuing with different materials, such as aluminum, silver, and sodium chloride. A new technique is being developed to measure the thrust resulting from the explosions.

Several attempts have been made to explain these pressure pulses. Analyses of the SPERT 1-D test all indicate that the energy transfer between molten aluminum and water must have been extremely fast. Zivi et al.[67,68] noted that it could not be fully explained by known heat-transfer mechanisms, such as conduction through a steam film. The heat-transfer rate must be due to the creation of surface area by extensive fragmentation of the molten aluminum. This was demonstrated in Ivins and Baker's calculations.[70]

The fragmentation and dispersion of the hot material is perhaps the key factor that determines the heat-transfer rate. Various

mechanisms have been proposed for the fragmentation, depending on the situation under consideration. For the SPERT 1-D test, the impact of a bouncing water column probably caused the molten fuel plates to fragment. For his experiments, in which molten aluminum was poured into water, Long suggested that the fragmentation was triggered by a minor explosion due to the sudden boiling of a thin layer of water trapped between the bottom of the tank and the aluminum. A similar mechanism was later proposed by Brauer et al.,[71] the only difference being that they suggested that water is trapped in a falling molten-metal drop, rather than at the bottom of the tank.

The case when water is injected under the surface of molten materials is of considerable interest. The water can be treated as a jet or drop moving through the other fluid, and its breakup may be similar to that in other processes, such as rainfall or atomization. A more complicated mechanism of fragmentation is the intensive convection effects of boiling at the interface of molten materials and coolant.

Some other factors are also involved in explosive pressure generation. Superheating of the water may account for the fast process of energy conversion. Highly superheated water would be unstable and would suddenly flash to steam. Water coming into contact with a fresh surface of molten material may undergo considerably higher superheating than normally. Less obvious, but perhaps equally important, is the role of collapsing vapor bubbles, which may give rise to violent pressure pulses. The present understanding of these factors is far from complete.

C. Systematic Study of Vapor Explosions with Alkali Metals

Vapor explosions involving liquid metals as the working fluid are currently being attacked at Argonne by considering the process of energy transfer from the hot material and the subsequent rapid expansion of the liquid metal.

Three separate factors determine the amount of energy transferred: the rate of heat transfer across a unit area of interface between hot material and cold, the interfacial area of the system, and the time available for the energy transfer. Each of these factors is currently being investigated in a separate experimental program.

1. Heat-transfer Coefficient

Forced-convection heat transfer from hot spheres in liquid sodium is being studied. In this program, a 1/2-in.-diam tantalum sphere, mounted on a sheathed thermocouple arm, is inductively heated to a predetermined temperature and then swung through a molten sodium bath by a motor-driven shaft to which the arm is connected. Heat-transfer

rates are computed directly from the sphere temperature history (as measured by the thermocouple). Tests to date show heat-transfer rates on the order of 1000 cal/cm^2-sec. A more detailed description of the apparatus and results is presented in ANL-7296.[72]

2. Fragmentation

Initial studies of the fragmentation process involved the dropping of various molten samples (e.g., zirconium, nickel and stainless steel) into liquid sodium.[73] These showed extensive breakup of the sample and violent eruptions of the sodium. In an experiment with stainless steel, the sodium eruption was so vigorous that about 15% of the sample was ejected.

Later experiments[74] compared the effects of sodium and water. In many cases, there was a marked difference between them; for example, molten silver did not fragment when dropped into water, but did fragment when dropped into liquid sodium. It was hypothesized that the breakup results from violent formation and collapse of vapor bubbles in the nucleate and transition boiling regimes. The melting points of the cladding metals fall within the violent boiling range for sodium (900-2300°C), and so the metals fragment. On the other hand, the metals are not molten in the violent boiling range for water (100-400°C) and therefore do not fragment. Experiments in water using low-melting-point metals support this hypothesis.[74]

The fragmentation of a molten mass of material may depend either on velocity or on acceleration. Fragmentation due to the relative velocity of the hot and cold liquids would be characterized by the Weber number, N_{We},[75]

$$N_{We} = \frac{2r\rho v^2}{\sigma}, \tag{1}$$

where

r = radius of the droplet,

ρ = density of the liquid in which it is immersed,

v = relative velocity of the two liquids,

and

σ = interfacial tension.

The droplet is broken up when N_{We} exceeds a value of about 10-20.

The effect of acceleration may be thought of as being due to Taylor instability of the interface when the lighter fluid is accelerated

into the heavier. Such acceleration might be caused by vapor-bubble growth or collapse in violent boiling. If very fine particles are to be produced in this way, very violent accelerations (at least hundreds of g's) are needed.

Currently the fragmentation process is being studied using a tin-water system.[76] Single drops of molten tin at different temperatures are dropped from various heights into water. Changing the height changes the entrance velocity and varies the velocity effects on fragmentation; changing the drop temperature changes the boiling behavior and thus varies the acceleration effects.

Runs made with metal at temperatures either considerably below (255°C) or above (550 and 600°C) the critical temperature of water (373°C) result in the formation of fragments having roughly the same surface-area density (~4 cm^2/g) for similar Weber numbers. At 310 and 385°C (close to the critical temperature of water), there is a significant increase in the surface-area density (~10 cm^2/g) and an increased scatter of the data. At all temperatures, there is a trend of increasing surface-area density with increasing Weber number. In all cases, the particle-size distributions were bimodal in nature, suggesting that there may be two fragmentation processes at work.

Time-exposure photographs, made under stroboscopic illumination, have been taken of molten tin falling into water. These show that an initial breakup occurs at or near the surface of the water, probably as a result of the dynamic forces acting on the molten metal or entrance. In some cases, a second and more violent fragmentation takes place some distance below the surface (about 2 cm below when the initial temperature of the tin is 330°C).

A few general observations about the distribution of fragments are:

a. In all cases, a double fragment-size distribution is observed.

b. Increasing the drop height (entrance velocity) increases the surface-area density of the fragments.

c. The greatest amount of fragmentation occurs at metal temperatures near the critical temperature of water (373°C).

Similar experimental programs using stainless steel or uranium dioxide and sodium are being started.

3. Time Scale and Sequence of Events

The third important factor determining the outcome of a vapor explosion is the time available for energy transfer. This is being investigated

in a series of tests in the TREAT reactor, the object of which is to determine the sequence of events and time scale in the sort of reactor accidents that might result in a vapor explosion.[77]

The autoclave used in these experiments is capable of withstanding 400 atm at 315°C. It contains a cluster of five fuel pins (containing uranium dioxide) and four empty pins, all submerged in static sodium. It is sealed by a movable piston attached to a linear motion transducer to measure the expansion of the sodium, and is also equipped with thermocouples and a pressure transducer.

The test of greatest interest so far involved the generation of 460 cal/g UO_2 in the fuel, and resulted in the destruction of all five fuel pins. The pressure trace showed five separate pulses, perhaps resulting from the separate failures of the five fuel pins.

The first two pulses were nearly simultaneous and were coincident with a rapid increase in the piston velocity from 6 to 302 cm/sec. The third pulse was coincident with a further increase in piston velocity from 302 to 472 cm/sec. The final two pulses occurred after the piston was locked in its final position.

The peak temperature of the sodium was 1100°C, recorded by a thermocouple located at the upper portion of the fueled section of the fuel rod 0.8 sec after peak power. All other thermocouples recorded temperatures below 500°C. Only a very small fraction (about 5×10^{-5}) of the nuclear energy was converted to mechanical energy. Very similar conversion ratios were observed in tests in the Capsule Driver Core, where similar experiments have been carried out with samples of thermal-power-reactor fuels in a water environment.

The residue indicated that the five fuel pins were completely destroyed during the transient. Only small (~1-cm-long) cylindrical sections of fused UO_2 remained. These were hollow (0.5-cm ID) and had an outside diameter of 0.673 cm. Since this is greater than the original diameter (0.635 cm) of the pellets, it appears that during the transient the pellets melted and expanded into intimate contact with the cladding (0.671-cm ID), whereupon the cladding melted and a portion of the UO_2 solidified. The rest of the oxide flowed away, leaving a solid annulus.

In spite of this complete destruction of the fuel, there does not seem to be any evidence that a vapor explosion occurred in this test. The pressure generated can be attributed largely to thermal expansion of the sodium and to the gases released to the coolant. Sodium boiling also occurred, but did not create any excessive pressures. This evidence, although very negative, is a first step in understanding the time scale and sequence

of events that would be necessary in a reactor accident to produce a vapor explosion of damaging violence. It is hoped that subsequent tests of this series will clarify the situation.

D. Vapor Explosions in LMFBR Accidents

The concern in an LMFBR is that if the fuel pins lose their integrity, hot fuel will come into contact with liquid coolant. The fuel-center temperature in normal operation is above 2000°C. If liquid sodium is heated rapidly at constant volume from 600 to 2000°C, its pressure rises to a very high level, probably well above 10^3 atm. Constant-volume heating is the worst imaginable case, but illustrates that the potential exists for a very damaging mechanism. The amount of damage actually done depends on other factors, such as the available work and the impulse in the shock waves.

Consideration of vapor explosions as a separate problem from other aspects of coolant boiling in a fast reactor is in a sense artificial. In fact, a vapor explosion is only one extreme of the range of possible interactions between hot fuel and coolant. This range includes coolant boiling from intact fuel surfaces (discussed in Chapter III), and all the ways disintegrating fuel may behave (discussed in Chapter VI). This artificiality is apparent in that the study of vapor explosions, particularly in in-pile tests, is best conducted as part of the wider problem of fuel-coolant interactions.

In the reactor-safety context, however, vapor explosions merit separate consideration because once the interaction is explosive a new range of more serious consequences becomes possible.

1. Possible Effects of Vapor Explosion

There are three main areas where a vapor explosion could have serious consequences. These may be distinguished by the scale of the possible explosion, and in particular by whether it involves a fuel pin, a subassembly, or the whole core. This division is arbitrary, and, for example, vapor explosions involving a few pins or a few subassemblies may be possible. It is convenient, however, to divide the problem up for consideration in this way.

a. Single Pin. If a fuel pin fails, can fuel be expelled through the hole in the cladding and cause a vapor explosion in the surrounding coolant? We are particularly concerned about cases such as a mild power transient which causes failure of a defective pin without boiling the coolant. The question is whether the vapor explosion, if it occurs, is violent enough to precipitate failure of another pin.

This is an important aspect of the fuel-pin failure-propagation problem, and its significance is discussed in Chapter IV. It must be remembered, however, that a vapor explosion is only one of several possible failure propagation mechanisms.

b. Subassembly. Can a vapor explosion involving a whole subassembly cause significant damage to the rest of the core? Such a vapor explosion might occur, for example, when a subassembly is blocked, the coolant boils and is ejected, the fuel melts, and then cold coolant reenters and comes into contact with the hot fuel. The consequences would be of concern if much of the rest of the fuel in the core were damaged so that it had to be replaced, if the control-rod mechanisms were damaged so that the reactor could not be controlled or shut down, or (the worst possibility) if the rest of the fuel could be compacted to cause a super-prompt-critical excursion.

If any of these can occur, it is important to know whether the events leading up to the vapor explosion can be detected, and how, and whether action can be taken in time to prevent damage. For example, if the reactor can be shut down before the coolant reenters, the potential for damage may be reduced considerably.

This is the most important aspect of the vapor-explosion question, and one of the critical problems of LMFBR safety.

c. Whole Core. Can a large vapor explosion occur in the core debris after an accident that destroys the fuel, and if so could this convert a significant amount of the residual heat into mechanical work? For example, most of the energy generated by fission during a super-prompt-critical excursion appears as heat, eventually in the form of very hot liquid fuel, and only a small fraction, typically around 10%, is converted to kinetic energy of the core components. It is this kinetic energy that makes the explosive effect of the excursion.

It is pointed out in Ref. 58 that it is thermodynamically possible for about 10% of the heat in liquid fuel at its vaporization temperature to appear as work done at pressures above 70 atm. If a vapor explosion could indeed convert this much heat into explosive work, it could do as much damage as the original excursion.

Precise definition of the importance of a vapor explosion in this context is difficult, since little is known of the possible effect of a second explosion on the surviving structure and containment. The vapor explosion in a single subassembly is likely to be more important than the vapor explosion in a whole core since, in the latter case, the vapor explosion cannot make a large qualitative change in the consequences of the accident. Major

damage to the core has already occurred; subsequent damage may not make the accident much worse (although the implications for containment design would have to be studied). On the other hand, a vapor explosion may play a very important role in an accident in which the fuel is destroyed but there is no super-prompt-critical excursion.

2. Analytical Problems and Data Needs

The most important need is for a better understanding of the mechanisms at work in a vapor explosion, and in particular of how the materials become so finely divided as to transfer heat so rapidly, and of the rate at which this subdivision process occurs.

The subdivision rate appears to be the most important aspect of a vapor explosion. If surface area is created rapidly, compared with the time scale for sonic relief of the pressures generated, an explosion results. This, rather than the surface heat flux from the hot particles, is the decisive factor, as pointed out above.

The experimental evidence described above suggests that the subdivision process takes place in two phases. First, something violent happens (such as the mechanical impact of a column of liquid on a hot molten surface, or the boiling of a trapped liquid volume), which begins to propel the hot material into the cold. This then in some way triggers an autocatalytic subdivision of the hot material, in which heat transferred to the cold liquid causes intense local pressure disturbances, which make the hot material break up and transfer heat still faster.

Both of these phases have to be understood, as far as possible, especially in the context of hot ceramics and liquid metals. For example, the hot material probably has to be liquid if a vapor explosion is to take place, but we wish to prove this point. We also wish to know what mechanisms that might exist in a fast-reactor accident could act as initiators to start the subdivision process. Most important, we wish to know what limits the generation of high pressures and what determines the efficiency of conversion of heat to work. As shown in Ref. 58, a theoretical thermodynamic upper limit can be placed on this efficiency. In some cases (for example, the accident described in Ref. 59), the actual efficiency may have approached this limit. We wish to know whether this could happen in a fast reactor.

We then have to understand what determines the scale of a vapor explosion. We have spoken above of a possible vapor explosion involving all the fuel in a subassembly, or even all the fuel in the core, but we are not certain whether such large quantities can act coherently, and break up simultaneously to produce a large explosion.

Having reached this understanding, we may need to check whether a vapor explosion that can be induced in the laboratory is likely to occur in a fast reactor. This would need in-pile tests, but this need cannot be defined clearly until the phenomenon is better understood, as outlined above.

Finally, the possibility of a vapor explosion has a strong bearing on the safety of LMFBR safety experiments, of FFTF, or any LMFBR demonstration plant. This consideration emphasizes the need for confirming as good a basic understanding of the phenomenon as possible as soon as possible.

In Section 1 above, we have described the limited data already available from the records of various vapor-explosion accidents, from out-of-pile laboratory tests, and from a few TREAT tests. From these, a fairly good understanding of the conditions necessary to cause a vapor explosion, and of the nature of some of the mechanisms involved, has been reached. A vapor explosion is very complex, however, and different mechanisms for the subdivision process may be important under different conditions or for different materials. Thus, although some aspects of the phenomenon are understood, there is a need to fill in the details. In addition, there is a lack of quantitative data at present.

3. Out-of-pile Program

The out-of-pile work needed can be classified as follows:

a. Subdivision Mechanism. The generation of a large surface area for heat transfer is at the center of the problem, and therefore must be studied if vapor explosions are to be thoroughly understood (although, of course, some progress can be made without this fundamental knowledge). This is a random process, which takes place very quickly on a very small scale, and so it is clear that any experimental observations can hardly be made in-pile.

Because of the randomness of the subdivision process, the best approach to a quantitative understanding of the rate and extent of subdivision is probably empirical, based on our qualitative ideas about the basic process. This requires large numbers of out-of-pile tests, all on a fairly small scale. Work on these lines, mainly with water as the cold liquid, is in progress as described in Section 2 above; and therefore the need is to pursue this vigorously and extend it to other liquids.

b. Pressure and Efficiency. One direction in which we may hope to proceed without a quantitative understanding of the basic processes is that of studying the macroscopic aspects of the explosions, such as the peak pressure generated, the time scale of the pressure pulse, and the efficiency of conversion of heat to work. All these quantities will depend on the

nature of the materials involved, and on the initial conditions, such as temperature, energy input, physical form, and the manner of mixing. It may be possible to measure these dependences by performing a range of experiments in which the initial conditions and the materials are varied and the pressures generated are measured.

Such experiments are being done out-of-pile with water and are planned with sodium. Perhaps these should be given greater emphasis. In addition, the autoclave experiments in TREAT may well be very useful, as TREAT is a convenient tool for putting a controlled amount of energy rapidly into the sample. (When TREAT is used in this way, no attempt is being made to simulate a reactor accident. TREAT is simply being used as a convenient heat source. It is useful to make this distinction between two modes of using the facility.)

c. Scale and Coherence. Another aspect of the problem which may be amenable to separate study is that of the influence of size on the consequences of an explosion. We wish to know whether the same pressures and efficiencies can be attained in a vapor explosion involving a great deal of hot material (say 100 kg, which is about the amount of fuel in an LMFBR subassembly) or in a small one involving a few grams.

Clearly this may need large-scale experiments. Up to the 1-kg scale, and perhaps beyond, meaningful experiments can probably be done out-of-pile, and it may be possible to extrapolate to larger quantities. A large reactor facility would be a useful tool for doing larger experiments.

4. In-pile Program

Finally, it is necessary to check whether the violent event that seems to be needed to trigger a vapor explosion and the necessary close contact between molten fuel and coolant, can arise in an LMFBR accident. A better understanding of initiators from analysis of existing experimental data may make this point clearer; but even so, in-pile tests may be needed, even though such tests would be very expensive.

The point here is that while it may be possible to induce a vapor explosion between hot fuel and sodium under the right conditions, these conditions may never arise in an LMFBR accident. For example, after the coolant is ejected by boiling from an overheated subassembly, it may be impossible for the coolant to return and contact the molten fuel.

Here the vapor-explosion problem merges with the wider one of characterizing the behavior of melting and disintegrating fuel: the problem of establishing the sequence of events in a fuel-melting accident after the coolant has boiled away. This is one of the crucial questions of fast-reactor safety (see Section III.B.3) and is discussed at length in Chapter VI.

The sort of questions that have to be answered to complete our study of vapor explosions are, for example, "What does a subassembly look like after voiding when the fuel has melted and the coolant is about to reenter?", or "What is the state of the core debris after a super-prompt-critical excursion?". These are exactly the questions that appear in the contexts discussed in Chapters VI and VII.

Thus, the need for in-pile tests is precisely the same as that pointed out in Chapters VI and VII, and the vapor-explosion question strongly reinforces the need for more data on the behavior of disintegrating fuel.

CHAPTER VI

MOTION OF MOLTEN FUEL

A. Introduction[78,79]

In earlier chapters, we discussed the program of work necessary to resolve problems associated with local anomalies such as single fuel pin failure. In this chapter, we are concerned with the more complex situation that arises when there is large-scale movement or melting of fuel and coolant motion on a rapid time scale, as in a serious accident. The main concern is that such fuel movement may initiate a super-prompt-critical accident. Such accidents are discussed thoroughly in Chapter VII; this chapter concentrates on this single possible initiating mechanism because it presents peculiarly difficult problems.

A nuclear characteristic of fast reactors is that the normal configuration of the fuel is not optimum as far as reactivity is concerned. This characteristic emphasizes the importance of the rate of fuel motion. Most fast-reactor cores, if optimally arranged, contain several times a critical mass. Thus, large amounts of reactivity are potentially available from any phenomenon that moves the fuel into geometrical arrangements of higher average fuel density. This might happen if the fuel material melted and settled with the fuel cladding, or it might result upon escape of molten fuel from the fuel cladding and reassembly into a more dense configuration, possibly near the center or bottom of the core.

B. Reactivity-addition Rate

It is instructive to consider the effect of the rate of addition of reactivity on the yield of a super-prompt-critical excursion. Chapter VII describes this effect in detail, but for the purpose of illustration the simple Bethe-Tait model is useful.

The Bethe-Tait model of a super-prompt-critical excursion contains the essential features needed to describe such an accident in broad terms.[80] This method is a semianalytical way of calculating the nuclear energy produced during a severe accident which is terminated by disassembly of the fast-reactor core under the action of high pressures generated in the fissioning materials.

Under simple assumptions, including the neglect of the Doppler coefficient, it can be shown that the energy yield will depend on the reactivity ramp rate (i.e., the rate of addition of reactivity at the instant of prompt criticality) as the 3/8 power for small yields and as the 3/2 power for very large yields.[79,80] The inclusion of other feedbacks than fuel dispersion, particularly those coming from coolant expulsion and from the Doppler effect, makes the analysis very much more complicated, but the rate of reactivity insertion remains of crucial importance.

Rapid-reactivity insertion may be obtained by the melting of fuel and its collapse from regions of low-reactivity worth such as the blanket interface to regions of higher-reactivity worth such as the core midplane. For the Fermi reactor, typical ramp rates corresponding to fractional collapse in core volume required to reach prompt critical were calculated to range from 0.35 to 1.42 $\Delta k/k$ per sec for collapse of the fuel uniformly under the action of gravity, with the lower core boundary being held stationary. (In recent 1000-MWe studies, ramp rates as high as 0.42 $\Delta k/k$ per sec were calculated for large, coherent accidents.) In addition to the insertion of reactivity through fuel meltdown and collapse, another potential source of rapid insertion of positive reactivity is the motion of solid fuel into a region of higher reactivity worth. Such motion might, for example, be induced by a vapor explosion in a portion of the core.

Another potential source of rapid reactivity in section is large-scale expulsion of coolant.

There is, of course, a liquid-phase sodium-density coefficient of reactivity which is small with respect to the Doppler effect and in general will have little influence on reactor stability or reactivity excursions. However, for large fast reactors, large positive-reactivity insertions are potentially possible, due to sodium voiding. The reactivity to be gained if sodium should be voided in exactly the worst fashion is in many designs greater than the compensating reactivity available from the Doppler effect. Thus, it is important to ascertain whether any real possibilities exist for introducing significant portions of the total reactivity that could be gained by the worst voiding configuration. The actual mode and timing of sodium expulsion is vital to any assessment of the amount and rate of reactivity insertion possible from this cause. In principle, partial sodium loss in the center of the core could give significant reactivity addition, even though the reactivity change due to total coolant loss were negative.

It has been estimated that a subassembly might be fully voided of coolant in times of the order of 50-100 msec, were sodium boiling to be initiated in that subassembly at full power. Knowledge of the overall time scale of events is required to assess the significance of such an expulsion.

Two prominent mechanisms that have a high potential for rapid coolant voiding are:

a. Fuel-cladding failure due to pressure from within might cause coolant-flow blockage or dispersal of fuel particles and fission gas into the coolant. This has been discussed in Chapter IV with respect to fuel-failure propagation. If a large number of fuel pins (in more than one subassembly) should fail simultaneously, then large coherent voids might be obtained simultaneously and large reactivity ramps introduced.

b. A gross power-flow mismatch can lead to coolant boiling from overheating. This mismatch might occur from pump coastdown with delayed scram, blockage of a subassembly inlet by foreign material, either suddenly or slowly, or failure of the regulating system.

The program of work to study such problems when the fuel is intact, or when only a small portion of the fuel is failing, has been discussed in Chapters III and IV. It must be recognized, however, that core structural or fuel-element damage might occur before the bulk-boiling failure point. Such events are the concern of this chapter.

Under conditions of a power excursion, the temperature rise in the cladding will lag that in the fuel. If the power excursion is rapid enough, pin failure can occur due to internal pressures while little, if any, coolant vaporization has occurred. This type of behavior would be particularly favored for flowing sodium, a large fission-gas inventory (either in a plenum or trapped inside the fuel structure), and oxide fuels with their large radial temperature gradients at steady state combined with low thermal conductivity. Fuel-pin failure before most of the channel has been voided has been found, even in TREAT loop experiments on high-conductivity metal fuel.[81]

Thus, failure before large-scale boiling needs to be considered if power transients are to be analyzed. Coolant boiling before failure in one region of the reactor can introduce a reactivity insertion through a local positive sodium void coefficient of reactivity. Such a transient might well cause fuel in other regions of the reactor to fail before large-scale boiling has occurred. For such reasons, we require some way of (1) characterizing the coolant motion under the very difficult conditions that arise when there is large-scale fuel failure and motion, and (2) estimating the reactivity-addition rates.

C. Analytical Problems and Data Needs

It is important to set the bounds of the magnitude of physical effects and determine the sequence and the time scale of the events. Precise knowledge is not as important as a knowledge of the bounds, because a precise description of super-prompt-critical accidents is not generally possible on a realistic basis. Reactivity effects can be estimated satisfactorily by combining the knowledge of fuel and coolant dynamics with reactivity estimates obtained from critical experiments and calculations.

To predict the motions that determine the reactivity insertion rate on theoretical grounds would require taking into consideration the failure mechanism of the cladding and the manner in which the fuel escapes from the fuel element, the forces of the coolant, gravity, and the driving force from the fission and bond gases that the fuel is subjected to after it leaves

the fuel element. In addition, one would have to consider the cooling effect of the coolant vapor or liquid and the possibility of freezing and sticking of the fuel and blockage of the coolant as the fuel moves in accordance with the net force exerted by gravity, coolant, and contacts with subassembly sidewalls, fuel-element spacers, support pieces, and blanket rods. For a given core a detailed analysis would have to be made for each independent variable such as fuel burnup, reactor power, and coolant velocity at the time of meltdown. There are a great many driving forces and other dynamic effects that require complex analysis that cannot be carried out at this time with confidence. An entirely different situation exists for each specific reactor where changes in fuel material, cladding materials, detailed fuel element and subassembly design, and core hydraulic characteristics are involved. It is probably not possible, on purely theoretical grounds, to predict fuel motion under the complex conditions that exist in the meltdown. Therefore, an experimental approach is essential.

Related to the problem of the description of fuel and coolant conditions producing the nuclear excursion is that of describing the postburst energy transfer from the hot fuel to the sodium in the core nearby. This problem was discussed in Chapter V. Estimates of the upper-limit damaging work that could be done in a "vapor explosion" of sodium indicate that the effects of the sodium explosion could be comparable to that of a super-prompt-critical nuclear burst.[58] Calculations of the magnitude of a sodium-vapor explosion following a given nuclear excursion will require information on the state of the core, its degree of subdivision, where the sodium is located, the time scale of sodium movement into contact with hot fuel, and the various regimes of fuel-coolant thermal interaction. Because thermal interactions after an energetic nuclear burst will include fuel vapor-coolant interactions for fuel vapor densities of ~1 g/cc, one can expect these postburst phenomena to be different from the fuel-coolant thermal interactions associated with fuel meltdown.

Data gathered in an experimental program would be used in several different ways:

1. Model Construction

The knowledge of the sequence and time scale of events is vital to setting up analytical accident models to calculate the energy release in super-prompt-critical accidents and the effects of accidents that do not lead to super-prompt-critical accidents. Such analytical tools are needed to assess the effects of design variations.

2. Reactivity Insertion Rate

Measurements of fuel and coolant motion can be combined with critical experiment data and calculations of reactivity worth to give an estimate of reactivity addition rate. The estimate is naturally imprecise

for a number of reasons, such as random errors in experimental data, neglect of complex spectral shifts and temperature effects, and changes in heterogeneity. But the precision needed is not great, particularly when existing conservative estimates may be pessimistic by orders of magnitude.

3. Scope of "Weak" Accidents

In the case of slower failures marked by events spread out in space in time, mixing of fuel and sodium, etc., the experimental data would be used to set the margins required for "engineered safety features" and to determine the probable extent of "less serious" accidents. Such accidents represent the average effect of many events. A detailed representation is therefore unreasonable, but the broad averages used in the analysis of super-prompt-critical accidents are not directly applicable, because of the difference in time scales. By use of scoping experiments to suggest effective models for calculation, these difficult calculational problems can be circumvented.

4. Plant Design Data

Postfailure examination of melted subassemblies would yield realistic information on the extent of these plant-design problems:

a. Will molten fuel damage the reactor structure?

b. Will the removal of damaged subassemblies from a reactor be a severe problem?

c. Can secondary criticality occur? Where does the molten fuel tend to deposit; will it tend to agglomerate; and what are its constituents?

d. What are the heating rate and heat-transfer conditions in collections of cooling fuel? Are special coolant-flow conditions required during shutdown following a subassembly meltdown? These conditions also bear on emergency cooling conditions and designs. (See also Chapters VII and VIII.)

5. Core Mechanical Design Data

Although the entire program would be related to core safety, we have in mind here the utility of dynamic measurements to yield the stress and temperature loads on core structures under accident conditions. The effectiveness of engineered safety features may depend on the ability of core structure such as control-rod guides to withstand pressure and temperature loads induced, for example, by coolant-fuel interactions.

6. Energy Exchange

This topic is discussed more thoroughly in Chapter VII. Here we refer to the exchange of energy between molten fuel in bulk and sodium in bulk, which could potentially generate large quantities of sodium vapor, such as may occur following a super-prompt-critical accident. Meltdown experiments, especially those on a rapid time scale, should yield information applicable at least to the early stages of this process. If the experimental test loop is large enough and decay-heat simulation can be satisfactorily maintained, data may be obtained that are directly applicable to containment design.

7. Phenomenological Models of Fuel Melting

At this time, assurance cannot be given that experiments on the dynamics of fuel meltdown would yield data to fit preestablished models (see 1, above). Rather, past experience with metallic fuels in TREAT indicates that preliminary experiments will be needed to define the modes of failure, and then models can be constructed. The data from more complex experiments can be used to determine correct values for the model parameters. The aim of an experimental program would be to produce data that could be extrapolated to a wide range of designs. Accordingly, considerable flexibility in the test facilities is required to ensure that there is capability to simulate all the important phenomena.

D. Out-of-pile Program

The program of work to provide the needed data would involve out-of-pile work as well as in-pile work.

In general, the detailed observations needed to determine the sequence of events and their time scale are difficult to make in-pile without some preliminary idea of what to expect. Out-of-pile simulations using, for example, electrical heating to generate heat in mockup fuel pins need to be performed to determine the orders of magnitude involved. An early example of tests of this sort is given in APDA-155.[82] The complexity of the events involved means that careful analysis of the simulation technique is required to avoid being led astray. Out-of-pile simulations also can establish the appropriate mechanical boundary conditions on a full-size subassembly.

E. In-pile Program

1. Objectives

A program of in-pile experiments to resolve the questions of fuel motion and coolant dynamics would have these objectives:

a. Determine the likely reactivity increase and rate of increase that can be achieved in fuel meltdown.

b. Find out if damage can propagate throughout a subassembly or between subassemblies to cause fuel meltdown.

c. Given a satisfactory degree of survival of fuel subassemblies after an accident, determine if the fuel power or temperature ratings of the subassemblies need to be reduced. (See also Chapter V.)

d. Determine the final disposition of the melted fuel.

e. Define an empirical model of fuel meltdown and coolant interaction.

The determination of the reactivity increase that can be achieved in fuel meltdown is a question of primary interest. The required program here is complex and involves tests on few pins, small-scale subassemblies, and full-scale subassemblies. As pointed out in APDA-151,[78] knowledge of meltdown dynamics in one subassembly permits a good estimate of the dynamics of a whole-core meltdown.

2. Test Scale--Spatial Incoherence

Few-pin tests have been carried out in TREAT and other irradiation facilities. The problem of test size arises because of the cooperative nature of the effects being examined and the crucial role of coherence on a large geometrical scale. Generally speaking, tests carried out on fewer pins are easier to instrument and observe in detail than are tests carried out on large numbers of pins. On the other hand, it is important to be able to determine effects that involve cooperation or the lack of it between many neighboring pins. For example, if boiling should start in one part of a large subassembly exposed to a power gradient (as it would be, for example, at the edge of an enrichment zone), how does the void propagate and cross the subassembly into the colder zones? Does the fuel in the colder zones break up before boiling starts? Are the rate at which the void propagates and its manner of propagation dependent on the gradient of power across the subassembly? If so, what is this dependence? It is unlikely that precise answers to such detailed questions can ever be attained, but indications of thresholds and bounds are important and useful in accident analysis.

Another consideration is the role of the subassembly wrapper itself. Normally the wrapper is operating at a lower temperature than the cladding and frequently possesses considerably different structural properties. Thus, it offers a much different boundary condition for flow effects and thermal effects than the normal channel boundary. In any given subassembly, a fraction of the coolant channels will have as one boundary the subassembly wrapper. In a 19-pin subassembly, 12 of the pins have

peripheral channel boundary conditions while seven pins have internal channels. In a 169-pin subassembly, which may be taken to be typical of large designs, less than 25% of the pins have channels with a peripheral boundary condition. Thus, in conducting tests on a fuel pin or small-scale subassemblies and extrapolating to full-scale subassemblies, we must be able to scale wrapper effects where necessary. In general, where only one phenomenon is predominant, such as coolant voiding or fuel melting, extrapolation may be made with confidence, albeit with difficulty. But where there are competing effects, and especially where coherence, or the lack of it, determines the course of the competition, extrapolation may not be a reasonable course of action.

3. Test Scale--Temporal Incoherence

By temporal coherence we mean the simultaneous occurrence of a given phenomenon. Thus, voiding a few channels simultaneously will, in general, add only a very small amount of reactivity. But if a large number of channels (usually involving more than one subassembly) are swept clear of sodium in the same short period, the reactivity addition rate could be very high. On the other hand, should the channels void incoherently (i.e., a few at a time), the average reactivity addition rate will be low, and there will be a better chance that natural feedbacks and safety mechanisms will have time to be effective.

While coherence effects can be estimated for a single phenomenon such as voiding of various channels with all conditions (flowrate, power distribution, channel size, etc.) but one kept constant, the estimation is impossible when, as in fuel meltdown, a large number of phenomena occur many of which are only crudely understood. The problem takes on a statistical nature which requires resolution by numerous tests with accurate simulation.

There are a few analytical and experimental studies of the effects of temporal and spatial coherence. Gopinath and Dickerman[83] have examined incoherence on the subassembly scale in EBR-II arising from local variations in power and flow. Their analysis, while detailed, is based on simplifying assumptions that give the work some extended applicability. Some of their results are summarized in Table I, where we list the expected spread in failure times in EBR-II subassemblies under various accident conditions and summarize the effects on accident calculation. These estimates do not include the effects of random variations in channel diameter, pin diameter, enrichment, etc. Such variations can only increase the spread.

More recently, studies of few-pin interactions with sodium in TREAT under meltdown conditions have revealed a large spread in times when pressure pulses occur.[76] Pressure pulses were observed over 0.5 sec. This is probably due to spatial incoherence between peripheral and central

pins. Similar time spreads have been estimated from earlier TREAT experiments, but the data are meager. Calculated spreads in failure times for metal fuels (~0.1 sec) tend to agree with the observations.

TABLE I. Expected Spread in Failure Times in EBR-II Subassemblies under Accident Conditions

Accident	Failure-time Spread, sec	Effects on Accident Calculation
Pump power failure at constant power	2 (for different pins in one subassembly)	Reduce reactivity insertion rate by factor of 3 to 5 over coherent assumption.
	+2 (for different subassemblies)	
Blocked orifice at constant power	0.15 (in one subassembly)	Reduce reactivity insertion rate by factor of 2.5 to 4 over coherent assumption.
	0.5-1.0 (between subassemblies)	
Dropped subassembly	0.04	Reduce ramp rate from fuel failure by factor of ~2 over coherent assumption.

Analytical studies of the effects of incoherence in super-prompt-critical accidents are summarized in Ref. 84. There it is shown that the energy release is reduced substantially if the mean time between random reactivity insertions is greater than ℓ/ρ_0, where ℓ is the reactor prompt-neutron lifetime (sec) and ρ_0 the reactivity insertion expected from individual events. For a reactor with about 50,000 fuel pins, the potential insertion, ρ_0, per pin may typically be of the order of 0.4×10^{-5}. For ℓ of the order of 0.5×10^{-6}, we estimate the critical mean time between failures to be about 0.125 sec. This time is a minimum, being calculated for the super-prompt-critical case. From Table I and the TREAT data, we conclude that lack of coherence on a subassembly scale can reduce considerably the consequences of serious accidents. It is important to know what this reduction is.

The issue of coherence is important, because it indicates a need for full-scale subassembly size tests. It is tempting, from the preceding arguments, to conclude that less than full-scale tests are at least conservative with respect to reactivity-insertion rates and other causes of damage, but this is not always so. Propagation of damage from subassembly to subassembly may be a problem, especially when it is caused by fuel-coolant interactions of the vapor-explosion type.

4. Damage Propagation between Subassemblies

Damage propagation from fuel-pin failure implies a type of coherence in that one failure induces another. If succeeding failures are larger, then the damage propagates at an increasing rate until some natural

limit is reached. Generally, the subassembly wrapper is expected to act as a barrier to damage propagation, but it may not always be effective in this regard.

A possible source of subassembly-to-subassembly propagation is the vapor explosion that may arise when coolant reenters a subassembly that contains molten fuel. Incoherence of vapor collapse in large subassemblies is likely to produce liquid coolant reentry patterns leading to local regions of intimate mixing of molten fuel and coolant, and thus make a vapor explosion probable. Since large-scale incoherence would promote this type of mixing, small-scale tests need not be conservative with respect to the frequency of occurrence of a vapor explosion. Similarly, small-scale tests might not accurately predict the extent to which the explosive energy approaches the theoretical (thermodynamic) limit. Should such a vapor explosion occur in a small-scale test, however, the test would provide a good simulation of the energy-transfer modes available to cause damage to propagate between subassemblies.

We do expect small-scale tests, similar to those involving loops in TREAT which would simulate channel geometry, to yield valuable data on regimes of mixing and vapor generation, the relative explosive energy efficiency for each regime, and some indication of system scaling laws. However, we have reservations about extrapolating those data to real subassemblies, since the tests in most cases do not include realistic sodium heads, representative ratios of wall area to coolant volume, coolant volume to fuel volume, etc. Since one can calculate theoretical efficiencies of the order of 10% and greater for conversion of thermal energy into explosive energy in a vapor explosion, proper characterization of real subassembly effects is vital. A vigorous effort on small-scale tests should precede large-scale subassembly work, in order to provide data for both planning and analyzing it.

Some problems of scale can be more directly visualized. The molten fuel-coolant interaction poses a safety problem whenever thermal energy is transferred to the coolant faster than it can be radiated away by acoustic radiation. Imagine a small experimental volume bounded by relatively cold walls. The heat sink at the walls represents a process for losing thermal energy not present in a large-scale experiment. An interesting variant on such an experiment would be the use of insulated walls.

Another possible source of damage propagation between subassemblies is the penetration of the wrappers by molten fuel. Approximate calculations indicate the penetration may typically require 10-30 sec. It is reasonable to expect that safety systems can cause effective action to occur in this time. More accurate calculations are needed to define the required action and instrument response. Tests on groups of small-scale subassemblies can be made to confirm the adequacy of instrument response and of the core protective system action.

In general, tests of the propagation of damage between subassemblies are expected to be large in scale and, hence, difficult and expensive. Where scaled-down tests indicate the need for full-scale simulation, they may be used to set the scope of the larger tests and hence limit the number needed.

A detailed outline of a possible series of in-pile tests on the problems described above is given in Appendix A. Problems of instrumentation of in-pile tests are discussed in Chapter IX and Appendix B.

CHAPTER VII

SEVERE EXCURSIONS

A. Introduction--Relationship to Safety

1. Sources of Reactivity Increase

A number of sources of reactivity increase can be visualized that are potentially capable of initiating a super-prompt-critical excursion.* A primary concern of the reactor designer is to provide inherent design features and redundant control and safety systems to render such accidental excursions acceptably low in probability. Nevertheless, thorough knowledge of the consequences and course of events in unchecked accidents of this type is a valuable aid to the designer, the regulating agencies, the reactor operator, and the insurance companies. For these purposes, it is useful to study both the accidents that are difficult to prevent with a high degree of assurance and those that seem trivially easy to prevent.

Possible sources of reactivity increase, both direct and indirect, have been touched on throughout this report. Some of them are restated here to help define the range of conditions and, in particular, the rates of reactivity increase at prompt critical, which ought to be covered in a reasonably complete experimental and analytical program of investigation on super-prompt-critical excursions. The possible sources of reactivity are:

Control-rod ejection
Dropping a fuel subassembly
Pump failure
Flow blockage
Fuel-pin failure and failure propagation
Sodium boiling
Gas bubble
Fuel melting and movement
Implosions

One way the range of conditions can be defined is by the reactivity addition rate, $(\Delta k/k)/\text{sec}$.** There are primary effects which trigger an accident situation and either produce a reactivity increase directly or through subsequent secondary effects that in some cases will only take place if the control and safety systems also malfunction. The control rods themselves are one obvious source of accidental reactivity increase. It is equally obvious that there are many ways of ensuring against this, but for the purposes of this report it is assumed that in spite of all efforts to the contrary,

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* In this chapter the emphasis is on super-prompt-critical excursions. Excursions below prompt critical can also cause severe damage, but this has been alluded to in Chapter VI.

** In super-prompt-critical excursions, the delayed-neutron fraction, β, loses much of its significance so that [illegible]. In dollars/sec, $1(\Delta k/k)/\text{sec}$ is typically about \$200-300/sec.

someday a control-rod malfunction can cause a reactivity excursion. The potential magnitude and rate of reactivity increase are very much dependent upon the reactor design. In most cases, a single control rod will carry less than 1% reactivity,[86,87] and the drive motors will be limited to moving it in the direction of increasing reactivity to perhaps a few inches per minute or less. One has to consider whether runaway motors are possible or whether ejection by some accidental source of pressure is possible. Next is the question of whether simultaneous ejection of several rods should be considered. In this latter extreme case, if one assumes the total rod reactivity is 0.05 and the ejection is ~0.1 sec, it is possible to visualize rates of reactivity increase of $\Delta k/k = 0.5\ sec^{-1}$ as an upper limit. This is somewhat larger than most reactor designers would consider reasonable, but we cannot preclude the possibility that they may decide to raise their figures to about this level. It seems suitable as a basis for defining the range of interest for the experimental safety program.

Subassembly dropping accidents have about the same potential as control-rod malfunctions. Fuel subassemblies typically have reactivity worth of ~1% or less,[85,86] and they are generally handled one at a time. One visualizes, at worst, a situation in which the worth of the assemblies has been underestimated so that the reactor is critical when a subassembly is about to be lowered into the reactor, and it is then dropped. Suppose, for example, the extreme case in which a subassembly supposed to be of 10% enrichment having 1% reactivity worth is by accident actually fully enriched and worth 10%. If this is dropped into the core, it is again possible to attain rates of reactivity increase of the order of $\Delta k/k = 0.5\ sec^{-1}$. Most people would class this as an incredible accident and dismiss it, but it is useful to help orient the program.

Flow reduction does not directly produce a positive reactivity addition. It would not be possible to attain prompt critical due to flow reduction unless the reactor also had a positive temperature coefficient of reactivity, or if the flow dropped sufficiently to induce widespread boiling and the reactor had a strong positive sodium void coefficient. Then one could get first the reactivity effects of the sodium void and shortly thereafter the reactivity changes from fuel meltdown, which may in some cases be positive and others negative. There are other ways of inducing sodium boiling and fuel melting. If, for example, a control-rod malfunction or refueling accident were not sufficiently strong and fast to produce prompt criticality directly, it might nevertheless lead to boiling and melting. In addition, flow blockage can clearly lead to this result in the blocked channels. There is also continued speculation that there may be a way for failure of a single pin to propagate to its neighbors and lead to general boiling and melting. (See Chapter IV.)

One sometimes also considers secondary sources of reactivity increase without defining a consistent sequence of events that can be shown capable of leading to that result. For example, one can assume that a large

gas bubble passes through the core even though it may not be possible to visualize how it could happen. Implosions are another example. If a particular core configuration could, in a weak explosion, result in implosion effects which temporarily dominate the simultaneous disassembly effects, then the weak explosion could be transformed into a very strong one. In treating accidents of these types, the analyst is forced to exercise considerable judgment. If no constraints are imposed, the magnitudes and rates of reactivity increase are limited only by the imagination. For example, if a gas bubble is assumed to enter coherently, say the central half of the core subassemblies, at about the rate of the normal coolant-flow velocity, in a core having a large positive sodium void coefficient, the rate of reactivity increase could again be of the order of $\Delta k/k = 0.5\ sec^{-1}$, comparable to control-rod and subassembly accidents. The maximum for severe sodium boiling accidents might also be of this magnitude. The implosion accidents, however, could be imagined to have potentially much higher rates. It is, in fact, very difficult to devise a configuration of normal fast-reactor composition that can have a net autocatalytic implosion.[87] It is probably not appropriate to require the experimental program to extend itself to such conditions, except perhaps to investigate what configurations, if any, can lead to this effect. We therefore consider the upper limit of interest of reactivity rates to be that associated with the other sources discussed above, i.e., $\Delta k/k \cong 0.5\ sec^{-1}$. All lesser rates are also of interest.

2. Initial Conditions of Material Distribution and Its Rate of Change

The energy release in a super-prompt-critical excursion depends, in addition to the reactivity-insertion rate, on the nature and configuration of the core and blanket materials at the time prompt criticality is passed. With some of the accident sources discussed above, the core would be very nearly in its normal state, either at shutdown conditions or more or less steady operation at power. In others, there are large deviations from the normal condition, and these deviations have a significant effect on the energy release. If the initiating source is sodium boiling or a large gas bubble, then a large part of the core is void of sodium. If the reactivity source is fuel movement following meltdown, then the prompt-critical configuration could conceivably be drastically different from the normal one.

The presence or absence of sodium is important for at least three reasons. It affects the neutron-energy spectrum to the extent that the Doppler effect is usually about one-half to one-third lower with the sodium completely voided.[88] It can affect the pressure-energy relationship (equation of state) needed for a disassembly calculation. Thus, in a mild excursion, the main source of pressure could be sodium vapor pressure as a result of mixing molten fuel and liquid sodium, while in a severe excursion, when the pressure is due to fuel vapor, the pressures achieved are dependent upon the amount of void available into which the fuel can expand.[87] Finally, if the core

is full of sodium, because sodium is much lighter than the fuel there is a tendency for the sodium to be expelled preferentially from the high- to low-pressure regions. This could possibly be a source of additional positive reactivity in a system having a positive sodium void coefficient.

The fuel distribution at prompt critical is also important from other points of view. The pressure-energy relation depends upon the fuel and void densities.[87] If the fuel has melted down and filled the spaces voided by the coolant, the pressure-energy relationship will be greatly different from what it would be with the voids. If, say, 25% of the core melts and reforms into a critical mass in the lower blanket or inlet plenum, this small critical assembly would be only loosely coupled neutronically to the rest of the core. The excursion would then be characteristic of a small core, rather than the original large one. The Doppler effect would be small; the mass of fuel vaporized would be small; the prompt-neutron lifetime would be shorter; the geometry could be very irregular.

B. Analytical Problems and Data Needs

1. When Fuel Elements Are Initially Intact

The excursion conditions are best defined when the excursion is initiated in a core in which the fuel elements are still intact at the time prompt criticality is reached. The two main subcases are (1) sodium absent and (2) sodium present. Consider subcase (1) first. The initiating source may have been sodium boiling. It is not very important what initiated the accident, except insofar as the rate of reactivity insertion and initial power level are concerned. Suppose the rate of reactivity increase is very high, i.e., $\sim \Delta k/k = 0.5\ \mathrm{sec}^{-1}$. The power then rises even more rapidly as the reactivity increases beyond prompt critical. The fuel temperature rises, and the fuel expands and melts. Inertial effects prevent significant axial expansion on the very short time scale of the other events,[89] and the expansion is instead into the void space left by the coolant. It therefore produces very little reactivity effect; but there would typically be a negative Doppler effect due to the increase in temperature. If the expansion is strong, it can bring the reactivity back below prompt critical before excessively high pressures are attained.[87,90,91] Whether this is the case or not, the fuel is vaporized and expands to fill the interior void space completely. The pressure it then exerts disassembles the core, terminating the excursion. Depending upon the success of the Doppler effect in slowing the excursion, the core is left at this point with more or less thermodynamic energy which is ultimately expended by a continued expansion process in which surrounding materials are accelerated outward, deformed, and perhaps ruptured.[92] This latter stage involves many complications, such as radiative heat exchange with a variety of materials, perhaps turbulent mixing and heat exchange with sodium, and propagation and reflection of shock waves in the surroundings.

If the reactivity insertion rate is lower ($\Delta k/k = 0.01\ sec^{-1}$), the behavior is similar, provided the excursion starts from a shutdown condition and the Doppler effect is not too large.[87] With a small insertion rate starting at high power or with a large Doppler effect, the excursion would perhaps still result in some vapor production, but with moderate pressures and essentially no destruction exterior to the core except for effects on components with which the hot fuel comes in contact. At very high rates of reactivity insertion, small Doppler effect, or low initial power, the potential damage is quite severe. The qualitative behavior is similar to that described above, except that the Doppler effect is no longer important. The main shutdown effect is explosive disassembly, and very high pressures are attained. It may not be necessary for the experimental program to extend to investigations of these extremely severe hypothetical accidents.

In subcase (2) when prompt critical is reached with the core still full of sodium, the qualitative behavior can be somewhat different. For the lower insertion rates, whether the sodium-void reactivity coefficient is strongly positive or not is important. If it is, then as the sodium boils and is expelled, it adds to the reactivity insertion; the accident then proceeds more like a no-sodium case with high insertion rate, depending, however, upon how much sodium is still in the core and on its uncertain effect on the pressure-energy relationship (equation of state). If the sodium-void reactivity coefficient is small, then for small insertion rates the time scale of the accident is sufficiently long to permit thermal interactions between fuel and coolant, which could be the dominating influence on the accident behavior. These phenomena are not well understood at present and should be a primary target of an experimental program.

For very high insertion rates with sodium present, the time scale may be too short for extensive fuel-coolant interactions until after the power excursion has been terminated by disassembly (10^{-3} to 10^{-4} sec).[87] But a strong Doppler effect can greatly extend the time scale and produce results not too different from the cases at low insertion rate.[87,91] If the Doppler effect is not large and the time scale is short, the sodium can have two effects during the excursion. First, it occupies space and prevents free expansion of fuel. Thus, the fuel pressure builds up sooner and tends to disassemble the core at lower energy yields than would be the case with voided sodium. Second, in an anisotropic system with high- and low-density components (such as a reactor core), the low-density component has a tendency to be preferentially accelerated outward (along the coolant channels) by the pressure distribution. Thus, while everything is moved outward from the center of the pressure distribution, the composition will also change, becoming deficient in sodium at the center and rich in sodium in the upper and lower regions of the core. This effect is being studied quantitatively to see if it warrants further investigation. Once the excursion is complete, the expansion of the hot mixture of core materials takes place on a time scale much longer (say 0.1 sec) than that of the excursion, because large masses of materials have to be accelerated and moved.[92] There is then ample time

to transfer large amounts of heat from the fuel to other materials well mixed with it, such as sodium and cladding. This can influence the destructiveness associated with the excursion.[93]

2. When Excursion Is Initiated by Meltdown and Core Compaction

This is the classic fast-reactor meltdown accident.[94] (See Chapter VI.) It has, however, a somewhat different character in a large oxide breeder compared to the relatively small fast reactors like EBR-II and FERMI. In the small reactors, one was concerned mainly with meltdowns in which a large fraction of the core participated. In a large reactor such as a 1000-MWe LMFBR, it is possible to have a small part of the core melted and rearranged so as to go prompt critical with only loose neutronic coupling to the remainder of the core. Another possibility is to have most of the core involved, but in almost any conceivable geometry. Other sources of irregular meltdown are fuel-failure propagation and subassembly flow blockage, which may take place anywhere in the core. There are, of course, particular sources of core meltdown which tend to order the meltdown to some extent. For example, if the meltdown is caused by an overpower or flow-failure condition, one expects the first fuel to melt to be that in the subassemblies having the highest power density. The melting would also be nearly axisymmetric, but after initial melting, there would be statistical effects in cladding failure and fuel-coolant interactions that could considerably disorder the core and destroy the symmetry. These effects require experimental investigation.

Once one can specify that either all or a particular part of the core is assembling to prompt criticality, then the excursion behavior can be described qualitatively in similar terms as for an accident with intact fuel elements. In the core meltdown, at least in and near the compacted part of the core there would be expected to be little liquid sodium. The first significant shutdown reactivity feedback would come from the Doppler effect. The Doppler effect of the compacted reactor would usually be reduced significantly from that of the normal core and would have to be evaluated for the configuration under investigation.[88] If the Doppler effect is sufficient to bring the reassembled mass below prompt critical, then it largely determines the energy release, although the pressure generated still disassembles the core.[87,91] If the Doppler effect is smaller (less negative), it still reduces the severity of the excursion, but the reactivity is then reduced below prompt critical, mainly by disassembly. In this case, the pressure-energy relationship (equation of state) is the main influence in determining the energy release. For intermediate Doppler coefficients, the shutdown may be shared about equally by Doppler and disassembly. For a given energy release, the nature and extent of the damage done will depend mainly upon the fraction of the original core in which the energy is deposited. The energy distribution can vary from the normal power distribution in the intact core to the case in which perhaps 50% of the energy is concentrated

in 10% of the core, depending upon the way in which the core melted and reformed. Aside from this, the evaluation of destruction would be similar to that for an excursion in the intact core.

3. The Power Excursion

Although our understanding of the qualitative behavior of the phenomena in a severe power excursion is fairly good, a few aspects of the problem make quantitative evaluations difficult and unreliable. There are also some basic physical data that are not known experimentally and cannot be calculated accurately from theoretical considerations. The purpose of this section is to discuss some of the details of the excursion analysis to clarify which areas are well in hand and which require additional experimentation or proof testing.

The areas considered here are:

a. Reactor kinetics

b. Power, reactivity-worth, and temperature distributions

c. Feedback from the Doppler effect

d. Feedback from material motion--small displacements

e. Coupled neutronics and hydrodynamics with large material displacements

f. Material properties--equation of state and heat capacity

g. Mechanical restraints and their influence on feedback.

a. Reactor Kinetics. We define reactor kinetics to be the discipline that deals with the space and time variation of the power, given the reactivity as a function of time. The calculation of feedback effects that produces the reactivity variations are considered separately. This division cannot actually be accomplished completely, since a unique reactivity is only definable to the extent that the space-time effects are separable. In most instances, the configuration under consideration would be sufficiently regular and symmetric to permit the separability assumption as a good approximation. In this case, our knowledge of reactor kinetics is good enough for practical purposes. If one part of the system goes prompt critical with only loose coupling to the remainder, then the separability assumption may not be good enough. There is, of course, the extreme case in which the coupling can be completely ignored, and this is no problem. In the intermediate situation, it may be necessary to treat the space-time problem to a better approximation.

A number of better approximations exist as working or experimental codes; two examples are, WIGL[95] and QX-1;[96] other codes using different numerical schemes are also being used. By and large,

these codes are currently limited to one-space dimension, although there are many efforts to extend the spatial capability. Two-dimensional spatially dependent kinetics codes and, at lower priority, three-dimensional codes are under development under Subtasks 9-3.5.1 and 9-3.5.2 of the Physics Program Plan.[97-99] These new methods may be sufficiently free of simplifying approximations to require little or no detailed experimental verification. They will be incorporated, as needed, into the accident analysis codes under Tasks 10-2.1 and 10-2.2 of the LMFBR Program Plan. They will probably not be able to handle completely arbitrary fuel configurations and will have some geometrical restrictions. This is not a serious disadvantage, as the analyst cannot in any case specify the detailed fuel rearrangement in a meltdown, but must assume a simplified model. Thus, while the present kinetics codes are not completely adequate, there are methods under development that will be good enough for safety studies and will not require special experimental verification. Some of the experiments that could be done as proof tests on the overall behavior of a prompt critical excursion would give a partial check on the reactor-kinetics codes.

b. Power, Reactivity-worth, and Temperature Distributions. The power and resulting temperature distribution determine the Doppler-reactivity feedback and the pressure distribution, which in turn establish the material movement and its associated reactivity feedback. The estimation of feedback requires also a knowledge of the reactivity-worth distribution of the core and blanket materials.

(1) Initial Distributions. The power and reactivity-worth distributions at the beginning of an accident can usually be accurately determined by a steady-state calculation, provided the geometry under investigation fits into the available codes. There still remains some question about the accuracy of the multigroup calculational procedures near region boundaries when there is a large change in composition across the boundary. The group constants are usually calculated from a many-group fundamental-mode calculation, rather than a spatial calculation. Better methods will be developed in the future.*[97-99] It will be desirable to check the calculations by critical-experiment measurements for typical configurations pertinent to the safety studies.

The initial temperature distributions are well enough defined by the power distributions and steady-state or, in some cases, transient heat-transfer calculations. In most cases, no experimental work is needed. However, if the excursion initially involves sodium boiling (see Chapter III) or fuel meltdown, then the initiating mechanism itself upsets the temperature distribution in a way that is not readily predictable. Our knowledge of transient heat transfer and transient fluid flow in two-phase systems is not adequate. This is an area in which more experimental data

*Methods currently at an advanced state of development include the 1DX code at BNWL, the XSDRN code at ORNL, and similar techniques of generating space-dependent cross sections.

may be needed. On the other hand, the situation has not been studied from the point of view of how much error is introduced into the accident analysis by the uncertainties in the initial temperature distribution. This may not be of great importance.

(2) Changes due to Space-Time Effects. Under some conditions, especially when there is weak coupling between one part of the core and another, the shape of the flux and hence of the power distribution can change with time in a severe excursion, even though the material distribution remains invariant. The material-reactivity worths also change, and, in fact reactivity is no longer a well-defined concept. This has already been discussed under Section 3.a above. Here it is merely stressed that if there are significant space-time effects, they must be accounted for in a determination of the temperature distribution and resultant pressures and material motions.

(3) Transient Heat Transfer. In an extremely short-period prompt-critical excursion (say 10^{-3}-10^{-4} sec period), nearly all the heat stays in the fuel and the pressures are then determined by the fuel equation of state and heat capacity and the volume available for free expansion. If the period is as slow as 10^{-3} sec and the mixing of molten fuel, coolant, and structure is quite good, then there may be some heat transfer. At high temperatures, the heat transfer will be mainly by radiation. Not very many heat-transfer data are directly applicable. The greatest uncertainty in the heat transfer, however, is often not in the heat-transfer calculation, but in the estimates of mixing effects.

It might be possible to do some experiments with initially intact pins out-of-pile using electrical heaters, although even this is not clear. In-pile experiments could be done either as complete core-destruct tests or as single-subassembly tests in a loop. (See Chapter VI.) However, the sensitivity of results to the transient heat transfer must be evaluated before a judgment can be made about the real need for this information.

c. Feedback from the Doppler Effect. In prompt-critical excursions in a large fast reactor (for example, a 3000-liter oxide-fueled reactor), the Doppler effect is the most important single feedback for limiting the magnitude of the excursion. In a very severe case, with a rate of reactivity insertion equal to $\Delta k/k = 0.5\ \text{sec}^{-1}$, the maximum reactivity attained will be of the order of 0.002 above prompt critical.[87] Suppose this reactor has a Doppler temperature coefficient of reactivity equal to $2 \times 10^{-6}/°C$ at a fuel temperature of 2000°K. If we allow that the coefficient will decrease to some extent with temperature, the Doppler effect can still bring the reactivity well below prompt critical with a temperature rise of about 2000°C, thus taking the fuel to 4000°K and terminating the excursion at moderate pressures without impairing the function of the containment structures. The same accident with no Doppler effect would be much more

difficult to contain. If the Doppler effect is very large, it effectively terminates all accidents having reactivity insertion rates up to the assumed limit of $\Delta k/k = 0.5\ sec^{-1}$ without pressures difficult to contain.[87,91] If the Doppler effect is very small, it is of little benefit. At either of these two limits, it is not very important to have an accurate knowledge of the magnitude of the Doppler coefficient. In the intermediate range, however, the consequences of a super-prompt-critical excursion can be quite sensitive to the magnitude of the Doppler coefficient.[87]

The Doppler coefficient for a power reactor is estimated using calculational techniques that are verified by experiments that are not direct measurements of the Doppler coefficient for the whole core. Rather, measurements are made on small samples usually placed in the center of a critical assembly.[100] Nevertheless, the procedure of using the experiment to check the calculational method and then applying it to a different and more complex situation is probably a reliable one. The validity of this approach will be tested by SEFOR experiments.[101]

The main problem with the Doppler effect is that it depends upon the core configuration and composition in the accident situation including the extent of sodium voiding. One must examine the expected conditions with a model that fits existing computer methods and then calculate the Doppler coefficient. Such a model is at best as accurate as our knowledge of sodium voiding and, in some cases, of fuel movements due to fuel-sodium interaction during meltdown. This again points to the need for the sodium-boiling and fuel-meltdown studies. (See Chapter VI.)

d. Feedback from Material Motion--Small Displacements. Although the Doppler effect is of major importance in limiting the energy release and pressure, it can compensate only a limited amount of reactivity insertion. If the reactivity insertion available from the accident source is greater than this, then the ultimate shutdown effect has to be material movement and disassembly. In this case, the Doppler effect tends to prevent the reactor period becoming exceedingly short, so that the disassembly can take place more slowly at moderate pressures and lesser energy input.[87,91] The feedback from material motions, however, is still important.

(1) Feedback from Intact Fuel and Sodium. In some instances, at least part of the core would have solid fuel and the normal liquid during most of the prompt-critical excursion. The behavior of these materials and the resulting feedback depend upon the exponential period of the power rise. If the rise is slow, then the temperatures and material motion due to expansion are simply calculable from transient heat-transfer data and the coefficients of expansion. This can be done only if the reactivity is not much above prompt critical. The sodium effect would usually be small compared to the fuel.

If the exponential period becomes comparable to or less than the time for a sound wave to propagate the length of a fuel pin, then the expansion is restrained by inertial effects,[89] and the equations of motion must be solved. This is further complicated slightly by the temperature dependence of the material properties, especially if the internal stresses eventually exceed the yield strength of the material. Given the material properties, however, the calculation of material displacements is well founded and reliable. Further, reactivity effects can be calculated accurately by perturbation theory, provided the configuration can be well modeled by a geometry acceptable to the multigroup computer codes. Thus, no experimental program, except possibly a small amount of critical assembly work, is contemplated for this aspect of the prompt-critical excursion.

(2) Feedback from Molten and Vaporized Core Materials. It is useful to discuss feedback from small (<1 cm) and from large displacements separately because the problem is much easier for small displacements. If the reactivity goes strongly negative (say -0.01 $\Delta k/k$) while the displacement is small, then it is unnecessary to consider the result of large displacements, unless there is a possibility that they can bring the reactor back again above prompt critical. This can usually, but perhaps not always, be ruled out.

There are two main aspects to the calculation for small displacements: the displacement produced by the pressure distribution, and the reactivity resulting from the displacement. If the core is in a molten or vapor state, the displacements can be calculated from the hydrodynamics equations, either by the approximate method embodied in the MARS code,[102] or by the more accurate treatment of a code now under development at Argonne.[103] The uncertainties involved in these calculations stem mainly from the pressure-distribution uncertainties due to our inadequate knowledge of the equations of state and heat capacities discussed below in Section f. If the pressures are not very high, then the residual core structure can influence the motion, as discussed in Section g. Also, the amount of sodium in the core and the extent of mixing and heat exchange between sodium and fuel are uncertain. If the core configuration is simple (for example, close to the original core geometry with only one core zone), then the reactivity feedback is readily calculable from the material movement by perturbation theory using existing computer programs. No experiments are required.

If the core configuration in the accident is more complex, with interior-void regions or regions of differing fuel density or enrichment, or if there are asymmetrical aspects which cannot be adequately modeled by an axisymmetric model, then it may be necessary to do some critical experiment work to determine the reactivity-worth distributions. The analytical techniques for reactivity calculation are continually improving, and it may, within a few years, be feasible to account for nearly all the axisymmetric complexities without special critical experiments. The ability

to treat asymmetry is somewhat further off in time. However, one must also ask what can be done with asymmetric-reactivity information were it available. No methods are being developed to calculate the hydrodynamics of asymmetric prompt-critical excursions. It is not likely that this will be found necessary. But if it is, then the complete asymmetric-accident problem may require some proof tests consisting of whole core destructive tests.

The other type of complex configuration that could be encountered is that in which a small part of the original core becomes rearranged in a prompt-critical configuration only loosely coupled to the remaining core. This could most easily happen in a very large pancake, annular, or modular core. In these cases, the usual static reactivity distribution may not be adequate and the problem may require treatment by space-time kinetics methods.

e. Coupled Neutronics and Hydrodynamics with Large Material Displacements. As implied in Section d above on small displacements, there can be situations in which the treatment for small displacements is inadequate. This may happen if there is a large material or composition discontinuity in a region where the power density is high. Examples are a large central-void region or a zone boundary in a reactor that has two or more enrichment zones for the purpose of flattening the power distribution. It is not clear that the treatment for small displacements gives a large or even significant error in such cases. Rather the situation is that there are questions about the simpler small-displacement treatment when there are discontinuities. These questions are under continuing investigation, and it must be assumed for the present that some instances will be found in which a more accurate treatment is needed.

When it is necessary to account for large displacements correctly, first the unsimplified form of the hydrodynamics equations must be used with a material equation of state that is explicitly density dependent. Second, the reactivity must be recalculated at selected time points by a complete solution of the transport or diffusion equations rather than by perturbation theory. There are computer codes that treat these effects to a reasonable approximation, and they are under continuing improvement under Task Area 10-2 of the Program Plan.[104] When the core configuration requires the more detailed treatment of coupled neutronics and hydrodynamics for large displacements, then there is also more uncertainty in the accuracy of the calculations. This is especially so if some of the material motions give positive-reactivity effects, so that the net effect is a balance between some positive and some negative components. The reactivity calculation can be partially checked by a series of critical experiments with successive loading changes to simulate the motion of the core during the excursion. This approach clearly has some limitations because we depend upon calculation to specify the material motion. This is an area that conceivably could benefit from a proof test of a complete whole-core destructive-excursion

experiment. A great deal more investigation would be required to determine the full benefit of such an experiment.

f. Material Properties--Equation of State and Heat Capacity. The core materials among which the energy is shared and that contribute to pressure generation during the excursion depend upon the shortness of the reactor period and the degree of mixing. In very fast excursions, the heat will stay mainly in the fuel. At the other extreme, an appreciable amount of energy may get into the sodium. One would not expect the structural material to heat up significantly on the short time scale of a super-prompt-critical excursion unless there were appreciable time for melting and mixing before the excursion. We are, therefore, primarily interested in the fuel materials and sodium.

(1) Fuel Materials. In current fast-reactor design studies, mixed uranium-plutonium oxides are the most popular fuels,[85,86,88] with the mixed carbides generally considered to have long-term potential but requiring more research and development. Although interest in metal fuels has waned in recent years, there is the possibility of a resurgence based upon experience with metal fuels for EBR-II. Thorium-bearing fuels are a very long-range possibility.

There are practically no experimental data on either the equation of state or heat capacity of any of the fuel materials in the high-temperature range of interest.[105] Accident studies to date have used extrapolations of vapor-pressure data and property values derived from the law of corresponding states and other empirical correlations of thermodynamic data. A first step in this procedure is usually an estimate of the critical temperature, pressure, and volume of the material in question. Two or three slightly different approaches can be taken, and one indication that there are large errors is that the different approaches sometimes give differences in the estimates of the critical constants of nearly a factor of two.[105] For the energy-release calculations, what is required is the pressure and temperature expressed as a function of energy content and density. At high densities, pressure is a strong function of energy and an extremely strong function of density, and at any density the pressure is a strong function of energy on the saturation line. Thus, one expects that the theoretical predictions of pressure for given energy input and volume may have very large error, in some instances possibly by a factor of ten. However, for the same reason (i.e., the strong dependence of pressure on energy), these errors in pressure do not produce correspondingly large direct errors in the calculated energy release. Answering the question "How badly do we need experimental data on the equation of state?" requires a study of the dependence of the calculated damage potential on the uncertainties in both the high-density equation of state involved in the power-excursion calculation and the lower-density equation of state involved in the adiabatic expansion of the vaporized core. The latter regime is discussed further in Section 4 below.

One must also consider the extent to which errors compensate for one another in the two parts of the calculation. Thus, although the problems are somewhat different in the high- and low-density regimes, the assumptions made in arriving at pressure-energy expressions should be consistent. Sensitivity studies of this type are being initiated under Task 10-2.1.4 of the LMFBR Program Plan.

There are two kinds of problems involved in making experimental determinations of the equation of state: first, the production of high temperatures and high pressures, and second, making reliable measurements of pressure and temperature. Above 20 kb no container can be made to contain the pressure to make static measurements, and furthermore in small samples at high temperatures the energy would be quickly radiated away. It is necessary to produce and observe the conditions in transient experiments. The smaller the sample, the shorter the time scale of the experiment for a given pressure and temperature. These problems are surveyed more completely in Section 4 below; Appendix C contains a thorough discussion of the equation of state and a summary of the problems in determining the material constants and using the results, as well as a summary of experimental techniques. Although integral in-pile tests may be of some help over a limited range and in special circumstances, their use for more general purposes does not appear to be indicated.

(2) Sodium. Compared to the fuel materials, the equations of state and heat capacity of sodium are relatively well known.[106,107] In view of the uncertainties in the estimates of the heat transferred to the sodium during a super-prompt-critical excursion, it is evident that there is no urgent need for extending the current knowledge of this area. When the state of the art is improved substantially in other areas, there may also be an interest in extending the sodium data.

g. Mechanical Restraints and Their Influence on Feedback. In nearly all calculations of super-prompt-critical excursions to date, the feedback included in the analysis has been Doppler effect and the displacement of the core material under the influence of pressure gradients, but assuming the core is a uniform homogeneous fluid, having no structural strength. This is a good first approximation for severe accidents in which the pressures achieved far exceed the strength of any residual core structure. But in weaker excursions, there may be some influence of the strength of the structure, and in all cases where the excursion is initiated with the fuel-rod structure still intact, there may be effects due to the anisotropy of the rod lattice.

(1) Strength of Residual Structure of Core. The rate of rise of power and pressure is sufficiently great, even in relatively mild prompt-critical excursions, that if there is some strength in the core subassembly structure at the beginning of the excursion, there is no time to

transfer heat to weaken it significantly during the time scale of the excursion. If the pressures do not greatly exceed the crushing strength of the matrix of subassembly cans, the motion of the material under the joint influence of structure and pressure is channeled preferentially along the subassemblies. The net effect is usually to decrease the rate of reduction of reactivity and thereby increase the energy release. The effect could be investigated analytically with some simplified approximations and would not necessarily require an experimental program. If full-scale prompt-critical excursions were carried out, these effects could be included in the test; but it would be difficult to identify the specific influence of the structure on the test results.

(2) Inertial-channeling Effects. In a rapid excursion, the mode of material movement to relieve pressure with minimum inertial resistance is for the rods to expand radially, compressing the sodium, and for the sodium then to move axially along the coolant channels. This is the result of the factor-of-ten or more difference between the fuel density and sodium density. The tendency is resisted to some extent by frictional forces; at the same time, the fuel will also move along the macroscopic pressure gradients. As a result of the sodium channeling, that motion will be less than for an isotropic system. The net effect, besides the reduction in macroscopic movement of the mixture of materials, would be a reduction in sodium content in the high-pressure regions. In systems having large positive sodium void coefficients, this could result in a significant reduction in the net negative feedback and could conceivably even give a net positive feedback for a very short period of time. This is an effect that has not been investigated previously and will be undertaken analytically in future studies. (Recent work is reported in Ref. 93.) The stability of the interface between fuel and coolant must also be investigated. Under certain conditions, the Rayleigh-Taylor instability can cause the interface to break up resulting in mixing of fuel and coolant. If this happens, the channeling effect is reduced. Confirmatory experiments in a reactor test facility would be desirable if the analysis indicates there could be a problem.

4. Destructiveness and Containment

After the neutronic excursion is essentially over, the propagation of the pressure wave through the core and blanket structure is of interest, since this determines how much damage will be done to the blanket structure (which acts as an absorber to the wave) and what the final spatial and temperature state of the core is. As the pressure wave progresses outward from the core, it strikes the blanket material so that part of the wave is transmitted and part of it reflected back into the core. The part that is reflected back into the core can be either a tension wave or a compression wave, depending upon the relative mechanical impedance of the core and blanket materials. If the wave reflects as a compression wave, the core may be partially recompacted. In a typical reactor, as a consequence of

usual radial- and axial-blanket designs, this recompaction might happen to a limited extent in the radial direction, but would not be likely to happen in the axial direction. It is unlikely that any such recompaction will result in a second power excursion.

As the pressure wave moves through the core and blanket, irreversible effects at the front cause it to steepen rapidly and it becomes a sharp-fronted shock wave. This shock wave crushes the blanket structure damaging it for further use, but at the same time the energy absorbed in crushing degrades the wave so that damage to regions outside the blanket is not so severe.

The determination of the response of the reactor vessel and shielding[108,109] to a given pressure pulse is a complex problem. In reality, the excursion is probably asymmetric while all existing theory and experiments are axisymmetric in nature. Thus, to correlate actual excursions with the idealized axisymmetric case, some method of determining the effect of asymmetry is needed. A useful approach is to reverse the question and ask "If we use axisymmetric results, what degree of asymmetry can we tolerate and still predict pressure vessel response to within say 20%?" In this manner, all the tests and theory can be put to good use in studying actual excursions and the response of the reactor vessel.

A considerable amount of small-scale testing has been done in this field at Stanford Research Institute (SRI),[110] Ballistic Research Laboratory (BRL),[111] and Atomic Weapons Research Establishment (AWRE).[112] Work on equivalence laws to utilize the empirical data from these tests in the design of containment structures is being carried out under Task 10-2.5.4 of the LMFBR Program Plan. The major problems in such experiments are simulation of the pressure pulse, and application of equivalence laws to complex structures. Neither of these problems is easily resolved; therefore a combination of basic methods development and proof testing is also required. Basic methods (Task 10-2.5.6)[104] use continuum mechanics and knowledge of the equation of state over a wide range of pressures and temperatures to develop mathematical models suitable for computation. Aside from the obvious difficulties of developing complex computational methods, there is the additional experimental problem of determining material properties and, in particular, the equation of state over a wide range. These problems and also those associated with pressure pulse simulation are discussed next.

a. Equation of State. In Section 3.d(2) above, the equation of state was discussed with reference to its effect in determining the course of a prompt-critical excursion that is terminated by reactor disassembly. The main interest there was in fuel materials at a fairly high density. It was also pointed out that sodium and structural materials may have some influence in weaker explosions. The interests are quite different when

considering the subsequent mechanical damage produced and the problem of containment. For this purpose, one needs, first of all, the equation of state of the fuel materials for a wide range of densities as they expand adiabatically pushing the blanket, thermal shielding, and reactor vessel outward. There is also the possibility of the sodium playing a major role in determining the amount of work that can be done, depending upon the heat transferred to it by radiation and mixing. The buildup, propagation, and attenuation of shock waves in the region surrounding the core is dependent upon the equation of state of the materials in this region. In this case, one is interested in moderate temperatures, pressures up to several kilobars (perhaps in rare cases, as much as 100 kb), and densities near the normal liquid and solid densities. Sometimes it is convenient to define an equation of state for a mixture of materials, rather than attempt to deduce the composite behavior from the individual properties and complex geometry. Thus, experiments are of interest for mixtures in the actual reactor geometry. Appendix C contains a discussion of the various kinds of equations of state needed, the current state of the art, and the kinds of experiments that might be done.

b. Blast Simulation Using Chemical Explosives.[113] Many attempts have been made in the past, using scaling laws and dimensional analysis, to derive a chemical equivalent to a nuclear excursion based upon equating the amount of energy released in an excursion to an equal amount of energy release in the chemical explosive. However, because the pressure-time history of the usual nuclear excursion differs from that of a typical chemical explosion, the estimates of damage were not consistent.

In a review of the application of data from pentolite explosions to the analysis of the SL-1 reactor accident,[113] Proctor found that given the energy partition in the nuclear excursion, an "analytical explosion model can be postulated that is capable of predicting the mechanical damage observed in the SL-1 accident." This predictive ability is based on the correlation tests in which both slow- and fast-burning changes were detonated in water-filled vessels.[114] Although this analysis is ex post facto, it does point out that a rational basis exists for correlating tests with differing sources of pressure. The conclusion of the work presented in Ref. 114 is that more work is needed on energy partition for PWR's. A similar conclusion should be anticipated for LMFBR's.

In a sudden energy release, whether from a chemical or nuclear source, in some medium, three items must be considered:

1. A sharp rise in pressure with time.
2. Adiabatic expansion of the medium once the pressure pulse has passed.
3. Condensation of the hot gas bubble.

To obtain an equivalence between sources of energy release over the entire time period may or may not be necessary, depending upon what phase of the damage one is interested in reproducing. Only item 1, the pressure profile, need be considered in equating damage from the shock wave. On the other hand, to determine how the missiles are formed and what damage they may cause requires that all three items be considered.

To obtain an equivalence based upon a specified pressure rise should not be too difficult, since this can be accomplished by controlling the burning rate of the explosive and/or chemical propellant. Several difficulties must be considered:

1. When the duration of a power burst is scaled, the duration limits may be appreciably longer than the detonation time for most high explosives, at lease for those involving sizes and amounts of explosives consistent with scaled reactor dimensions and energies. Special explosive configurations may be necessary to achieve the desired duration.

2. Propellants and pyrotechnic materials, having much slower rates of burning than high explosives, lend themselves to the simulation of long-duration excursions. They have the disadvantage, however, that the rate of energy release is affected strongly by the degree of pressure confinement.[110]

3. Since both high explosives and slower-burning propellants generate high pressure in gaseous reaction products, these high pressures must be prevented from interfering with a proper functioning of the source of pressure. That is, the pressure-time history depends on the degree of confinement.

4. Because of the need to simulate reactor excursions on a small scale, explosives having low detonation velocities should be considered.

Simulation of the adiabatic expansion of the medium poses a serious problem because the vaporized UO_2 condenses at 3600°C, while the products from a high explosive may remain gaseous even at room temperature. There are two approaches to this problem. As mentioned previously, one approach is to use an explosive whose products are mainly solid or liquid, although this is difficult to accomplish in practice. Another approach is to devise some means for equating the total impulse (after the pressure pulse has decayed) delivered to an object in each case. In this manner, it may be possible to equate damage between an excursion and an explosive, particularly in the case of production of missiles.

The first part of the experimental program needed is to develop explosive materials that produce a large percentage of solid reaction products, high-energy yield, and a low detonation velocity. One way of

achieving this is to make a train of explosive strands coiled in the shape of a helix, which furnishes an efficient way of satisfying space, time, and power requirements from an energy source. Each coil of the train, however, must be protected from the shock from preceding coils. This is done by using metal ribs. The length of practical train on a simple 3-in. spool is restricted to about 7 ft or less. Composite spools (a set of spools fitting inside one another) have been made to accommodate trains 15 ft long. Greater lengths could be achieved if shock strength were reduced. Two commercial products which are available and may be useful are MDF and Pyrocore. Both consist of lead or lead-alloy encasing a reactant core. MDF[110] has detonation velocities of 2700-5900 m/sec and yields 99-117 cal/g depending upon the mixing proportions and strand size. The most important difference between MDF and Pyrocore is their detonation products. The products of MDF are entirely gaseous (even at room temperature), which results in a very high shock pressure. Only a fraction of the detonation products of Pyrocore are gaseous, so that the pressure produced is much lower and the majority of the energy released is in the form of heat.[110] The development of materials with lower detonation velocities is also needed. Stanford Research Institute (SRI) has developed two mixtures called Calpoke and Slowpoke, which may be typical of what can be obtained.[110]

The second part of the experimental work would use the high explosives in such a manner that the gaseous products behind the shock front give a total impulse to a structure that is the same as that given by the adiabatic expansion phase of the nuclear excursion. The experiments would involve very crude pressure-vessel structural models with a blast shield in front of the part of the structure that is to receive the impulse loading. The shock wave from the blast would destroy the shield but leave the pressure vessel intact. This would be accomplished by inserting a gas space between the shield and the required structure. The experiments could be run for a wide range of energy releases to test the validity of the concept. Pressure, impulse, and motion could be measured on the pressure vessel and the explosive in order to establish empirical relationships. The experimental results could then be compared with theoretical calculations to check the accuracy of the method when the pressure loading is known.

Although some development of explosives to simulate nuclear accidents could begin now, attempts to tailor explosives to specific pressure-time relationships, amounts of condensible reaction products, and adiabatic expansion characteristics, should wait until the analytical studies have proceeded a little further so that the desired characteristics are more reliably known.

C. Summary of Analytical Needs

The central goal of the experimental program is to provide the data for an accurate description of the phenomena taking place during an accident and, where possible, to provide verification of the mathematical models.

This is in contrast to the alternative point of view of doing experiments that are proof tests of the safety of a particular reactor design. A severely limited number of the latter type of experiments might also be done, but the sensitivity of results to variations in initial conditions requires an experimental program that develops a more fundamental understanding of the accident behavior. This point of view with respect to mathematical modeling has certain consequences that affect the experimental program.

1. The Geometry of the Model

The geometry of a mathematical model for calculating the consequences of an accident is a major factor in determining the complexity and cost of obtaining a solution. Several years ago, such calculations were limited to one dimension, usually with spherical geometry. Now and in the near future, it is feasible to work in two dimensions, and this would usually be in cylindrical coordinates. Sometime further in the future, it may be practical to solve the complete three-dimensional problem without any symmetry.

There are two important aspects to experimental checks of an axisymmetric model. First is the accuracy of the model for situations that are truly axisymmetric; second is the reliability of simulating an asymmetric situation by some "equivalent" axisymmetric model. One might be interested, for example, in an accident initiated by flow blockage and meltdown of a group of several subassemblies. The group might be located off the central axis, and the problem is then basically asymmetric. Another type of asymmetry occurs when the reactor core is not located on the axis of the reactor vessel, or the reactor vessel itself is not axisymmetric. The important question to answer in regard to the experimental program is "When is it necessary to do asymmetric experiments?" If one never simulates asymmetric situations by symmetric ones, the required number of experiments would be increased considerably and computer programs developed would be rather limited. A better approach is to model the asymmetric situations by symmetric ones, do most of the analysis and experimentation on the symmetric models, and then do only a few asymmetric experiments to prove the reliability of the symmetric modeling. If the experiments refute the modeling procedure, it will be necessary to advance the schedule for developing three-dimensional computational methods in the mathematical models. If this is done without geometrical approximation, one can still use the symmetric experiments to check the physical modeling and assume that there is no real need to check the asymmetric calculation with asymmetric experiments. However, a confirmatory experiment could be done if some authoritative opinion develops that it is necessary.

2. Mathematical Models of the Physical Phenomena

Nearly all the experimental work proposed above is directed toward an understanding of the physical phenomena taking place during and

after a super-prompt-critical excursion so that these phenomena may be described accurately by suitable mathematical expressions. Once this has been accomplished, these expressions would be used in calculations of the behavior in cases for which no experiments have been done. There are possible pitfalls in this approach. In areas such as heat transfer and mixing, it might not be possible to obtain a completely fundamental understanding. Then it would be necessary to rely on empirical correlations of behavior with initial conditions of the experiment. When there are not many data to correlate, one can easily be led astray in this approach. In real reactor accidents there are statistical aspects to the behavior. These arise, for example, from manufacturing tolerances in dimensions, composition, metallurgical structure, etc., and from variations in fuel burnup. The analyst, on the other hand, tends to favor deterministic models in which the result is directly determined by the specified initial conditions.

Some of the statistical effects will also enter the experiments. Here they have the effect of increasing the required number of experiments as it becomes necessary to determine not only the probable result but also the spread in possible results and perhaps even the most unfavorable of the various possible results.

Although we have pointed out that there are certain difficulties in proceeding with mathematical modeling as the ultimate approach to safety analysis, these same difficulties make this the only feasible approach. The alternative of simply doing safety-proof tests on mockups of specific reactor designs is impractical because of the necessity of doing tremendous numbers of experiments on each design to establish the effect of many different initial conditions and to account for the statistical possibilities.

D. Experimental Program Summary

In earlier portions of this chapter, some problems were identified that could be better understood and perhaps resolved by suitable experiments. Some of these can be done out-of-pile, and when this is possible it is usually preferable, either for economic reasons or because of the inherent difficulties of in-pile instrumentation for observing and recording test results. Some of the in-pile tests required are merely steady-state zero-power critical-experiment measurements of power or reactivity-worth distributions. Of the transient or steady-state experiments requiring power production, some might be done in existing facilities such as TREAT or PBF. Others may require a new facility or a modification of existing facilities. In the following discussion, the experiments have been included without detailed examination of facility capabilities.

E. Out-of-pile Program

1. Heat Transfer and Fuel-Coolant Mixing (see also Chapters V and VI)

Experiments are required on the amount of heat exchange between fuel and coolant on a short time scale ($\sim 10^{-3}$ sec). The fuel-temperature range of interest is 3000-10,000°K. Since the heat exchange depends strongly on the degree of mixing of fuel and coolant, the experiments need to simulate this properly. Because of the high electrical conductivity of sodium, there are obvious difficulties associated with producing heat in the fuel by electrical means. The high temperatures and chemical activity of sodium may make it difficult to use chemical reactions as a source of heat. As a result, it may not be possible to carry out realistic tests on these phenomena out-of-pile. It is only recommended that the possibilities be investigated.

This short time scale corresponds to the duration of the power excursion itself. There is also a need for the same kind of heat-transfer information on the somewhat longer time scale (0.1 to 1.0 sec) of the period after the power excursion while the mixture of vapors, liquids, and solids is violently expanding and in some cases rupturing the reactor vessel. One needs to know the energy partition to determine the destructive capability.

The problems in simulating the conditions out-of-pile are similar on the longer time scale. The problems of experimental observation are somewhat different. On the short time scale, one requires very fast response instrumentation, but is dealing with a quasi-static mechanical situation. On the longer time scale, if one actually achieves the temperatures and pressures of interest, observations must be made on a system in violent motion. It may be possible to do some experiments in which the fuel is heated and melted separate from the sodium, and then the two are rapidly and intimately mixed.

2. Equation of State and Heat Capacity

Experiments are required to establish the pressure-energy-volume relationship for fuel materials and mixtures of fuel, coolant, and structure, and also for the material mixtures surrounding the core. The highest priority is for the mixed U-Pu oxide fuels. Data on other materials like the uranium-plutonium carbides, thorium-uranium alloys and compound mixtures, plutonium-uranium alloys, various cermets, and mixtures with structure and coolant may also eventually be needed. In addition to the pressure-energy-volume relationship, the temperature is required to determine the Doppler effect and would also be useful for estimating radiative heat exchange. If the experimental method determines the conventional

PVT equation of state, it is important to have a separate heat-capacity determination to relate the energy content to temperature. For the core the range of interest of the variables is roughly:

T: 2000-10,000°K;

P: 0-100 kb;

V: 1-1000 times the solid specific volume.

Techniques have been developed for measuring such data over parts of the ranges of interest, but the accuracy is not generally good enough. Some improvements in experimental methods are needed. It is possible that some of the data can better be obtained by in-pile measurements. Appendix C contains a complete discussion of these problems.

3. Channeling Effects

The channeling effects discussed earlier might be susceptible to out-of-pile experimental investigation. One could perhaps use exploding wire techniques with bundles of 0.25-in.-diam rods in subassembly geometry and with sodium simulated by water. A measurement of the axial flow of water versus time would indicate the extent of the channeling effect, and the onset of the Rayleigh-Taylor instability could probably be determined. A possible source of ambiguity is the effect of the electrical forces, which would not exist in the reactor-accident situation. A limited in-pile experiment, in which channeling was induced but the damage threshold not crossed, would also be useful.

4. Response of Structures to Shock Waves and the Expanding Vaporized Core Material

To evaluate the propagation and attenuation of a shock wave through a structure having the capacity of exerting shear forces, in addition to compressive and tensile forces, requires more than a knowledge of the isotropic PVE equation of state. This is further complicated if the structure is not even axisymmetric, or if it consists of layers of different materials. Experiments are required that simulate the blanket, thermal shielding, reactor vessels, and biological shielding surrounding a reactor core. They should be of a fundamental nature to test sample models of complex arrangements. The experiments should yield both information on the effect of the shock waves on the materials and the effect of the materials on the attenuation of the shock waves. Experiments are also needed to evaluate potential sources of missiles in particular reactor designs. Fairly detailed mockups are required.

5. Explosive Simulant Studies

To carry out meaningful out-of-pile studies on the response of structures to shock waves (4 above), we must have explosives or propellants that produce shock waves comparable to those expected from reactor excursions. Similarly, we must have a simulant that produces the pressure-volume relationship expected in the expansion of the liquid-vapor mixture of core materials subsequent to the excursion. Thus, experiments are required to develop explosives that detonate at a prescribed pressure. To obtain the proper pressure-volume relationship probably requires the development of an explosive that has condensable reaction products. The goal is not a single explosive but a number of different ones which cover the range of interest in rates, pressures, and pressure-volume relations. But, as mentioned earlier, the tailoring for specific conditions should await further analytical studies.

F. In-pile Program

More often than not, available reactor facilities have built-in limitations that render them unsuitable for the needs of the fast-reactor safety program. Besides inherent physical limitations, there can be programmatic and safety limitations. Many of the experiments relating to prompt-critical excursions require putting the test reactor on a short exponential period. Clearly, if the reactor is being used for other tests, they would have to be removed during this one. Thus, it is necessary for particular excursion tests that the reactor must be devoted exclusively to that test. Besides asking if the test can be accomplished if the instrumentation and controls function properly, it is usually necessary to ask also whether hazardous results can occur if the controls malfunction during the test. There are exceptions to this generalization because some useful critical experiments can be done. In addition, the TREAT and PBF reactors are designed specifically for transients and might possibly be suitable for some of the super-prompt-critical experiment needs. The experiments are described below without a determination of the facilities required to carry them out.

1. Critical Experiments

Critical experiments are required to determine criticality as well as power and reactivity-worth distributions in accident configurations involving complications that make calculations impractical or unreliable. Examples of simple cases are situations with internal voids or discontinuities, such as zone boundaries, in cores that have been zoned to flatten the power distribution. In these simple cases, it may be shown that calculations are adequate and only a few experiments will be required. A more complex situation arises if the configuration of interest is not axisymmetric so that it does not fit an existing computer program. One may then wish to do a few experiments to evaluate the feasibility of replacing the real case

by an "equivalent" axisymmetric model. It is not possible to specify this in any more detail, as it is the kind of situation that might arise in the evaluation of a particular reactor design rather than in a general reactor safety program.

2. Transient Heat Transfer

Experiments are required on transient heat transfer between fuel and coolant. Data are needed both for very short times (10^{-3} to 10^{-4} sec) and longer times up to about 0.1 sec, as explained above. For the prompt-critical excursions, we are interested in fuel temperatures of 2000-10,000°K. The experimental problems become more difficult at higher temperatures, because the pressures can be quite high. Quantitative evaluation of the amount of heat transferred versus time challenges the available techniques in instrumentation and interpretation. The main interest in the heat transfer is for prediction of pressures. Since the transient pressure can probably be measured quite accurately, one could tolerate reasonable errors in the heat-transfer determination. The heat transferred and pressures achieved will depend upon the intimacy in contact between fuel and coolant. Experiments are needed that simulate accidents in which the fuel elements are initially intact and others in which the fuel and coolant have been mixed before the transient.

3. Motion of Fuel and Sodium as a Result of Fuel Melting and Fuel-Coolant Interaction

This topic has been discussed in Chapter VI. It is pertinent to super-prompt-critical excursions as one of several possible sources of reactivity increase.

4. Equation of State and Heat Capacity

A transient facility could be used as a means of generating heat in small samples of fuel to raise the temperature to the 2000-10,000°K range for studies of heat capacity and equation of state. Only the relatively low-pressure part of the equation of state is accessible in small samples, as the high pressures require inertial containment, which above about a few kilobars is only effective for very short times, shorter than the exponential periods that can be attained. The techniques involved in obtaining quantitative and reliable data have not been developed.

5. Channeling Effects

Fuel subassemblies or partial subassemblies (perhaps 19 pins) could be subjected to severe power excursions producing temperatures in excess of 5000°K. The reactor periods required are 10^{-3} to 10^{-4} sec. Observations would be made on the preferential expulsion of sodium and the

onset of the Rayleigh-Taylor instability at the fuel-coolant interfaces, if it occurs. These experiments are closely related to those on fuel-coolant mixing and heat transfer and might be combined with them. Observations with a high-speed neutron hodoscope or the equivalent are nearly a necessity for detailed interpretation. A partial interpretation could be obtained from axial sodium flowrate measurements.

We restate the qualification given earlier that further analysis is being conducted to evaluate the need for these channeling experiments.

6. Integral Core-destruct Tests

The need for proof tests of the integral core-destruct type cannot be justified at this time. Such tests could provide valuable information for confirming the adequacy of the analytical models being developed to estimate the magnitude and destructiveness of super-prompt-critical excursions. However, the determination of need for these tests should be delayed until more information is accumulated from the other experiments and the analytical models have been more fully developed. Integral destruct tests initiated at this time would have a poor chance of proving effective. We discuss here some of the benefits and difficulties of such tests.

None of the other experiments discussed above shed light on the effectiveness of the Doppler effect for terminating excursions. It is not clear that this is necessary. The Doppler reactivity coefficient can be established to sufficient accuracy by calculation and zero-power critical experiments. Once it is known, then there is small uncertainty in evaluating its effect on an excursion. In addition, the SEFOR program is directed toward verifying the effectiveness of the Doppler effect in terminating excursions.[101] But if further demonstration is needed, the integral proof tests would provide it and extend the SEFOR results to much higher destructive temperatures.

The tests of heat exchange; mixing; equation of state and heat capacity; material-displacement feedback; sodium voiding; fuel-coolant interaction; mechanical restraints; response of surrounding structures; buildup, propagation, and dissipation of shock waves; and perhaps even missile production would be checked simultaneously in total core-destruct tests.

There are several difficulties:

a. The cost would be exceedingly high, even if some scheme is devised for recovering most of the large amount of fuel required.

b. The hazards would be difficult to cope with, especially if it is necessary to use plutonium fuel.

c. It would be difficult to extract information on individual phenomena from an integral proof test. Thus, if good agreement is obtained between predicted and experimental gross behavior, one may be satisfied that the individual processes are well enough understood. But if there are significant differences, it may not be possible to pin down the source of the differences.

d. If there is not good agreement between theory and experiment, a large number of tests would be required to evaluate reproducibility and the influence of small changes in initial conditions.

If some particular aspect of the excursion is in question, it may be possible to design integral tests which emphasize that aspect and permit specific interpretation of the results in terms of individual phenomena thus mitigating objection c above. Another alternative, to lessen the cost problem, is to do only integral tests in which damage is limited to the fuel elements. This of course also limits the information attainable.

CHAPTER VIII

POSTBURST PHENOMENA

A. Introduction--Relationship to Reactor Safety

Following a severe accident, the release of radioactivity to the environment depends on a number of processes that can conveniently be considered under the following headings:

1. Ultimate containment of core debris
2. Fission product and plutonium release
3. Sodium fires

Ultimate containment of core debris is a problem if there is no effective way of removing the decay heat following an accident. If the decay heat is not removed, the fuel will melt and eventually attack the containment structure. If this structure can not dissipate the decay heat adequately, the containment can be compromised and radioactive material released. A related problem is the possibility of secondary nuclear criticality as the core melts and slumps.

Fission product and plutonium release is not so clearly related to core safety, but it is a major topic of plant safety. With the exception of such highly specialized questions regarding vented versus nonvented fuel, and failed fuel-element detectors, it is unlikely that further work on fission product and plutonium release will greatly influence LMFBR core design. Largely this is because so much is already known about this topic. On the other hand, plant safety and, hence, plant design rest in crucial ways on data regarding methods for containing fission products and plutonium which may be released from the fuel under normal or near-normal operating conditions. Since much of this data will come from in-pile experiments, we treat this topic briefly in this report.

Sodium fires similarly are not closely related to core design. Large in-pile tests may, however, furnish an opportunity to test models of sodium fires in simulated plants. Such fires could be a means of dispersing radioactive material, and also of pressurizing the containment, possibly causing it to fail.

B. Analytic Problems and Data Needs

1. Ultimate Containment

a. Objectives. If an accident occurs that causes a meltdown of a significant fraction of the core, the molten debris must be prevented

from melting through the containment and reaching the environment. If protection is not provided, the containment barriers may be penetrated while there are large quantities of fission products and plutonium available for dispersal. The basic approach at present is to design the system to maintain the integrity of the primary tank (or cell) after the design basis accident, and to provide an emergency cooling system that continues to operate after the accident.

b. Current Designs. Several designs have suggested means for dealing with the debris from a failed core. The designers of the British Prototype Fast Reactor have considered placing a receptacle under the core region to catch the debris. One method they have considered is to collect the debris in a series of trays. The trays would distribute the debris in such a way that it can be cooled by natural convection of the sodium. The same approach has been used by GE in their conservative core concept under the follow-on studies (Task I report[115]). (The advanced concept that GE chose for the reference design, shown in Fig. 1 uses extensive core instrumentation to effect a reactor shutdown before widespread core damage can occur. As a consequence, engineered safety features such as meltdown trays, containment, and blast shielding are not provided in the advanced design.) In the AI design,[116] a cooled floor is provided in the bottom of the cell (Fig. 2). Other designers have not addressed the problem directly in any detail in published reports, but have indicated that the problem should be considered.

c. Accident Types Defining Design Problems. To examine the problem, we can study the following two extreme types of accidents:

1) Melting of part of the core due to local loss or flow blockage, or partial melting and slumping of the whole core due to loss of flow with delayed or partial scram. No serious secondary criticality or reactivity transient has occurred. The possible core configuration after the accident is as shown in Fig. 3a.

2) Reactivity accident, either following the above or caused by other reasons. Part of the core has vaporized and condensed outside the original core region. The remainder of the core is molten and left behind. Significant damage to the radial blanket has occurred, and vapor explosions may have occurred. The possible core configuration after the accident is shown in Fig. 3b.

These two are representative of the accidents that might be considered in connection with postaccident heat removal and the measures needed to provide ultimate containment.

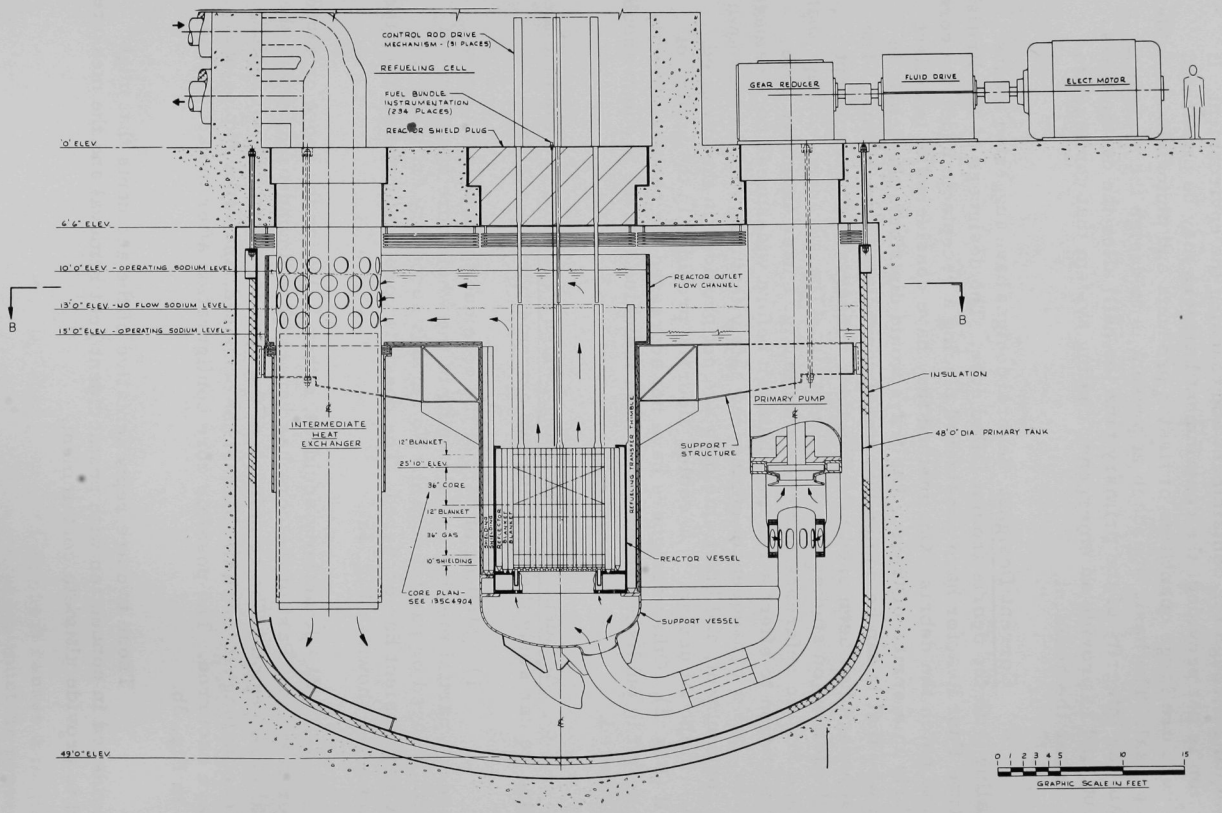

Fig. 1. General Electric 1000-MWe Follow-on Study, Concept of Primary Tank (Vertical Section). ANL Neg. No. 113-3184.

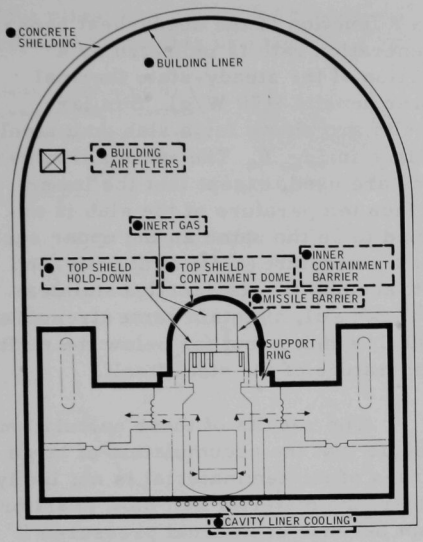

Fig. 2. Atomics International 1000-MWe Follow-on Study, Concept of Primary Containment. ANL Neg. No. 113-3180.

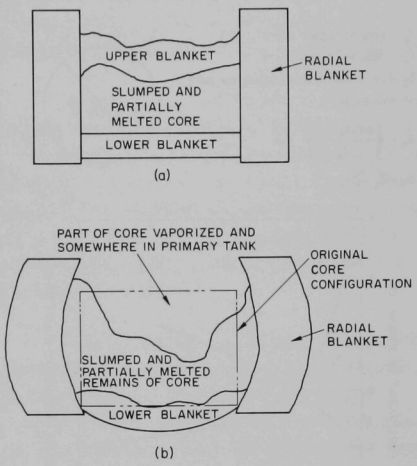

Fig. 3. Slumping Models

d. Data Requirements. Data are required about the behavior of the core materials in sodium. If the system has been designed to maintain the integrity of the primary tank or cell, the molten core debris should remain under the sodium. In sodium, a maximum of 300 W/cm^2 (see Fig. 4) can be removed from a horizontal surface in pool boiling at pressures above 1 atm. This means that sodium boiling can maintain a constant temperature boundary condition on fuel pieces of approximately 1000°C. However, if temperatures in the center of a fuel mass exceed the boiling point of UO_2 (assumed to be ≅4000°C), continual eruption of the molten mass appears likely. The size of pieces that may be stable can be estimated. The radius of a sphere having a center temperature of 4000°C can be easily calculated assuming an average thermal conductivity for the fuel of 0.018 W/cm-°C. Results

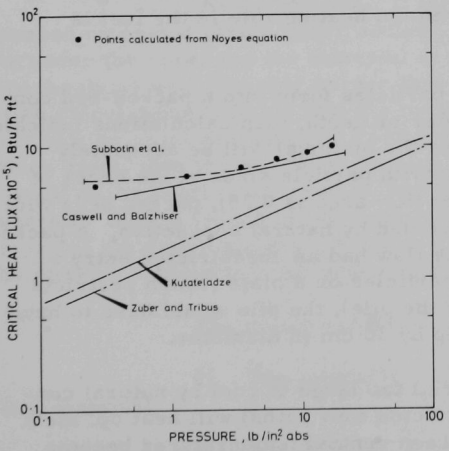

Fig. 4. Experimental Measurements and Theoretical Estimates of the Pool-boiling Critical Heat Flux for Sodium (Fig. 21 of A.E.R.E. R.5436).

are shown in Fig. 5 for sphere sizes as a function of the decay-heat generation. In the figure, the decay-heat generation rate is indicated as a fraction of the steady-state thermal power density (150 W/g). Similar results are shown for a slab on a steel surface in Fig. 6. The same assumptions are used, except that the lower surface temperature of the slab is assumed to be the same as the upper surface temperature (a) for simplicity, and (b) to account for the fact that stainless steel can still maintain some strength at 1000°C, which is not far below the melting temperature of the steel itself.

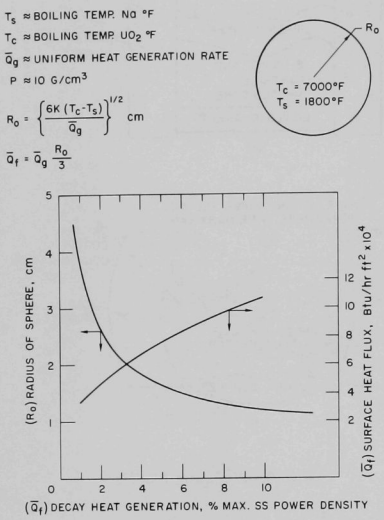

Fig. 5. Stable Sizes of Uniform Spheres as a Function of Decay Heat

The results of these calculations indicate that the accumulation of large masses of molten material is not likely. Rather, the material will blow apart or erupt because of internal pressures developed from high temperatures. These calculations do not take into account internal circulation of the molten material, which would tend to increase sizes. Stable sizes would also increase with time as the decay heating rate decreases. The decay-heat sources may also separate from the fuel pieces into the coolant or cover gas. If this separation is neglected, the maximum heating rate in the fuel is considered.

If collections of the particles form into a packed-bed configuration less than ~30 cm in diameter or depth, then calculations indicate this may be a stable configuration and the material will be adequately cooled. These calculations show that with particle sizes of the order of 1 cm (assuming the fractional coolant flow area is 0.35), packed beds with flow-path lengths of 30 cm could be cooled by natural convection. A packed bed 30 cm deep would be cooled if the flow had an unrestricted entry to the bottom of the bed. For piles of particles on a plate (which restricts entry for coolant flow to the sides of the pile), the pile would have to have smaller dimensions, e.g., 15 cm deep by 30 cm in diameter.

Collections of material too large to cool by natural convection in a packed bed (or with restricted flow paths) will heat up, melt again, and then erupt or blow apart when central temperatures become sufficiently high. During this process, vapor explosions may take place. This action would disperse the material. Pieces that fall back together

into piles than cannot be cooled will remelt, and the process will be repeated. Studies are needed of the dynamic details of these processes of internal circulation and eruption to get accurate estimates of thermal and mechanical effects on the surroundings.

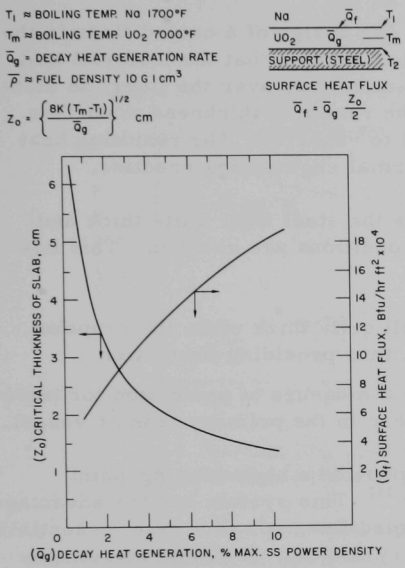

Fig. 6. Stable Sizes of Uniform Slabs as a Function of Decay Heat

When the core meltdown is considered, it is reasonable to assume that the problem of secondary criticality is confined to the core region. Once the material has fallen below the core support plate, natural or engineered dispersion processes can force it into a less reactive form.

Calculations on a fully densified sphere of 20% PuO_2-80% UO_2 indicate that the critical diameter is about 56 cm (22 in.) and that, if the material is in an infinite slab, its thickness must be 16 cm (6.3 in.) for criticality to occur. The previous heat-transfer calculations show that these dimensions are an order of magnitude larger than those of the largest pieces that the effect of the decay heat will allow to cohere.

e. Design Options. If the molten core material can be dispersed on cooled plates under sodium, it can be cooled. If a series of trays is under the core, and the material is caught on them, the natural dispersion mechanisms will probably ensure that thin layers are formed. The previous slab calculations indicate that the maximum thicknesses of layers that will not boil or erupt are between 3 and 5 cm. If the plate material is 0.6-cm-thick steel, then the temperature drop through the plate (maximum of 290°C at full decay heat) is acceptable, and the material will cool by natural convection. Such a series of trays must be designed against the sudden collapse of the grid plate and the entire core on top of it. This can probably by done by placing an open grid structure just above it.

The major problems and unknowns for the tray system are:

1) The interaction of core debris and the tray construction material.

2) The unknown behavior of molten fuel in sodium, which could cause vapor explosions in the tray system with consequent damage.

3) The lack of protection against material dispersed elsewhere in the primary tank.

The second possibility is the design of a cooled floor in the primary tank or cell. In this case, we must assume that the material will eventually distribute itself approximately uniformly over the floor, no matter where it was originally dispersed. The resulting thickness must then be less than that which will cause the fuel to vaporize. The resulting heat fluxes appear to be within the range of normal engineering practice.

One possibility is to make the steel floor quite thick and allow it to melt until quasi-steady-state conditions are reached. This has two advantages:

1) The steel floor is still quite thick while the eruption and dispersal processes are taking place, thus providing protection.

2) The system provides a measure of protection for materials that were initially dispersed elsewhere in the primary tank or vessel.

Another possibility is to provide a high-melting-point insulating material above the cooled floor.[117] This system has the advantage of providing time and protection to the cooled floor. In this case, essentially all the heat is removed through the primary sodium. It should be remembered that 10 min after an excursion the decay heat rate would fall to about 2.5% of the original reactor power level; thus, either a thick floor or an insulating layer provides a significant advantage in time before quasi-steady state conditions are reached. Within 3 hr, the rate would be reduced to 1.0%.

Figure 7 shows the result of a transient heat-transfer calculation for the case in which the UO_2 is assumed molten and dispersed over the bottom of a tank or primary cell. A 6-in. layer of insulating material (with the same thermal conductivity as ThO_2) is assumed to lay between the UO_2 and tank surface. A 2-in. layer of debris with the thermal conductivity of steel is assumed to cover the surface of the molten fuel. Initially, it is assumed that the fuel is at its melting point and the debris is at the melting point of steel.

The results show that if the insulating material remains in place and does not react it provides adequate protection to the steel vessel. They also show that after $1\frac{1}{2}$ hr the fuel has resolidified and is slowly cooling.

Some tests of insulating materials have been conducted.[118] These tests did not cover ThO_2, which might be expected to have some interaction with the molten fuel. However, these tests do indicate that potential insulating materials that are not attacked by the fuel are available.

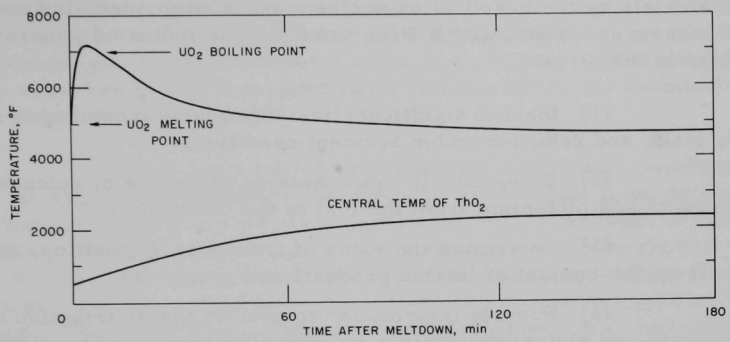

Fig. 7. Maximum UO_2 Temperature as a Function of Time after Meltdown. Material assumed uniformly spread over bottom of 35 in. diam vessel.

The potential hazard of insulating materials is that they will melt and float on the top of the fuel, thereby decreasing heat transfer to the sodium. However, if materials such as ThO_2 and ZrO_2, which form a continuous series of solid solutions with the fuel, are used, this should not be a problem.

Any of the possibilities discussed above, singly or in combination, appear to be reasonable approaches to the problem of postaccident cooling. The major unknown areas are:

1) The interaction of the molten core material with potential structural or insulating materials.

2) The behavior of molten fuel in sodium.

3) The point at which pieces larger than a certain critical size rupture, or erupt, and the nature of the eruption and dispersal.

4) If a flat floor is used, whether the material disperses naturally or whether trays must be used to ensure distribution.

5) The distribution of decay heat sources between fuel, coolant and gas phases.

2. Fission Product and Plutonium Release[119]

a. Objectives. An understanding of the chemical and thermodynamic properties of the fission products, as well as the kinetic processes involved in their transport, is required for predicting the release rates of these materials as the result of an accident and also for devising suitable control measures. The LMFBR Program Plan has indicated general objectives in this area:

(1) Identify significant fission products according to relative hazard, yield, and behavior under accident conditions.

(2) Determine the mechanisms and rates of release of fission products and plutonium from the fuel to the coolant.

(3) Determine the rates of transport, deposition, and release from the coolant of fission products and plutonium.

(4) Provide information to predict the distribution of fission products and plutonium within a reactor plant during a hypothetical accident.

b. Current Status. There is a considerable background of coordinated research, development, and operating experience from which qualitative and semiquantitative predictions of fission-product transport in LMFBR's have been made. Designs of containment systems and engineered safety features have been based largely on these predictions. Also, many of the water-cooled reactor data on and experience with fission-product buildup in fuels, high-efficiency air filtration, treatment of noble gases, and other factors are applicable or adaptable to the control of fission products in LMFBR's. On the other hand, fission-product transport in LMFBR's is significantly different, primarily because of the effect of a sodium environment on the physical and chemical behavior of fission products.

Experimental and theoretical studies on the behavior of fission products in sodium are being conducted at Brookhaven National Laboratory to provide three types of fundamental data: the thermodynamic properties of fission-product-sodium solutions; the extent and rate of release of fission products during vaporization of the sodium, and the chemical states of fission products vaporized from sodium. Experiments on the vaporization of CsI-Na solutions, NaI-Na solutions, and CsI in the absence of sodium have been reported by Castleman, Tang, and Mackay.[120] Under the conditions of the experiments, including concentrations expected in a reactor, CsI was not stable in a sodium environment. Iodine vaporized as NaI from sodium while cesium was transported as the element.

Experiments have been performed at Atomics International[121-128] to evaluate the extent of retention by the sodium, of fission

products released into high-temperature sodium. Sodium temperatures were 260-540°C for in-pile studies and 480-700°C for out-of-pile studies. In the in-pile studies, small disks of preirradiated uranium metal were melted, and a portion of the fission-product inventory was dispersed in molten sodium by a transient neutron burst. In the out-of-pile study, fission products were injected by recoil during the preirradiation and by diffusion during the heating. Data were obtained in both cases by sampling the helium cover-gas space above the sodium and relating the fission-product activity observed to that measured in the sodium and to the estimated inventory of the preirradiated fuel.

When expressed as distribution coefficients, experimental results at Atomics International for the release of iodine from an equilibrium sodium system showed some agreement with a theoretical distribution coefficient. Experimental results for cesium and strontium showed fairly logical distribution coefficients, although theoretical correlation for them was not yet available. Extensive plating and deposition on surfaces were observed. Therefore it was suggested that fission-product retention studies should be related to the effects of configuration and plateout. From SRE studies and other work at Atomics International, it seems that carbon (and perhaps other impurities in the sodium) may be important in the depletion of fission products circulating in the sodium coolant after release from the fuel.

In major accidents postulated for sodium-cooled reactors, transport of fission-product aerosols within the secondary containment system would undoubtedly be associated to a great extent with the transport of sodium oxide aerosols. This is also true of uranium and plutonium from the fuel. Any major breach of the primary coolant-blanket gas system would release sodium vapor, liquid, or both. The sodium, containing ^{24}Na, would be accompanied by fission products accumulated from failed fuel elements or by volatile fission products from vented fuel elements.

The coagulation of aerosols produced by the vaporization of fast-reactor oxide fuel materials and sodium is being studies at Atomics International[129] and at Brookhaven National Laboratory.[130]

c. Additional Work Needed. Additional research and development and extensions of present work are needed in the control and containment of fission products and other radioactive materials in LMFBR plants. Examples of areas that require additional work are the sampling, characterization, agglomeration, and filtration of aerosols produced in the combustion of sodium containing fission products, activation products, and fuel materials; scale-up of the results of small-scale experiments; studies of accident models, fission-product release models, and simulation techniques; further evaluation of vented fuel elements; investigations of transport of fission products in the primary sodium coolant-blanket gas system; and studies of release of fission products from fuel to coolant and blanket gas.

3. Sodium Fires[131-133]

a. Task Objectives. The objective is an analytical description of the release of chemical energy and radioactivity within primary and secondary containment as a result of a sodium fire. The task (see Task 10-2.6 of the LMFBR Program Plan) includes experimental and analytical study of the two major types of fires. Pool fires can occur if large amounts of sodium are discharged to the floor of areas of the containment or into equipment vaults. Spray, or explosive-ejection, fires can occur if a sodium spray is released through cracked or broken pipes or if large amounts of sodium are ejected as a result of a large rupture.

Needed are:

(1) Information required to predict the course and characterize the consequences of sodium pool and spray fires.

(2) An understanding of the formation, settling, and filtering characteristics of sodium oxide aerosols and any associated fission products and fuel.

(3) The basic information required to guide the development of consequence-limiting safety systems and techniques for the cleanup of fuel, fission product, and sodium aerosols that results from a sodium fire.

(4) Mathematical models of pool burning, spray burning, and aerosols behavior for inclusion in sodium fire codes for analysis of hypothetical accidents.

The scope of the task includes basic studies, out-of-pile simulations, and the development of mathematical models of sodium fires. The mathematical models are needed for the development of practical codes for the analysis of hypothetical accidents. The models must describe the following:

(1) The release, transport, and disposition of mass and energy from sodium-pool fires.

(2) The pressure-time history of sodium spray and explosive ejection fires.

(3) The formation, particle-size distribution, agglomeration rate, and settling rate of sodium oxide aerosols formed as a result of the release of sodium, whether or not accompanied by burning.

The need is to develop a reliable understanding of the behavior of sodium fires so that meaningful limits can be placed upon the pressure, temperature, and aerosol burdens which might be placed upon the

containments and consequence-limiting safety systems. Also needed is basic information to guide the development and testing of safety systems and cleanup methods.

b. Current Work. Current work on sodium fires is described in Refs. 131-133. Basically, the technique is to initiate a fire in a chamber whose atmosphere is carefully controlled. Pressure and temperature are monitored and atmosphere samples taken to determine the chemical composition of products and particle-size distributions of aerosols. Both explosive initiation (by rapid heating of a sample representing fuel) and pool fires have been studied. A preliminary calculational model is contained in the SOFIRE code.[133]

C. Program of Work

1. Ultimate Containment

The approach taken here is to list the problems that need to be solved. Associated with each listed problem is a description of the analytical or experimental approach that should be undertaken. The program needed to cover the problem adequately is also outlined.

a. "Molten Core Mass" Containment and Cooling. The design of distributing surfaces and cooled vessel walls or floor, and the choice of suitable materials, are major problems. Concepts of design and candidate materials should be sought in both analytical studies on the meltdown problem and in design studies such as the 1000-MWe Follow-on and those associated with FFTF.

Experimentally, laboratory experiments on a bench scale, i.e., experiments involving gram to kilogram quantities of fuel, should be undertaken to examine materials interactions. The interaction of candidate insulating materials, such as ZrO_2 or ThO_2, with the UO_2-stainless steel mix would be examined in these small-scale experiments. These experiments would examine the high-temperature effects that could occur during the initial period of a few hours when the fuel mass would be at temperatures near or perhaps above melting.

Larger-scale experiments could then be performed with cooled surfaces, insulated with selected materials (the selection based on the above efforts), to examine the longer-term effects as well as providing checks on heat transfer and interaction models used in design and analysis. Effects such as thermal shock, and structural strength would be examined in these tests, which probably would be conducted in-pile.

b. Secondary Energy Release Associated with Reassembly--Analysis of Material Distribution and Criticality. The problem of critical reassembly of the core after the initial meltdown or excursion is another major problem. An early analytical effort should be carried out with the objective of realistically assessing the configuration the fuel might adopt, and the possibility of criticality. This should be performed in the framework of an analysis of "real" accident situations.

In the analytical study, reassembly in the original core position would be assessed; e.g., what radial expansion is needed in a pancake type core to allow complete collapse to 100% fuel density without criticality?

Such as assessment would also look at various sizes and shapes that might collect in lower regions of the vessel or cell. This part of the analysis would take account of the possible geometry and disposition of debris distribution surfaces below the core.

These analytical guidance studies would lead to a longer-range study, developing or making use of computer codes, such as CHEMLOC-II that describes fuel movement and core collapse. The approach would be to construct codes with arbitrary parameters, as has been done in the CHEMLOC code, to examine the effects of various models of core movement (i.e., fluid, plastic, etc.) on reassembly problems.

c. Distribution and Behavior of Molten Core Masses in Sodium. Initial analyses of core meltdown can be carried out by assuming that equilibrium conditions exist or by assuming that the molten mass is a homogeneous mixture of core materials. However, it will be necessary to examine experimentally the behavior and properties of the real mixtures.

Small-scale experiments should be conducted in which core materials (mixed oxide fuel clad in stainless steel) are melted and held at high temperatures for varying periods (up to several hours) both in sodium vapor and under sodium. In-pile techniques may be needed for the latter condition. Such experiments would be performed to identify components, the degree of segregation of various components, melting temperatures of mixtures, etc.

Larger-scale tests, probably conducted in-pile, would also be needed to examine the character of the cooling process at the "molten core mass"/sodium interface. The eruption process (possibly violent) that will occur between collected piles or layers of the "molten core mass" needs to be examined. This process will probably provide a mechanism for efficiently cooling the mass during the early stages. It will also provide a mechanism for redistributing or spreading out the core materials on available cooling areas.

d. Secondary Energy Releases Associated with Thermal Interaction between Molten Core Mass and Sodium. There is a considerable effort in the LMFBR program studying the problem of energy release from molten fuel-sodium interactions. This is being carried out as part of the sodium-boiling studies and the in-pile transient tests (see Chapters V, VI, and VII). The major need in examining the "molten core mass" problem is providing for such events in the analytical studies.

e. Distribution of Decay-heat Sources. During the core meltdown, some of the decay-heat sources will separate from the "molten mass." Based on physical and chemical equilibrium considerations and estimates of elemental sources and their abundance, a large fraction (>50%) of the decay-heat generation could be lost from the molten core mass. Under realistic conditions, mass-transfer processes proceed rapidly enough to provide a large separation of heat producers from the molten mass, and could significantly lower the thermal burden on a cooled surface.

The decay-heat sources lost from the "mass," however, would then be in the sodium coolant or cover gas. Decay heat sources could "plate out" on cooled surfaces of the emergency cooling system, and information as to the species, the quantities involved, and their behavior would be needed for design of these systems.

Initial analytical efforts should involve the identification of decay-heat sources from fast fission with mixed-oxide fuels. The analyses should consider their chemical and physical state based on known equilibrium data. The analyses should include an assessment of the separation and mass-transfer processes.

To provide reasonable data on the mass-transfer processes which affect the efficiency of the separation, small-scale experiments will be needed. These would most likely be conducted in-pile. Typical fuel samples (preirradiated) would be melted in a sodium pool, and the fate of decay heat sources examined.

f. Scaled Integral Tests. As a test of analytical and design capabilities, it would be necessary to conduct meltdown experiments on a larger scale. An integral test program may be necessary with a single assembly or perhaps several subassemblies. Considerable confidence could be gained and scale effects could be checked by carrying out an integral simulation experiment with a single multipin assembly.

One approach might be to allow a preirradiated bundle of rods to melt down over a pool of sodium in an out-of-pile facility.

2. Fission Product and Plutonium Release

Investigations of fission-product release would probably not benefit significantly from in-pile studies. Although some TREAT type experiments may be useful, out-of-pile experiments should provide most of the necessary data: i.e., solubility in coolant over the temperature ranges expected (varying from normal operating conditions to the accident situation); thermodynamic data and vapor pressure information for the various fission-product and fuel species possible under postburst conditions; corrosion interaction of dissolved fission products and cladding; interaction of fission products in their appropriate condensed phase with containment materials (to help estimate what effects could result from decay heating after plate-out). In addition to these studies, kinetic investigations are necessary to elucidate the mechanism and rates of transport of fission products and plutonium from the failed fuel to the coolant and from the coolant to the surrounding reactor system. Techniques for control of fission-product and plutonium transport and their plateout should be a fertile field for study.

More experimental efforts in these areas will be required to provide the basic data needed for predicting release rates and for devising control measures. However, most of the work necessary can be performed out-of-pile, and the large-scale in-pile experiments, which would more realistically simulate fuel assemblies, sodium flowrates, and quantities of materials released, can be carried out as proof tests for the prediction methods developed in the out-of-pile studies.

3. Sodium Fires

There will probably not be any substantial need for in-pile work on sodium fires, except possibly for proof tests that might best be carried out in a TREAT type facility.

Large-scale in-pile tests should furnish an opportunity to assess more realistically the possibility of conditions necessary to initiate a sodium fire.

The current program described above, plus the use of in-pile tests as proof tests, should resolve this problem area.

CHAPTER IX

INSTRUMENTATION

A. Introduction

The in-core instrumentation associated with the LMFBR safety program can be divided into two categories:

1. Instrumentation in support of in-pile safety experiments
2. Instrumentation for core protective systems

B. Instrumentation in Support of In-pile Safety Experiments

The instrumentation and control portion of the LMFBR Program Plan adequately describes the state-of-the-art and outlines the development work that remains to be done for sensors, components, circuits, and systems in support of the safety program,[134] with the exception of the measurement of fuel motion.

As a means of understanding the mechanisms controlling the magnitude of nuclear excursions and the reactor kinetic behavior, measurements must be made of reactivity, energy, pressure, temperature, and the propagation of damage. The information obtained from these measurements can be used to evaluate the effects of thermal expansion, fission-gas release, and disassembly on the course of an accident. Displacement, strain, and acceleration measurements can provide information concerning the mechanical-energy release and the transport of radioactive materials.

If the accident extends far enough so that large parts of the core melt, the sodium-fuel interaction is a potential source of energy that may be comparable to the nuclear source and may contribute to the radiological and physical hazard. Experimental measurement procedures must be designed to give a knowledge of the relative magnitude of these secondary reactions and whether they are important in contributing to the overall destructive effect.

Additional information concerning damage and disassembly can be obtained during posttest measurement and cleanup. Reactor components and support structures can be analyzed in hot cells to determine radiation and physical damage. Experimental instrumentation must be checked to determine the damage to the detectors and electronic equipment and to their supports and protective housings if they are to be salvaged and used in additional experiments. Information gained from such efforts can be applied to improving future experimental measurements and decontamination techniques.

In general, the sensors developed in support of safety experiments can be used as sensors for core protective systems, but core protective sensors, cables, and connectors require additional development to guarantee acceptable reliability over the required life of the sensor. Also, sensor installation and replacement as part of routine fuel-handling operations must be considered.

The program plan includes a comprehensive list of references and a bibliography that documents prior work. Therefore, no attempt has been made to duplicate this material in all its detail, but a brief review of TREAT, SEFOR, EBR-II, and some of the latest developments in safety instrumentation are described and areas requiring future development are indicated. Experience from these facilities is compared in turn for each variable of interest.

Following this a review is made of problems of testing core protective systems for adequate functioning and reliability.

1. Temperature Measurements

In the report by Popper and Knox,[135] and in the state-of-the-art study,[136] current practice and problems in thermocouple use are summarized.

a. TREAT Experience. Thermocouples have been relied upon for temperature measurements, although there are difficulties in their use. Because of the small size of typical fast-reactor fuel pins, difficulties arise, for fuel pin instrumentation, in

(1) Obtaining good thermal contact with fuel.

(2) Avoiding severe perturbation to fuel power and temperature distribution.

(3) Obtaining fast thermal time response.

(4) Avoiding interference with fuel motions.

(5) Handling the fragile thermocouples developed for this service.

(1) Fuel Temperature. Techniques were developed, in collaboration with Advanced Technology Laboratories[137] under Argonne National Laboratory subcontract, for casting a miniature refractory metal thermocouple into a 0.366-cm-diam EBR-II fuel pin and then bonding and jacketing the pin for experimentation in TREAT capsule or loop. The thermocouple consists of a 0.114-cm-diam, tantalum-sheathed unit,* containing

*Other promising sheath materials include tungsten-rhenium and molybdenum-rhenium alloys.

0.013-cm-diam tantalum and molybdenum wires that are cast into direct contact with the fuel alloy to provide a minimum response time. This device has been used in a limited number of experiments on meltdown of dry elements,[138] as well as in one stagnant sodium capsule experiment,[139] and in one sodium loop sample.[81] In addition, some interesting results were obtained in the stagnant sodium experiments using a 0.10-cm-diam stainless steel-sheathed thermocouple located in the bond sodium between the fuel and the cladding, inside a small slot milled along the surface of the fuel cylinder. In this position, the thermocouple sheath occupied a position--for transient heating--roughly analogous to that of the steel cladding of the fuel pin. Thus, transient temperature measurements from the sheath-grounded junction of this thermocouple provided an "analog" of cladding temperatures. Because of the experimental difficulties in fabricating, handling, and assembling into capsules or loops, and (more important) the fact that these thermocouples are actually undesirable for general use because they can interfere with fuel motion, these fuel thermocouples were relied upon principally to validate calculations of transient temperatures.[81,139]

Figures 8 and 9 compare calculated and measured central fuel temperatures for stagnant sodium[139] and flowing sodium.[81] Figure 8 gives evidence of thermocouple detection of a fuel specific-heat change which can be related to the beginning of the melting range.

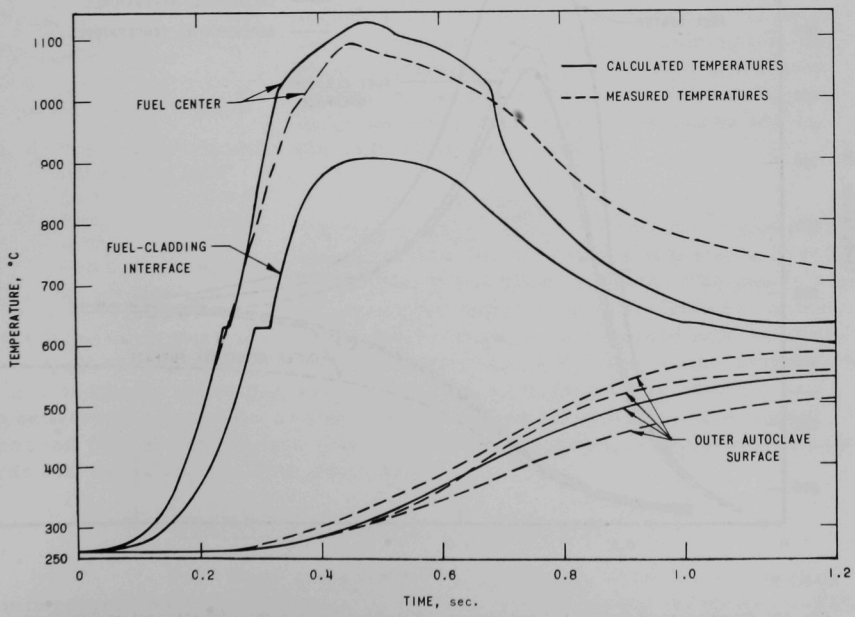

Fig. 8. Comparison of Calculated and Measured Transient Temperatures for Stagnant-sodium on EBR-II Pin for Transient 334. ANL Neg. No. 112-2781.

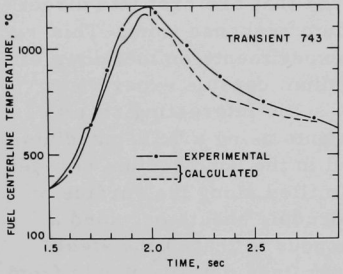

Fig. 9. Comparison of Calculated and Measured Central Fuel Temperatures for Mark-I Loop Experiment on EBR-II Pin for Transient 743. ANL Neg. No. 113-3181.

Figure 10 shows calculated and experimental temperatures from the "analog measurement" of sodium bond temperatures in an EBR-II pin contained in a stagnant-sodium capsule.[140] Again, agreement is encouraging, as a basis for general reliance upon calculations.

Attempts have been made to measure central oxide fuel temperatures during TREAT transients. These have not been successful. W-5% Re/W-26% Re thoria-insulated thermocouples, inside 0.15-cm-OD tantalum sheaths, were inserted, spring-loaded, into holes in two UO_2 test specimens. In both cases, the thermocouples showed unacceptably slow response and severe perturbation of the maximum temperatures.[53] Similar results were obtained in a related study performed to check the feasibility of a method in which

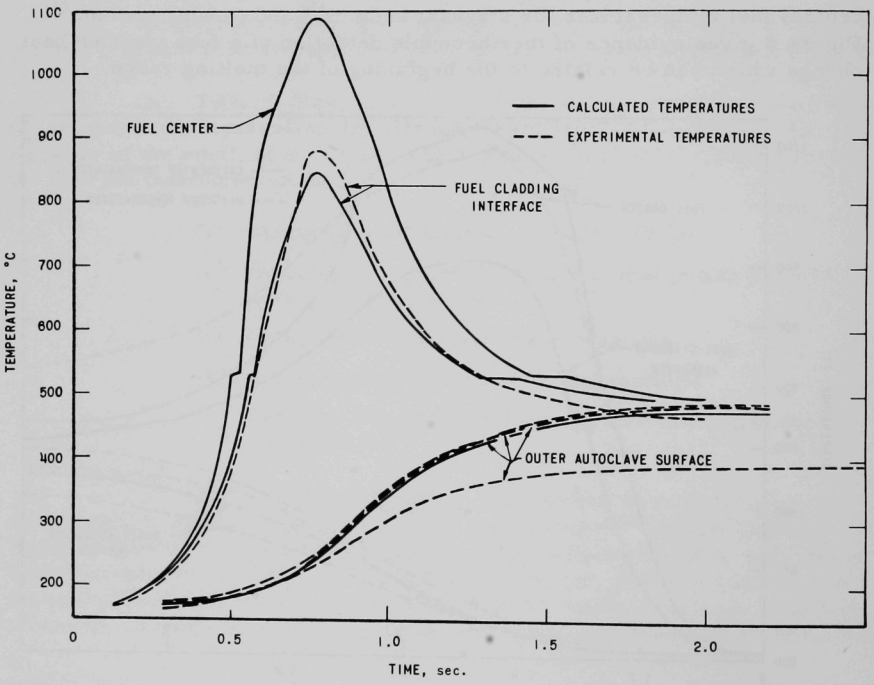

Fig. 10. Comparison of Calculated and Measured Transient Temperatures for Stagnant-sodium Capsule Experiment on EBR-II Pin for Transient 280. ANL Neg. No. 112-2782 Rev. 1.

TREAT is used to measure oxide thermal conductivity at high temperatures.[141] An unsuccessful attempt was made to sinter bare tungsten-rhenium wires directly into individual UO_2 pellets. The wires were sintered in place, but were too brittle to survive careful handling of the pellets.

A useful technique developed in conjection with Power Burst Facility (PBF) instrumentation is the "fission thermocouple." This is a small sphere of oxide fuel with a thin layer of tungsten. This device is described more fully in Ref. 142.

(2) Fuel-cladding Temperature. Cladding thermocouples, with each wire welded separately to the cladding to provide minimum response time, have been used extensively in the experiments on samples contained in an inert-gas environment.[143] Couples used successfully include Chromel/Alumel, Pt/Pt-10% Rh for higher temperatures, and W-5% Re/W-26% Re for temperatures up to ~2600°C.[143-145]

(3) Coolant Temperature. Thermocouple measurements of transient coolant temperatures have been made using stainless-steel-sheathed thermocouples of 0.10-cm OD. Figure 11 compares thermocouple response at the outlet of the test section of a loop experiment in which a single active pin was run short of failure inside a ring of six dummy pins.[81] Although response times as short as ~23 msec can be calculated for such thermocouples, the sheath dimensions are comparable to coolant channel dimensions and can produce very large perturbations in local temperatures for the case of stagnant sodium.

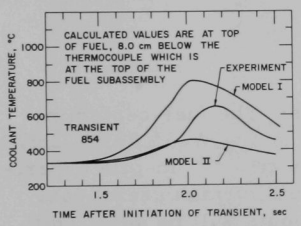

Fig. 11. Comparison of Calculated and Measured Outlet Coolant Temperatures for Mark I-Loop Experiment on EBR-II Pin for Transient 854. ANL Neg. No. 112-7365.

(4) Techniques Other Than Thermocouples. Other techniques of measuring temperatures during TREAT transients have been investigated, but have received little use in actual safety experiments. These techniques include internal temperature measurements by continuous measurement of ultrasonic pulse transit times, measurement of local surface temperatures using a fast-response recording pyrometer with a telephoto lens adapter, and measurement of general surface temperatures by means of high-speed infrared film in conjunction with continuous recording of temperature standards and densitometer film recording.

b. SEFOR Instrumented Fuel Assembly

(1) Fuel-centerline Temperatures. The fuel-centerline temperatures will be monitored by thermocouples having W-5% Re/W-26% Re

thermoelements and sheathing of W-26% Re. Some of the thermocouples will use BeO insulators, and the remainder will use ThO_2. (High-purity hafnium oxide appears to be a promising insulator as well.) Three different junction sizes will be used:

0.356-cm-OD sheath with 0.025-cm wires
0.160-cm-OD sheath with 0.013-cm wires
Bare 0.025-cm wires

The W-5% Re/W-26% Re thermocouple was selected over W-3% Re/W-25% Re because of the availability of a ductile, compensated lead wire that matches the emf-versus-temperature characteristics of W-5% Re/W-26% Re from 0 to 650°C. The bare-wire thermocouple has approximately a 100-msec time response, and the sheathed thermocouples have a response time of ~3 sec.

(2) Coolant Temperatures. The coolant thermocouples will consist of Chromel/Alumel junctions (0.030-cm-diam wire), MgO insulation, and Inconel sheathing (0.157-cm OD and 0.023-cm wall thickness). The time constant is less than 0.1 sec.

c. EBR-II Instrumented Subassembly

(1) Fuel-centerline Temperatures. The fuel-centerline temperatures will be monitored by thermocouples having W-3% Re/W-25% Re thermoelements, thoria insulation, and tantalum sheaths.[146] In-pile experience with this type of assembly is limited, but should furnish useful data for fuel temperatures below 2400°C. Fuel enrichments will be adjusted to achieve centerline temperatures in the range 1800-2400°C. Figure 12a shows the proposed method of installing the thermocouples.

(2) Fuel-cladding Temperatures. The fuel-cladding temperatures will be monitored by four 0.159-cm-diam thermocouples of swaged construction, with Chromel/Alumel thermoelements, alumina insulation, and Type 304 stainless steel sheaths. As shown in Fig. 12b, the lower portion of each thermocouple will be swaged to a 0.102-cm diameter, attached to the wall of, and replacing the spacer wire (above the hot junction) around, each capsule instrumented for fuel-centerline temperature measurements. The upper portion of the thermocouples will lead from the top of the capsules to junctions with flexible lead wires above the bulkhead.

(3) Coolant Temperatures. All coolant thermocouples will be of swaged construction, with Chromel/Alumel thermoelements, alumina insulation, and Type 304 stainless steel sheaths. This type of thermocouple has performed reliably in environments similar to those of the EBR-II core region.

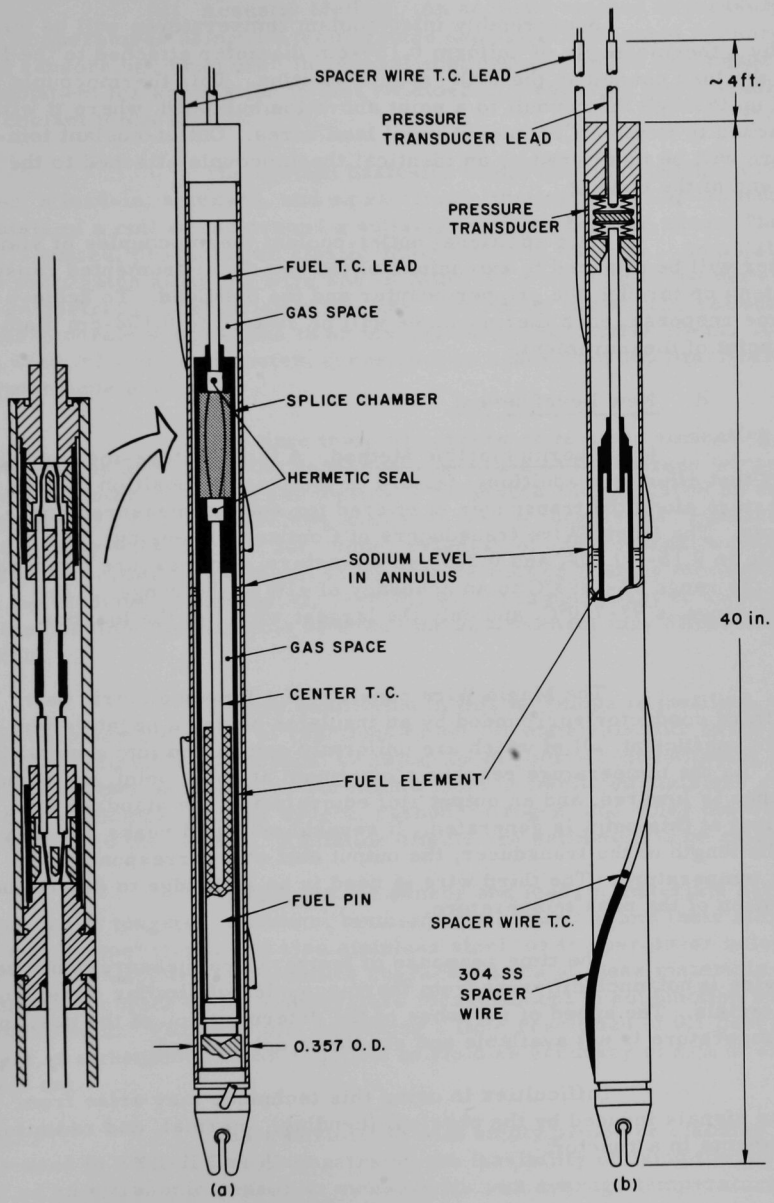

Fig. 12. Installation of Fuel and Cladding Thermocouples and Fission-gas Pressure Transducer in Irradiation Test Capsule. ANL Neg. No. 113-2516.

Subassembly inlet-coolant temperatures will be monitored by a thermocouple of uniform 0.159-cm diameter attached to the lower end of the lead conduit in the 19th capsule position. This thermocouple will extend up through the conduit to a point above the bulkhead, where it will be attached to flexible Chromel/Alumel lead wires. Outlet-coolant temperature will be monitored by an identical thermocouple attached to the upper end of the conduit.

Two additional outlet-coolant thermocouples of similar diameter will be attached to extensions of the two noninstrumented capsules and extend up through the gripper adapter and the bulkhead. To achieve a fast time response, each thermocouple will be swaged to 0.102-cm diameter at the point of measurement.

d. New Development

(1) Thermoelectric Method. A temperature-measuring product that offers the additional feature of locating the position of peak temperature along the transducer is offered for special measurement problems. The Magic Wire transducers of Continental Sensing, Inc., available in 0.15-, 0.20-, and 0.225-cm diameters, can measure temperature in the range 93-1093°C to an accuracy of ±1% of the range. The location range is 93-760°C, and only the largest wire has the location feature.

The Magic Wire consists of a thermoelectric pair, plus a third conductor surrounded by an insulator having a negative temperature coefficient, all of which are uniformly compacted into a metallic sheath. As the temperature reaches a maximum at some point, the shunt impedance is lowered, and an output emf equivalent to the standard millivolt output at this point is generated. If several nonequal peaks are located along the length of the transducer, the output emf will correspond to the highest temperature. The third wire is used in an ac bridge to determine the location of the peak temperature.

The time response of temperature measurements using this device is not much different from thermocouples of similar dimensions and materials. The speed of response of the determination of the position of peak temperature is not available and should be determined.

Difficulties in using this technique may arise from spurious signals induced by the physical (bending), thermal, and radiation environments in a reactor.

(2) Acoustic Method. An acoustic method of measuring high temperatures using the temperature dependence of sound velocity in solid sensors has been used in several applications, and is now under development for application to fast reactors in measuring fuel, cladding, and coolant temperatures.[147,148]

The system basically comprises four parts: a transducer, a lead-in, a sensor, and an electronic instrument. The transducer consists of a coil wound around a suitable magnetostrictive wire. The coil is pulsed, producing an elastic deformation in the wire. An axial wave then propagates along the wire and is reflected according to the materials and geometric configuration of the line. On return to the coil, reflected waves generate a signal due to an inverse magnetostrictive effect. A single thin wire, ~1 mm in diameter, occupies less space than any practical thermocouple pair.

The time interval between reflections emanating from the beginning and end of the sensor is a measure of its average temperature. This time interval can be automatically measured and displayed on an oscillograph or a commercially available device such as the "Panatherm" produced by Panametrics, Inc. Since extensional and torsional waves propagate at velocities proportional to the square root of Young's modulus, E, and the modulus of rigidity, G, respectively, it is necessary to consider all the possible mechanisms by which the environment may affect these moduli.

Before confidence in this technique is justified, we must consider the effect of reactions of sensor materials with oxide or carbide fuel materials, absorption or rejection of poorly bound alloying agents such as carbon, nitrogen, or chromium from unstabilized stainless steels used as claddings, and the oxygen, carbon, nitrogen, and other dissolved impurities in the coolant. Radiation effects are estimated to be small.

Candidates for sensor and lead-in materials have been narrowed to tungsten, rhenium, tantalum, molybdenum, and their alloys for high temperatures and also stainless steel for temperatures below 870°C. The velocity-temperature characteristics of these materials determine the accuracy of the temperature measurement in conjunction with the sensor length. For example, assuming a time resolution of 0.1 μsec, the length of a rhenium sensor required to yield an accuracy of ±1% at 2500°C is 2.5 cm.

Tests performed with empty prototype cladding of the type used in EBR-II fuel demonstrated the feasibility of using the tube itself as an ultrasonic sensor to measure its own average temperature. The effects of the fuel and sodium bond, however, remain to be investigated.

Further, the use of the spiral spacing wire as a sensor was also demonstrated for possible use in measuring coolant temperature external to the cladding. Sodium immersion tests performed at Argonne National Laboratory demonstrated that the ultrasonic thermometer can work in sodium. Also, a rhenium sensor was tested at temperatures up to 2760°C.

This application of ultrasonics shows great promise, but much additional development is necessary to realize the potential. The time response is about 100 msec, but this can be decreased by using larger sampling frequencies. Problems arise from spurious echos due to kinks and fastenings.

Applications in the water-reactor safety program are being studied at Idaho Nuclear Corp.

2. Determination of Material Position

a. TREAT Experience

(1) High-speed Cameras. Extensive use has been made of high-speed color film in experiments on samples contained in an inert-gas environment.[149,150] Speeds up to 4000 frames/sec have been used with commercial color film with high-speed pitch. (Higher speeds are possible with the existing equipment, if black and white film is substituted, but the color offers much better contrast and has been the standard.) With the optical camera system, it has been possible to observe changes in physical shape of the sample, onset of hot spots, condition of the sample at the time of failure, timing and modes of failure, and motions in general. For sodium-bonded pins, unfortunately, the release of bond sodium obscures subsequent events. Even in this case, though, the initial velocity of the sodium cloud can be measured and related through calculations to estimates of internal gas pressure at the time of failure.

(2) Fast-neutron Hodoscope. See Appendix B.

(3) Other Devices. Linear-motion transducers, with which movements are detected by coupling the sample to a slug inside a differential transformer coil, have been used to measure the transient expansion to bare fuel cylinders in TREAT.[151]

An eddy-current device was developed to detect motion inside an EBR-II fuel pin by impedance changes in a coil surrounding the pin. Because of calibration difficulties (including high sensitivity to fuel and sodium resistance changes due to heating), the use was limited to detection of the time that the fuel-sodium interface passed through the coil

plane of symmetry. A different type of electromagnetic detector has been adapted, as part of the related EBR-II driver fuel TREAT program, for actual measurement of the fuel-rod end motion.

Improvements in the use of eddy-current devices have been reported by Idaho Nuclear Corporation in Instrumentation Development Branch Annual Report, IDO-17299 (July 1969), pp. 1-6.

b. New Developments. Because of the opaqueness of the sodium and structural material to visible and near-visible electromagnetic radiation and the presence of radiation in the X-ray region during reactor operation, these portions of the frequency spectrum are not available for following material motion. High-frequency sound waves, however, have been applied in development programs at Atomic Power Development Associates and Pacific Northwest Laboratory and have shown promise.

(1) Acoustic Imaging. Acoustic radiation can be used to form images in much the same manner as other forms of radiation, subject to the attenuation and scattering properties of the transmitting media. Two basic types of systems exist.

In the first, objects are examined by the acoustic shadows they cast when in the line of an acoustic beam. This generally requires the acoustic receiver to be located apart from the acoustic transmitter; the mechanical equipment required to align the transmitter and receivers becomes more complicated, and application presupposes that the location and size of the object are known approximately. Such a system is limited in application in searching for foreign objects or missing pieces. In general, this system will work best when the specific application is known in advance and the system is installed as part of the reactor.

A second type of device depends on the formation of an image from reflected sound waves. It is generally most convenient (but not essential) to combine the transmitter and receiver in a single device (usually a piezoelectric crystal), and this has been demonstrated to be practical. This unitized concept can be operated using the time delay between an outgoing signal and its reflections by displaying intensity versus time on an oscilloscope or similar device. This mode is called the A-scan mode.

Alternately, the acoustic-system signal display can be given an additional dimension by means of line-by-line scanning in what is called the C-scan mode. The intensity can be displayed on a television screen using electronic acoustical imaging techniques similar to those developed for radar or sonar. This mode works best for objects that are normal or approximately normal to the beam. By appropriate circuitry,

the signal-to-noise ratio can be improved by: (1) eliminating weak signals corresponding to spurious reflections and (2) controlling the time delay to restrict the response to signals from a selected distance.

Limitations inherent in the acoustic techniques are the rapid attenuation of the high-frequency signals (up to 10 MHz) necessary to produce good resolution. At frequencies of 1 to 3 MHz, beams can be transmitted up to 300 cm in 315°C sodium; at 10 MHz, the useful range is about 30 cm.

Barring inhomogeneities in the transmission through the sodium due to turbulence and temperature gradients, the A-scan mode appears to have a resolution of a few tenths of a millimeter at 30 cm. The C-scan mode gives a picture-like reproduction of the object shape, but the resolution is not as good as in the A-scan mode. Much additional development is necessary for the optical imaging of the C-scan signals.

(2) Acoustic Holography. Recent developments in optics and acoustics have merged in a promising technique known as acoustic holography, in which the well-known reconstructive-wave fronts technique of optics is extended to acoustics. This two-stage process first produces the recorded diffraction pattern of an object irradiated by sound waves and biased by a coherent reference wave. This record is hereafter referred to as an acoustic hologram. A realistic three-dimensional visual image can be created by interrogating the acoustic hologram with a suitable coherent light source.

The resolution of a holographic system is comparable to the resolution of a conventional system, except where turbulence or turbidity are present; then the holographic method results in better resolution. If an object is diffusely reflecting and can be considered an assembly of unresolved scattering points, the signal-to-noise ratio is independent of the area of the hologram. The increase in size does, however, increase the resolution of the individual scatterers.

Preliminary experiments on the application of acoustic holography to liquid-metal systems show definite promise of successful viewing of objects in a liquid-metal medium with external equipment only.[152] Acoustic shear-wave generation in the liquid-metal surface is the principal difficulty. Acoustic waves used in holography are longitudinal. At an interface, shear waves are generated that travel along the interface and, in turn, generate other longitudinal waves. The receiver has no way of differentiating between desired and undesired longitudinal waves. Consequently, the image may be distorted.

The potential advantages of acoustical holography for studies in a safety test facility are its simplicity, enormous depth and angular field of view, insensitivity to turbulence or turbidity of the medium, and the reconstruction of the image in three-dimensional form. Additional development is required to establish the full potential of holographic systems.

3. Flowmeters

a. TREAT Experience. Electromagnetic flowmeters have been used routinely in TREAT loop experiments.[81] Their response is essentially limited by that of the recording oscillographs used for transient data (~3000 Hz, or higher, depending on the galvanometers). In addition, during a severe loop meltdown with seven EBR-II pins, the loop flowmeter apparently measured the flow of molten uranium accurately.[153]

b. SEFOR Instrumented Fuel Assembly. A magnetic flowmeter, mounted on the bottom of the instrumented fuel assembly, consists of an Alnico VI cylindrical magnet canned on the inside and outside with 0.051-cm-thick stainless steel tubes. The meter has an overall length of 17.8 cm and an OD of 4.67 cm. The ID of the test model was 3.05 cm, but other throat sizes are used to conform with the orificing pattern across the core radius. The design flow rate is from 0.8 to 4.5 liters/sec. The meter was tested in a sodium flow loop at temperatures up to 540°C. At 6.5 liters/sec, the meter generated a signal of 12.5 mV. The measured strength of the magnet was 648 G.

c. EBR-II Instrumented Subassembly. A permanent-magnet (PM) type flowmeter has been selected to sense coolant flowrates in the prototype subassembly. This selection was based upon successful operation of a similar unit for over 2000 hr up to 540°C.[154,155]

The following design criteria have been established:

Geometry	Hexagonal
Distance across flats, cm	11.15
Overall length, cm	59.28
Flow-conduit ID, cm	4.03
Maximum flowrate, liters/sec	3.8
Pressure drop at max. flowrate, atm	0.6
Integral extension lead wire	
Wire size	24 AWG
Wire length, cm	208
Sheath diameter, cm	0.318

The magnetic circuit provides a fairly uniform field within the flow conduit and minimizes end effects. As shown in Fig. 13, the magnetic field is produced by Alnico-5 magnets operating in parallel. The pole pieces are fully annealed soft iron. The electrodes and the extension lead wires are fabricated of Type 304 stainless steel to minimize thermoelectric and thermal-expansion effects; they are insulated with high-purity alumina. It is planned to backfill the flowmeter with helium (through the evacuation tube) to prevent initial corrosion of the pole pieces and magnets, and to provide a medium for removal of gamma heat generated in these components.

Operation of a PM flowmeter is similar, in principle, to that of a dc generator. As the conducting liquid (sodium) flows through a tube positioned in a stable magnetic field, a potential is developed across the tube and at right angles to the field. This potential is proportional to the volumetric flowrate of the liquid and the magnetic-flux density. Based upon the expected magnetic-flux density and the conduit ID, a sensitivity of 5.15 mV-sec/liter is anticipated at a sodium temperature of 430°C, or about 19.5 mV output at the maximum flowrate (3.8 liters/sec).

4. Pressure Measurements

a. TREAT Experience. Pressure transducers exposed to the high transient radiation fields in TREAT show response to both reactor power and integrated power.[156] Power sensitivity has been essentially eliminated by use of certain variable-reluctance transducers or by means of a balancing circuit. For the miniature strain-gauge units used, signals proportional to integrated power have been as high as ~15% of full scale. (This sensitivity appears as a smooth "zero change" and has been tolerated.) A variable-reluctance unit has greatly decreased sensitivity to integrated power. Because of the relatively high dynamic range predicted to be necessary for ceramic experiments in which pin failure may be followed by rapid thermal mixing between coolant and fuel at around 2500°C, these transducers have been adapted for service in advanced TREAT integral loops.[157] They are similar to the transducers selected in an independent study as most suitable for the SNAPTRAN-2 transient pressure measurements.[158,159]

Use of strain-gauge transducers for TREAT sodium loops, similar to those used to measure internal fuel-rod pressure in SNAPTRAN Test 1,[159] required development of special techniques for coupling the instruments to the loops.[160] Data have been reported from tests performed on the frequency response of these transducer installations.[161] Included were shock-tube studies on response of individual transducers and complete NaK-filled transducer coupling subassemblies and also studies on response of the transducer-subassembly-loop system (for Mark-I integral TREAT loops). The principal problem disclosed by these latter tests was the strong influence of loop design on effective response of the

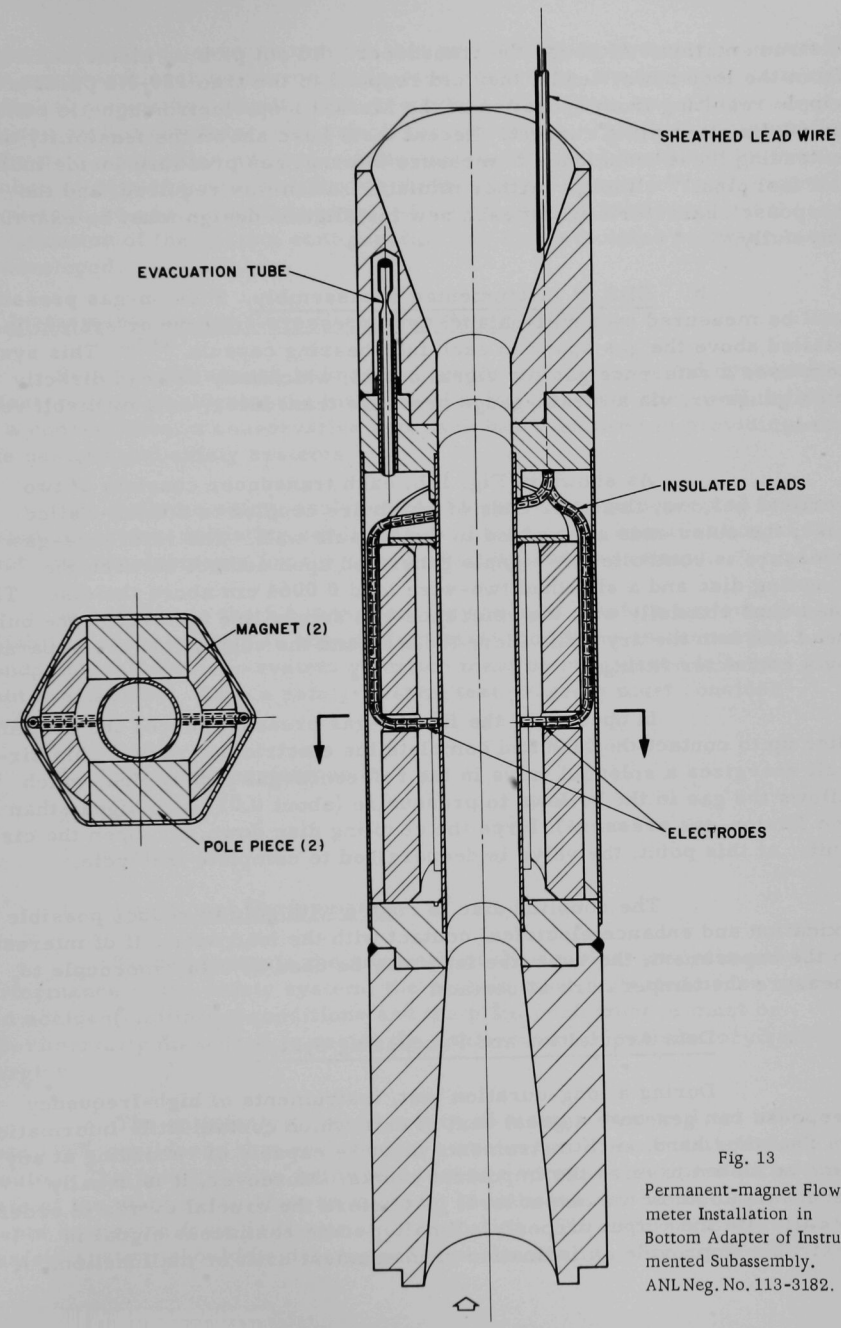

Fig. 13

Permanent-magnet Flowmeter Installation in Bottom Adapter of Instrumented Subassembly. ANL Neg. No. 113-3182.

instrumentation. Although the transducers did not pick up electrical signals from the loop power leads, they did respond to the true 120-Hz pressure ripple resulting from operation of the Mark-I loop electromagnetic pump on 60-Hz alternating current. Recent tests have shown the feasibility of extending these techniques to measure internal gas pressure inside individual fuel pins,[162] although further miniaturization was required, and the response characteristics of each new installation design must be examined carefully.

b. EBR-II Instrumented Subassembly. Fission-gas pressures will be measured by a null-balance type pressure-transducer system installed above the gas space in each fuel-bearing capsule.[129,163] This system uses a reference gas for signal output, which may be read directly on a gauge or, via a strain-gauge pressure transducer, on a millivolt recorder.

As shown in Fig. 12b, each transducer consists of two vertical bellows, the inner ends of which are coupled to a thick-walled disc; the outer ends are welded to the capsule wall. The reference-gas pressure is controlled by signals generated upon contact between the coupling disc and a sheathed two-wire lead 0.0064 cm above the disc. This lead runs coaxially with the reference-gas supply line up through the bulkhead and into the dry well. Here the lead and the supply line are separated by a connector fitting.

In operation, the fission-gas pressure forces the coupling disc up to contact the lead and complete the electrical circuit. This circuit energizes a solenoid valve in the reference-gas supply line, which allows the gas in the bellows to pressurize (about 0.017 atm higher than the fission-gas pressure), force the coupling disc down, and open the circuit. At this point, the valve is deenergized to complete the cycle.

The coupling disc is coated with gold to reduce possible oxidation and enhance electrical contact with the lead wire. If of interest to the experiment, the two-wire lead may be used as a thermocouple to measure the temperature at contact.

5. Data Acquisition and Processing System

During a long-duration test, instruments of high-frequency response can generate a great deal of data which contain little information. On the other hand, such instruments must be capable of recording at any time so as not to miss the important events. Moreover, it is usually necessary to know the output level just before the crucial events in order to calibrate the output properly. Finally, some continuous signal is desirable to provide an indication of instrument drift or malfunction. A

variety of methods exist to resolve this problem using commercially available equipment such as fm magnetic-tape recorders, multiplexers, analog-to-digital converters, digital-tape recorders, disc files, and a general-purpose digital computer with a fast-core memory. The final configuration of the data acquisition and processing system will be determined by the number of data channels and the required time resolution for each channel. In general, no special R&D is required for component hardware, but the specification of the system configuration and the associated software must be developed.

C. Instrumentation for Core Protective Safety Systems

Because not enough is known about the mechanisms and dynamics of fast-reactor accidents, the margins of safety have not been well defined. As a consequence, a conservative approach has been used in providing safe designs and safety systems.

The systems safety analyses of reference design concepts, using assumed accident initiating conditions, will lead to the specification of the functional requirements for the core protective safety systems.

In addition to the functional performance of a safety system, the reliability of the system is equally important, and from an economic standpoint, false safety-system actuation must be considered in terms of plant availability. Thus a safety-system test program must consider the following:

1. Functional performance
2. Reliability
3. Availability

1. Functional Performance

To demonstrate the adequacy of the accident analysis and the performance of the safety system, the system must be tested under specified accident-initiating conditions and the plant performance must be experimentally measured to verify the achievement of satisfactory safety margins.

The satisfactory performance of a single test will not in itself yield 100% confidence that each test in a series would yield satisfactory results. The results of an individual test are completely dependent on the state of the plant at the time an accident is initiated and on the chain of events occurring from initiation to termination. Thus the performance of a safety system should be determined experimentally under accident

conditions that are as realistic as possible. The level of confidence to be associated with a series of tests will be determined by applying statistical methods to the experimental results. Testing will also determine whether an accident could prevent successful operation of the safety system through some consequential effect which had not been included in the original analysis.

2. Reliability and Availability

Reliability is defined as "the probability that a system will perform satisfactorily for at least a given period of time when used under stated conditions," and availability is defined as "the probability that a system is operating satisfactorily at any point in time when used under stated conditions." The functional performance of a safety system can be experimentally verified if a series of tests to explore all anticipated accident conditions and event sequences can be made in a period of time that is short compared to the anticipated life of a plant, i.e., experimental time of one month or one year as compared to an anticipated plant life of 20 or 40 years. Thus, an experimental program designed to establish functional performance will verify system reliability only in the early life of a plant. It is extremely important that the reliability of a safety system be numerically evaluated for the entire life of a plant.

Reliability analysis has been successfully applied to military systems.[164-167] The success achieved with the complex multicomponent missile systems prompted the application of reliability analysis to nuclear-reactor systems.

The latest application of reliability theory to a nuclear-reactor system is the fault-tree analysis of the Power Burst Facility (PBF) rod-drive system.[168] Fault-tree analysis provides a systematic procedure to identify and record the various combinations of component states or events that can result in the occurrence of a predefined system state or event. The PBF analysis indicated that the probability that four rods would be withdrawn from core at an excessive velocity and fail to stop at the position crossover point was 0.041, which indicates that a reactivity accident is well within the bounds of credibility. As the result of this analysis, a velocity-limiting device was recommended to be incorporated into the final system design with an expected reduction of several orders of magnitude in the probability of a reactivity incident.

A preliminary reliability analysis of the FFTF safety circuits was based on a conceptual version of the design.[169] The analysis, using two-out-of-three redundancy, showed a fail-to-unsafe probability of 26×10^{-6}, a reliability of 0.999974 for an operating time of 1000 hr, or a probability of failure in 20 years of 4.5×10^{-3}. Similarly, the analysis revealed a reactor unavailability due to spurious tripping of 0.134 hr per 1000 hr.

To simplify the reliability analysis, the following assumptions are generally employed:

a. Components are operating in their useful life phase where the failure rate is constant and the distributions of time between failures can be expressed by exponential distributions.

b. Component failure rates are estimated using published data for similar equipment.

c. Component failures are independent.

d. Testing and repairs have neither altered the original design, nor added new faults.

3. Experimental Verification of System Reliability and Availability

Without recourse to a mathematical model of a system, demonstration of compliance with a 10^{-5} probability of failure for a reactor protection system would be a formidable task.[170] If the concept of a confidence limit is used, the system failure rate can be obtained by

$$\lambda = \frac{\chi^2_{2r+2,\alpha}}{2nT},$$

where

n = total number of units in service,

T = total time the equipment has been in service,

α = confidence limit on the parameter λ,

λ = failure rate of the unit,

r = total number of failures,

and

$\chi^2_{2r+2,\alpha}$ = chi-square distribution evaluated at 2r + 2 degrees of freedom at the α confidence limit

If the safety system is assumed to be operating in its useful life range and is tested periodically to confirm that it is operable, then the average reliability is given by

$$R = 1 - \frac{\lambda\tau}{2}, \quad \lambda\tau << 1,$$

where

τ = test interval.

If it is also assumed that a test interval of 8 hr is used, then

$R = 10^{-5}$,

$r = 0$ (no failure),

$n = 1$ (single system),

$\tau = 8$ hr,

and

$\alpha = 0.95$ (95% confidence).

Solving the reliability equation for λ gives

$\lambda = 0.25 \times 10^{-5}$ failures/hr,

and solving the chi-square equation for T results in

$T = 1.2 \times 10^{6}$ hr (137 yr).

Thus, the system would have to be operated for 137 yr with no failures. (Note that the product nT enters the equation for λ, not n or T separately. This product is sometimes known as "system hours".)

As indicated by the preceding example, the experimental verification of system reliabilities of the order of 0.99999 is exceedingly difficult, unless the number of systems tested is increased. At best, data on the experimental failure rate, failure mode, and failure distributions for individual components can be accumulated and used to modify the assumed probability models.

The only practical means to obtain some measure of the reliability of a highly reliable, complex system is by mathematical modeling. The design of such systems should, therefore, enable the justification of the basic assumptions underlying presently available mathematical modeling techniques (constant failure rates, independent component failures, etc.).

With these qualifications, a core-protective safety system can be experimentally tested to verify functional performance. Tests conducted on less than full-size cores, however, must be analyzed correctly in order to extrapolate and predict the functional performance of similar systems on full-size cores. Thus, operation of a demonstration plant will yield valuable

information on the reliability of fuel and coolant thermocouples. But the information on the reliability of fission-gas and boiling detectors will be much less. If a three-level safety scheme (see the LMFBR Program Plan, Vol. 10, pp. 10-34) is to become effective, the desired reliability objectives must be quantified and the operational schemes for demonstration plants and in-core safety experiments reviewed to see if routine testing in such facilities is adequate. If not, an improved testing schedule will need to be devised.

D. Conclusions

Although the sensors developed in support of safety experiments can be used as sensors for core protective systems, core protective sensors, cables, and connectors require additional development to guarantee acceptable reliability over the required life of the sensor. Also, sensor and cable installation and removal as part of routine fuel-handling operations must be considered.

The current state of instrument development includes some effort on thermocouple design for use with fuel pins. There is a need for development of small thermocouples for surface-temperature measurements and for reliable means of getting coolant temperatures in the channel without perturbing channel flow. This may be possible by using acoustic probes, and such probes may also be useful for local void detection.[171] Continued effort is required in this direction.

Flow measurements appear satisfactory, although difficulties are caused by high temperatures.[172-174] Effort should be continued to reduce these difficulties.

Pressure and strain measurements require correction for background effects.[175,176] In long-duration tests such corrections may become substantial; a continued investigation of the problems involved here is indicated.

The details of fuel motion present the biggest single problem of observation. For larger sodium-cooled systems, an alternate approach to the neutron hodograph is acoustic holography. Work in this field is proceeding at a low level and seems to be aimed at static rather than dynamic techniques. A careful evaluation of this technique and its potential for dynamic observations is needed; if such an evaluation shows that acoustic holography can be as useful as it appears, an increased effort is required for its development.

If a fast-neutron hodoscope is to be used in an in-pile test program, special conditions must be met in the facility design. These are

discussed in Appendix B; chief among them is a slot through the core. Thus there is an incentive to seek viable alternatives including in-core instruments which might be destroyed during an experiment. Such alternatives are being sought during these studies, but to this date there are no promising approaches other than the acoustic method.

E. General Bibliography for Chapter IX

T. T. Anderson and T. A. Grate, Acoustic Boiling Detection in Reactor Vessels, IEEE Trans. Nucl. Sci. NS-16, No. 1, 260-265 (Feb 1969).

M. N. Baldwin and J. E. Rogers, Calibration of the Self-Powered Neutron Flux Monitor in a Simulated PWR Lattice, IEEE Trans. Nucl. Sci. NS-16, No. 1, 171-176 (Feb 1969).

J. C. Chen, S. Kalish, and G. A. Schoener, Probe for Detection of Voids in Liquid Metals, Rev. Sci. Instrum. 39, 1710-1713 (Nov 1968).

G. J. Dau, R. R. Bourassa, and S. C. Keeton, Nuclear Radiation Dose Rate Influence on Thermocouple Calibration, Nucl. Appl., 5, No. 5, 322-328 (Nov 1968).

S. Fam, L. C. Lynnworth, and E. H. Carnevale, Ultrasonic Thermometry in LMFBR Systems, NYO-3906-5 (Dec 1968).

J. R. Folkrod, Design and Performance Characteristics of a Miniaturized Null Balance Fission-gas Pressure Transducer and Controller, ANL-7573 (to be published).

J. R. Folkrod, Design Parameters and Performance Characteristics of a Miniaturized Pressure Transducer System Using a Fluid-filled Bellows Sensor, ANL-7574 (to be published).

W. P. Gilles, Radiation Effects on Nerva Instrumentation, IEEE Trans. Nucl. Sci. NS-15, No. 6, 239-243 (Dec 1968).

H. W. Glauner, G. R. Pflasterer, and J. F. Momberger, In-Core Instrumentation and Fast Digital Data Acquisition for the SEFOR Nuclear Test Reactor, IEEE Trans. Nucl. Sci., NS-14, No. 1, 293-300 (Feb 1967).

M. S. McDonough, L. C. Lynnworth, and E. H. Carnevale, Ultrasonic Measurement of Core Material Temperature, Phase 2, NASA-CR-72476 (Aug 1968).

A. Padilla, R. E. Peterson, and J. J. Regimbal, FFTF Core Instrumentation Bases and Requirements, Trans. Amer. Nucl. Soc. 11, No. 5, 693-694 (Nov 1968).

H. J. Worsham and H. D. Warren, An Investigation Into the Effects of In-pile Radiation and Temperature on the Lead Wires of Chromel-Alumel Thermocouples, IEEE Trans. Nucl. Sci. NS-16, No. 1, 188-191 (Feb 1969).

F. Bibliography on Testing of Core Protective Systems

L. M. Arnett, Quantitative Analysis of Reactor Safety, DP-1168 (Nov 1968).

N. C. Barford, Experimental Measurements: Precision, Error and Truth, Addison-Wesley Publishing Co., London (1967).

R. E. Barlow, F. Proschan, and L. C. Hunter, Mathematical Theory of Reliability, John Wiley & Sons, Inc., New York (1965).

R. E. Barlow, Some Recent Developments in Reliability Theory, ORC 68-19 (July 1968).

E. L. Battle, Basic Reliability Concepts, Air Univ. Quart. Rev. 13, No. 2, 83-92 (Fall 1961).

I. Bazovsky, Reliability Theory and Practice, Prentice-Hall, Inc., Englewood Cliffs, N. J. (1961).

R. L. Calvert, Some Notes on the Estimation of Reliability, AECU-3425 (Sept 1956).

W. S. Connor, Interpreting Reliability by Fitting Theoretical Distributions to Failure Data, Ind. Eng. Chem. 52, No. 4, 71A-72A (April 1960).

M. J. Cowper and D. Wray, Estimation of Accident Probability Due to Protective System Failure in Power Reactors, AHSB(S)R35 (1962).

O. L. Davies, The Design and Analysis of Industrial Experiments, Hafner Publishing Co., New York (1954).

J. S. Donaldson and F. K. Heiden, "A Reliability Analysis and Prediction Technique Applicable to Electronic and Nonelectronic Systems," in Proceedings IAS Aerospace Systems Reliability Symposium, Salt Lake City, Utah, pp. 46-55 (April 1962).

G. W. A. Dummer and N. B. Griffin, Electronics Reliability - Calculation and Design, Pregamon Press, Inc. New York (1968).

G. W. A. Dummer, Reliability, Pergamon Press, Inc. New York (1968).

A. R. Eames, Reliability Assessment of Protective Systems, Nucl. Eng. 11, No. 118, 188-192 (March 1966).

D. R. Earles and M. F. Eddins, Reliability Physics, N-63-18306 (March 1962).

D. R. Earles, Reliability Application and Analysis Guide, MI-60-54 (Rev. 1) (July 1961).

S. H. Fitch, Reactor Safety Device Reliability Determination Using Sequential Analysis, NAA-SR-4769 (Feb 1960).

M. Freitag and W. A. Cleveland, "One-shot System Reliability Prediction and Demonstration," in Proceedings IAS Aerospace Systems Reliability Symposium, Salt Lake City, Utah, pp. 26-32 (April 1962).

D. A. Gardiner, Predicting Reliability of Safety Devices, Nucl. Safety 2, No. 2, 27-30 (Dec 1960).

B. J. Garrick and W. C. Gekler, Reliability Analysis of Engineered Safeguards, Nucl. Safety 8, No. 5, 470-479 (Sept-Oct 1967).

B. J. Garrick, W. C. Gekler, and H. P. Pomrehn, Are Engineered Safeguards Reliable? Nucleonics 25, No. 1, 45-49 (Jan 1967).

B. J. Garrick, W. C. Gekler, and H. P. Pomrehn, An Analysis of Nuclear Power Plant Operating and Safety Experience, Vols. 1 and 2, HN-185 (Dec 1966).

B. J. Garrick, Reliability Analysis of Nuclear Power Plant Protective Systems, HN-190 (May 1967).

Gilbert Associates, Inc., Reliability Study. Experiment and Components in Nuclear Power Plants, GAI-1512 (July 1960).

A. E. Greene and A. J. Bourne, Reliability Considerations for Automatic Protective Systems, Nucl. Eng. 10, No. 111, 303-306 (Aug 1965).

A. E. Greene and A. J. Bourne, Reliability Considerations for Automatic Protective Systems, AHSB(S)R-91 (1965).

C. W. Griffin, Introducing the Fault Tree as a Tool for Nuclear Safety Analysis, Trans. Amer. Nucl. Soc. 9, No. 1, 157-159 (June 1966).

J. Hedde, Calculation of Reactivity by Digital Processing, CEA-R-3691 (in French) (Dec 1961).

G. Hefferon et al., Theoretical and Experimental Analysis of Availability in Nuclear Power Plants, AFWL-TDR-64-24 (Aug 1963).

W. G. Ireson, Reliability Handbook, McGraw-Hill Book Co., New York (1966).

I. M. Jacobs, Safety-System Design Technology, Nucl. Safety 6, No. 3, 231-245 (Spring 1965).

C. E. Jowett, Reliability of Electronic Components, Illiffe Books, London (1966).

O. Kempthorne, The Design and Analysis of Experiments, John Wiley & Sons, Inc., New York (1962).

G. J. Levenbach, Systems Reliability and Engineering Statistical Aspects, Amer. Sci. 53, No. 3, 375-384 (Autumn 1965).

D. K. Lloyd and M. Lipow, Reliability: Management, Methods and Mathematics, Prentice-Hall, Inc., Englewood Cliffs, N. J. (1962).

A. E. Mace, Sample Size Determination, Reinhold Publishing Corp., New York (1964).

T. L. McLean, Some Aspects of Reliability in Reactor Instrument Systems, Nucl. Safety 8, No. 4, 356-359 (Summer 1967).

C. W. McKnight and W. H. Hatton, An Automatic Mathematical Reliability Model, NA-64777 (Aug 1964).

J. O. Muench, A Confidence Limit Computer, SCR-159 (April 1960).

R. H. Myers, K. L. Wong, and H. M. Gordy, Eds., Reliability Engineering for Electronics Systems, John Wiley & Sons, Inc., New York (1964).

K. C. Peng, The Design and Analysis of Scientific Experiments, Addison-Wesley Publishing Co., Reading, Mass., (1967).

E. Pieruschka, Principles of Reliability, Prentice-Hall, Inc., Englewood Cliffs, N. J. (1963).

A. M. Polovko, Fundamentals of Reliability Theory, Academic Press, Inc., New York (1968).

J. Rasmussen and P. Timmermann, Prediction of Failures in Reactor Safety Instruments, RISO Report No. 37.

J. Rasmussen and P. Timmermann, Analysis of Failure Data for Electronic Equipment at Risö, RISO Report No. 38 (Sept 1963).

N. H. Roberts, Mathematical Methods in Reliability Engineering, McGraw-Hill Book Co., New York (1964).

R. A. Rohrbacher, Reliability Analysis of Nuclear Reactor Electronic Systems, DUN-SA-66 (June 1968).

G. C. Rudy and J. T. Stone, Use of Reliability Theory to Examine Safety-System Designs, Trans. Amer. Nucl. Soc. 8, No. 1, 138-139 (June 1965).

G. H. Sandler, System Reliability Engineering, Prentice-Hall, Inc., Englewood Cliffs, N. J. (1963).

CHAPTER X

SUMMARY REVIEW

A. Introduction

In this chapter we summarize and review the work presented in the main body of this report, concentrating on that not requiring large experimental installations.

The various areas of studies refer to the description of the processes in fast-reactor accidents in terms of basic phenomenology. The phenomenological areas are related by thermal and mechanical effects and the neutron field of the reactor. The phenomenological descriptions represent different aspects of the accident occurring at very nearly the same time. It is the thesis of Chapter II (1) that such accidents can be satisfactorily analyzed, provided one has an understanding of the phenomena that are important, and (2) that phenomena are classified as important if they control the limits of damage. Such a thesis requires a statement of what is meant by "satisfactorily analyzed."

The field of accident analysis is diverse enough so that no universal statement of what constitutes satisfactory analysis can be made that is not empty. But if we restrict ourselves to core-related accidents and, in particular, to serious accidents, we can come to the following conclusion: Satisfactory analysis of an accident consists of an analysis based either on first principles or on firmly established empirical data that uniquely relate the upper limits and, if possible, the lower limits of the damage caused by a sequence of events to the design parameters of the system.

Such a definition emphasizes the role of safety analysis in determining the basic design of the reactor system. A program of reactor-safety research will not, by itself, perform such an analysis. The aim of reactor-safety research is to furnish the methods and the data whereby the analysis can be made.

We now review each area of study discussed in the preceding chapters to establish the main areas of research needed to fulfill this objective of an LMFBR safety program.

B. Sodium Boiling

The existing program of work in sodium heat transfer shows considerable promise and has already answered many questions. By and large, the program of work is an out-of-pile program similar to that carried on with many working fluids. Considerable attention has been paid to the use of simulant fluids because of the difficulty of working with sodium. The

areas of study are familiar to those who have worked in reactor-coolant dynamics and are: nucleation and two-phase flow; heat transfer under conditions of varying power and constant flow and correspondingly, constant power and varying flow; local flow blockages. This list has to do with behavior when coolant channels remain intact. An important problem, particularly in conjunction with local flow blockages is the appropriate description of cross flow. The main point is that up to this level of description the topics are familar even though the working material and the heat fluxes may not be. An important problem susceptible to both analysis and empirical verification is the determination of scale effects, particularly in multichannel flows. The reason is that since it is desirable to verify that the presence of radiation and other components of the in-pile environment do not substantially alter the conclusions based on an out-of-pile program of research, the knowledge of scaling laws will permit the use of small-scale in-pile experiments, which are easier and cheaper to perform than full-scale tests.

On the borderline of traditional heat-transfer studies and studies peculiar to LMFBR safety is the behavior of coolant when the flow channels become very hot and massive amounts of coolant are expelled. The expelled coolant condenses in the upper plenum and then flows back into the hot channel. This is a problem of high priority in the analysis of the initial events in an accident sequence. Here we are no longer studying boiling and expulsion as separate phenomena. On reentry, violent boiling may be produced by interaction of molten fuel with coolant. On the other hand, bulk boiling may be avoided by a process wherein small amounts of coolant are vaporized and the vapor pressure prevents the reentry of large amounts of liquid coolant into the hot channel and interaction with molten fuel.

Definitive experiments on this problem will probably have to be conducted in-pile because of the difficulty of generating heat in molten fuel. Coupling of the heat source with the driver seems to be possible in this case only through the neutron field. If an in-pile program is to be informative and of utility to accident analysis, it should be designed to test the effects of important parameters. Thus, it is desirable at an early stage to have some analysis of this problem, even though it may be a very crude one, to indicate which of the design variables are most important.

C. Fuel-pin Failure and Failure Propagation

Failure of the coolant flow is by no means the only cause of fuel-pin failure, but whatever the cause, the most important single safety question about fuel-pin failure is that of propagation. That is, under what conditions can the failure of one pin precipitate the failure of others and set up a chain reaction affecting so many of the pins in a subassembly that the subassembly has to be replaced? Moreover, how long can operation of fuel proceed before the risk of propagation becomes important? These

questions are so important because single pins will probably fail frequently, but the need to replace a whole subassembly with the same frequency would be unacceptable. In addition, any possibility of the damage spreading beyond the subassembly would be very undesirable. The questions of fuel failure are also important for many reasons other than reactor safety, and these have been discussed in both Chapters II and IV.

Basically, cladding failure is the main limit on the irradiation life of the fuel. There is inevitable economic pressure to extend this limit and improve the fuel, and our ability to do so depends on understanding the mechanisms that cause it to fail. The problem of fuel failure, as well as the consequent problem of propagation of the failure, is likely to be with us whenever design changes are contemplated.

An understanding of fuel-pin behavior and failure requires both basic studies of the individual mechanisms at work and empirical data derived from experiments that simulate the situation of interest. In addition, reactor operating experience can provide some information.

Consider first analytical problems, or basic studies. These include: codification of information on the behavior of oxide fuel with respect to cracking, swelling, creep, and fission-product gas release under normal operating conditions. This is required for proper specification of experiments. Codification of similar data for cladding behavior is also required, although the metallurgy of cladding materials at normal temperatures is probably well understood. On the other hand, the behavior of cladding at elevated temperatures under neutron irradiation is very imperfectly understood at present, and the swelling behavior of stainless steel under high fluence presents a major design and consequently safety problem at this time. As with most such problems, we expect that ways to design around it will appear in time, but basic analysis, as well as tests, are needed to elucidate this problem.

Work on pin behavior after failure, the behavior of the released gas, and the behavior of the released fuel is almost certainly largely empirical until some sizable body of relevant information has been accumulated. Any tests in these areas would be mainly in-pile, although out-of-pile tests on the behavior of gas and vapor in the coolant stream between clusters containing various numbers of pins are currently planned.

D. Vapor Explosions

Although there exists a fair amount of empirical information on the interaction of liquid metals and water and some data on other molten material interactions with cold liquids, there is no analytical framework against which a program of research related to LMFBR safety can be planned. The

absence of such a framework is a major deficiency at this time because the use of fuel of high heat content and high working temperature, such as the oxide or carbide, implies that the interaction of molten fuel with coolant must be carefully considered in analyzing the consequences of an accident. The interaction between the molten fuel and the coolant could play the major role in determining the consequences of such an accident, and, in particular, in determining whether a sequence of events becomes autocatalytic.

There are approximately three areas in which the molten fuel-coolant interaction is most significant.

First, in the case of fuel-pin failure, the interaction of molten fuel expelled from a single pin with the surrounding coolant conceivably could give rise to sufficient pressure to deform neighboring pins and cause their failure. Should these pins in turn release molten fuel into the coolant the failure could become autocatalytic.

Second, if coolant should be lost from the major part of a single subassembly, coolant vapor may condense in the upper plenum, allowing liquid coolant to reenter the subassembly to come into contact with molten fuel, giving rise to a pressure wave which could conceivably damage a number of neighboring subassemblies, possibly deform control-rod channels, and thereby set the stage for further serious consequences.

Third, should there be an accident involving collapse of a major portion of the core, the residual energy in the collapsed core is sufficient to vaporize a substantial portion of the sodium coolant. This high pressure and temperature vapor could do considerable work and hence damage before the vapor condenses.

In concept, the interaction process is divisible into these portions:

First, there is the process of fragmentation whereby the hot material has its surface area multiplied very rapidly. The fragmentation and dispersion of the hot material are perhaps the key factor in determining the total heat-transfer rate. On the other hand, the phenomena that determine the rates of fragmentation and dispersion may be remarkably different in, for example, the three areas cited above.

The second major process is the process of heat transfer from the hot fluid itself to the working fluid.

Finally, the third important factor determining the outcome of a vapor explosion is the time available for energy transfer before boundary effects give rise to pressure relief.

An intensive effort is being made to investigate these phenomena both in-pile and out-of-pile. The wide variety of circumstances under which the molten fuel-coolant interaction can occur implies, however, that substantial analytical guidance is required to avoid an extremely massive empirical program. An additional need, just as with sodium-boiling experiments, is for analytical guidance on the importance of scale effects. Here such guidance is even more important then in the sodium-boiling studies because major reliance for basic empirical data must be placed on in-pile tests. Establishing a firm foundation for such data requires the knowledge that the size of the tests did not incorrectly perturb the data observed.

E. Motion of Molten Fuel

A complex situation arises when there is large-scale movement or melting of fuel and large-scale coolant motion on a rapid time scale, as in a serious accident. We then can avail ourselves, in principle, of accurate analytical methods to describe the coupling to the neutron field and thus get an estimate of the energy production during an accident. The real question is not, therefore, the analytical framework, as in some of the previous cases, but how the fuel and coolant move. Because of the wide variety of competing forces, the answer is not straightforward and is likely to be resolved only by an intensive program of in-pile research involving sophisticated instruments such as the fast-neutron hodoscope. As in all such cases, accurate estimation of scale effects is required to get the most out of the experimental program.

As the experimental program develops more of the details of how fuel and coolant move in various typical situations, it may be possible to develop design principles that will minimize the consequences of large-scale movement or melting of fuel.

The work involved in this program is largely technical. A fruitful field for investigation is the program of development of adequate instruments for observing the fuel and the coolant. Some of these problems are discussed in Chapter IX and Appendixes A and B. Fundamentally what is needed is a way of taking high-speed motion pictures through sodium and through massive amounts of steel.

Small-scale out-of-pile experiments may be of use in elucidating some of the competing mechanisms at work in this complex situation. The utility of such experiments is by no means clear, but with careful analysis they might provide some help in understanding the problem and in guiding or interpreting in-pile tests.

F. Severe Excursions

The mode of analysis of excursions involving reactivities above prompt critical, while complex, is at least in principle reasonably well determined, and, if there is extensive work to be done in making such calculations straightforward, the program of work is clear-cut and the difficulties come not in discerning the framework but in filling in the pieces.

At this point in time, it is too early to tell if an extensive program of work of an empirical nature is required to verify the analytical methods of predicting the energy release in super-prompt-critical excursions. What is needed, and what is being done, is to analyze the sensitivity of the calculated energy yield to the parameters of the system. In particular, the equations of state of the materials of the reactor and the initial accident conditions are known to be significant.

It is important that a really careful and thorough study be done to determine the sensitivity of accident consequences to the uncertainties in the equations of state. The starting point for this study must be a study of the equations of state. They should be cast in a form involving a small number of parameters on which uncertainty bounds can be set. A major problem is that the setting of these uncertainty bounds will require technical judgement where different authorities may disagree. In spite of this, a start on this effort is essential.

Having the equations of state with uncertainties, one can then investigate the effect on accident consequences for a chosen range of accidents and reactor types. There are also judgments to be made here on accident initial conditions and calculational models. There is not yet a single code that accepts initial conditions and then goes through the complete series of events leading to final consequences. The analysis is still done piecewise. In super-prompt-critical excursions, this usually means making an estimate of the rate of insertion of reactivity and material configuration at prompt critical, making an energy-release calculation, and then making an estimate of the resulting damage. The equations of state are involved in both the energy-release calculation and the damage estimate. Different ranges of the equations of state are required in the estimates of the energy released and of the damage done, with some overlap in ranges. It is important that there be consistency here.

Lacking authoritative estimates of the uncertainty bounds in the equations-of-state parameters, one can nevertheless perform a formal study of the sensitivity of accident consequences to these parameters.

The energy release in super-prompt-critical excursions is most sensitive, however, to the initial conditions. This is an area in which it is

difficult to do meaningful sensitivity studies. Any result whatever can be obtained, unless some relationship to reality is imposed on the range of conditions studied. A great deal of judgment is required in the choice of conditions to be covered in a particular study. There are two basic reasons for the wide range of conditions to be studied.

First, suppose that the detailed behavior of the core and the real accident conditions were fully understood. It is nevertheless likely that the initial conditions would be very significant in determining this behavior, so that large variations in the rate of increase of reactivity and in available reactivity could be obtained or large variations in the amount of sodium in the core at prompt critical. That is, small changes in initial conditions might yield large changes in final results.

The other reason for varying initial conditions in accident studies is that we do not fully understand the detailed core behavior under accident conditions and so have to study also ranges of conditions that are not physically achievable to be assured of a conservative estimate. Many of the experiments suggested in the earlier chapters of this report are directed at reducing this uncertainty. Nevertheless, it must be assumed that when we are ready to do experiments relating to super-prompt-critical excursions, there still will exist some uncertainty about detailed core behavior and its effect on an excursion and relationship to initial conditions.

An experimental program cannot possibly cover the complete range of conditions. It is necessary in planning the experimental program to rely on analysis to guide the selection of initial conditions for a relatively small number of experiments. Thus, an extensive analytical program is required on sensitivity of energy release and its consequences to initial conditions. This probably should not be undertaken on a large scale immediately, but should wait some further refinement of codes that permit a consistent analysis from start to finish of an excursion.

Finally, we have not differentiated too clearly between the analysis of the energy release and the damage it does. But we may be able to study damage produced by an excursion through simulations involving chemical explosives. This is discussed in detail in Chapter VII. Clearly, more analysis is required on the principles that govern the response of complex structures to explosive energy releases before this question can be fully resolved.

G. Postburst Phenomena

Following a severe accident, the release of radioactivity to the environment depends upon a number of processes that are lumped under the heading of postburst phenomena. These processes include the

ultimate containment of core debris, fission-product and plutonium release, and sodium fires. All of these problems are related to the ultimate problem of reactor safety, which is the retention of fission products and other sources of radiation to protect the public. In our studies, we have focused most of our attention on those aspects of safety that are related to core design. The postburst problems of safety are no less important, but they do not as directly affect the design of the core as do the other problems we have discussed.

1. Ultimate Containment

Ultimate containment of core debris after a core-disassembly accident is a problem if there is no effective way of removing the decay heat following an accident. A related problem is the possibility of secondary nuclear criticality as the core melts and slumps. The interaction of molten fuel and coolant is once again a problem to be considered here in two phases:

a. There is the possible violent boiling that might occur if sodium once expelled from the core returns to a completely disassembled core following the nuclear burst. This is the source of energy release considered by Hicks and Menzies.[58]

b. There is the process of heat transfer at a lower rate from the molten core or partly molten core to the sodium coolant during the time when the decay heat is sufficient to keep the core in a molten state.

There appear to be straightforward solutions to this problem, and the problem of ultimate containment would not loom so large were it not so important that an adequate solution be obtained and were there not a few major points of real uncertainty. Principal among these is the question of how the core slumps in the first place. A few analyses have been performed, based largely on the slumping of soils, but an attack related more directly to the nuclear core and its structure is desirable.

There are many conjectures on the nature of the heat transfer and the possibility of violent eruptions of molten core material into the cooler sodium. And although it appears reasonable to perform experiments both in-pile and out-of-pile to determine the general nature of the mode of heat transfer from the molten core, it is desirable to have some analytical representation of the effects of scale.

2. Fission-product and Plutonium Release

Fission-product and plutonium release is a major topic of plant safety. It is unlikely that further work in fission-product and

plutonium release will greatly influence LMFBR core design. Largely, this is because so much is already known about this topic. On the other hand, plant safety and, hence, plant design rest in crucial ways on data regarding methods for containing fission products and plutonium that may be released from the fuel under normal or near-normal operating conditions.

An understanding of the chemical and thermodynamic properties of the fission products and the kinetic processes involved in their transport is required for predicting the release rates of these materials. There is an extensive program of work to determine these properties, and major verification of the out-of-pile and analytical investigation is expected from in-pile experiments.

3. Sodium Fires

There is a need for an analytical description of the release of chemical energy and radioactivity within primary and secondary containment as a result of a sodium fire. A substantial program of work is being pursued on this problem, and a number of analytical and experimental tools are available for analysis of the problem.

H. Instrumentation

In-core instrumentation associated with the LMFBR safety program can be divided into two categories: instrumentation in support of in-pile safety experiments, and instrumentation for core protective systems.

1. Instrumentation in Support of In-pile Safety Experiments

A major reason for the difficulty of performing in-core experiments is the difficulty of obtaining the needed information. The problem of providing an adequate in-pile space and test environment is the other major difficulty. The review in Chapter IX indicates the state of the art and some of the more promising avenues of development of better instrumentation. The biggest single need is a way to measure the movement of fuel and coolant during a power transient in large experiments. The only tool currently available with any prospect of success is the fast-neutron hodoscope.

2. Instrumentation for Core Protective Systems

While maximum advantage must always be taken of inherent reactor properties, the major dependence for safe operation of economical LMFBR's must ultimately be on an active and reliable system of safety instrumentation. One purpose in understanding the phenomena involved in accident sequences is to be able to predict with some confidence the

sensitivity and dynamic response required of safety instrumentation so that it may be effective. Assuming that such requirements have been determined, it is equally necessary to describe and define adequately a program to ensure the effectiveness and reliability of the instrumentation. Such programs have been carried out with a high degree of success within the Aerospace program, most notably in the space-exploration program. Many analytic and computational techniques of considerable power have been developed. It is desirable that such an analytic and computational base be supplied for the study of safety instrumentation. Execution of the program probably requires the use of massive in-pile facilities.

I. Conclusion

A program of research in LMFBR safety is multifaceted. Many of these facets resemble work programs typical of engineering and applied-science university departments. Participation by such groups in LMFBR safety research can yield significant gains in a vital area of work.

APPENDIX A

Detailed Description of In-pile Tests on the Motion of Molten Fuel

1. Types of Accident Conditions

The recommended in-pile test program on the dynamics of fuel meltdown consists of tests carried out under conditions simulating conditions in-pile during an accident. We consider four types of accident conditions.

a. Power-flow Mismatch

Power-flow mismatch can be induced by:

i. Pump failure with delayed scram.

ii. Sudden flow blockage by foreign material.

iii. A prolonged overpower transient stemming from a reactivity increment (typically ~30¢ to $1), or reactivity ramp expected from severe damage to another fuel subassembly, control-rod malfunction, etc.

iv. Large gas bubble entrainment or a large fission-gas release.

(1) Test Conditions--Pump Failure or Flow Blockage. Tests simulating pump failure or flow blockage would typically take place in times of the order of 10 to 30 sec, with a period of prior operation to reach equilibrium and one of postoperation to simulate afterheat. The variables in the case of pump failure or flow blockage (i and ii, above) are the rate of loss of flow and the rate of decrease in power. The course of an accident induced by pump failure with delayed scram can be reasonably well estimated analytically to the point of fuel failure. The program on coolant dynamics should enhance the accuracy of the estimation. Such analyses yield a maximum permissible time delay for scram. Some tests of instrument response and control rod design can be performed out-of-pile. But for sudden, massive flow blockages, failure of scram instruments, or other cases in which the maximum permissible delay time is too small, or the case of blockage of one subassembly, fuel melting on a large scale will occur.

For small-scale tests, a facility such as TREAT or PBF appears to offer a good environment. Basic TREAT capabilities* for this type of test are discussed in Section b of this appendix. The problems of no flow can be studied with a simplified environment for single pins using

*PBF capabilities are expected to be similar to TREAT with the added capability of extended operation at "steady-state" power levels in the test zone; decay heat can also be simulated.

stagnant sodium capsules, a type of experiment already well-established for both metallic[139] and mixed-oxide[177] pins. Greater realism can be afforded using a channel geometry (but no flow) in a sodium loop; this has been done with metallic pins, both for the case of a single pin surrounded by a ring of dummy pins,[81] and for the case of a seven-pin cluster. Larger clusters are feasible, but the short time at power limits the degree of simulation available for flow decay. Larger-scale tests are necessary to get accurate simulation of full-scale conditions.

(a) Test Instrumentation. The utility of large-scale meltdown tests rests crucially with the adequacy of the instrumentation. Determining fuel motion is necessary, not only to predict the reactivity insertion at full scale, but also to derive an understanding of the processes governing fuel motion. In this way, the effects of design variations and variations in accident history can be predicted.

The determination of fuel motion is feasible using a fast-neutron hodoscope. This device is described in Appendix B, where it is estimated that massive fuel movements should be readily detectable if proper provision is made for the use of the hodoscope. The hodoscope detects fuel position as a function of time. There are other possible methods, and some of these are described in Chapter IX. Nevertheless, the success of these tests depend upon the ability to observe the fuel motion on a gross scale (and possibly the sodium motion). Data on flow, entrance and exit temperature and pressure also can be used, but are not enough by themselves. Other possible means of detecting fuel motion are largely speculative at this time. For this reason, considerable effort has been spent on estimating the probable performance of a fast-neutron hodoscope in massive fuel tests. In addition to the hodoscope data, reactivity-feedback measurements can provide a check on the estimated fuel motion. Since the inherent errors in estimating individual feedbacks are often very large, such feedback measurements are not expected by themselves to yield useful estimates of fuel motion,* but they can give useful integral checks.

It is useful to have some indication of the resolution required of the hodoscope to yield information on the fuel motion. From the point of view of reactivity change induced by coherent motion, a resolution corresponding to about 10¢ reactivity in the complete core being simulated would be desired. Much less than this represents a prompt power jump that is difficult to detect by conventional means on a short time scale;

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*Even under the most favorable circumstances of no competing effects, the statistical error in the feedback from the test region is expected to nearly equal the statistical error in the total signal, multiplied by the ratio of power in the driver zone to that in the test zone. With such ratios being typically of the order 10-20, phenomenal accuracy in the total reactivity feedback would be required to get even modest accuracy in the derived feedback from the test zone.

much more represents an increment which, in an accident situation, would be capable of causing extensive damage. Using reactivity estimates presented in Ref. 78, one can deduce that 10¢ reactivity change would be induced by the meltdown of about 4.4 liters of a typical PuO_2-UO_2-fueled, 2500-liter core. This corresponds to one-third to one-half the volume of a typical fuel subassembly. As shown in Appendix B, there is ample evidence to expect that this type of resolution can easily be obtained with a time resolution of ~0.001 sec. It is less obvious how to proceed in the case of noncoherent fuel motions. The estimates in Appendix B are that to resolve the motion of a single pin in the center (the worst position) of a large subassembly would require observation times of at least 0.1 sec, if each 0.25 in. of the pin length is to be monitored. Such times are short compared to the periods induced by the removal of a single pin and hence are adequate. But if differential motion is occurring, the instantaneous period may be much shorter. Moreover, if the experiment involves such phenomena as flow coastdown, environmental changes can be expected in 0.1 sec. To monitor the entire length of the pin, however, requires only an estimated 0.01-sec observation time, and this is adequate. For this reason, both small- and large-scale experiments are needed. The hodoscope can readily monitor gross motion and, with reasonable accuracy, the meltdown of a single pin in a large-scale test. But to determine the behavior in detail of a single pin, small-scale tests are needed.

The presence of the hodoscope would give rise to certain operational problems. If extremely massive test loop walls are needed, the hodoscope performance will be degraded. In addition, the hodoscope slots will influence the power generation in the sample. In TREAT, these slots have been beneficial in flattening the power. It is not clear this is always so in other designs.

There are other vital tests that do not involve observation with the fast-neutron hodoscope. These include damage propagation between subassemblies and vapor explosions.

(b) Larger-scale Tests. In-pile tests on a larger scale, using a cluster of small-scale subassemblies, may be required to assess the potential for damage propagation between subassemblies; in-pile tests with at least one large subassembly are required to determine fuel motion, especially when incoherence is expected to be effective.

The vapor explosion possible upon reentry of sodium is mentioned in several contexts in this report and requires a program of its own; but the significance of these tests is clear. Tests on a cluster of specially instrumented small-scale subassemblies may be required to determine the extent of a vapor explosion. If large autoclave tests indicate a strong dependence on coherence, tests on a full-scale subassembly may be needed.

Propagation between subassemblies involves the questions: "Does the damage spread from one subassembly to its neighbors?" and "Does the propagated damage then give rise to further damage propagation?" The answer to the first of these can be obtained from the small-scale tests and tests on a full-scale subassembly with (perhaps) the neighboring subassemblies mocked up by partial subassemblies. The small-scale tests should also provide an answer, in part, to the second question. Confirmation can be obtained from out-of-pile tests using chemical explosives. Verification beyond this requires large in-pile tests requiring at least seven full-scale subassemblies.

Such tests would raise the question: "Are seven enough?" The problem is that propagation throughout the core from a single subassembly accident may itself involve the same sort of randomness that is apparently involved in a vapor explosion in a single subassembly. A conservative approach is to verify that a vapor explosion, if it can occur at all in a single subassembly, does not cause damage beyond the subassembly. It may not be possible to verify this, so that an extensive in-pile test program of multiple subassemblies may be needed. But out-of-pile tests would still be useful to guide the design of such in-pile tests.

(2) Test Conditions--Prolonged Overpower Transient. In the case of fuel meltdown from simulated power-flow mismatch from a prolonged overpower transient, the problem is mainly one of fuel damage thresholds. Tests of the TREAT "flattop" kind can ascertain these thresholds. If they are exceeded, the problem becomes similar to, but in an aggravated state, that of fuel meltdown from propagating fuel pin damage, as discussed in Section b below. Change in the power-flow ratio may cause severe changes in the time scale of events, requiring more large-scale tests to understand the different behavior.

(3) Test Conditions--Bubble Entrainment. Fuel meltdown from power-flow mismatch from a simulated bubble entrainment or large fission-gas release has been mentioned in Chapter III. Results from this program and FEFP results should yield an estimate of the potential danger from this source. In the case of fission-gas release, it may be that simultaneous release from a large number of pins may precipitate fuel melting, if such release is possible. If it is, full-scale subassembly tests can yield information about this process.

b. Fuel Melting from Propagating Fuel-pin Damage

Propagating fuel-pin damage, as discussed in Chapter IV, may cause fuel melting. The chief objective here is to determine the coolant flowrate and temperature as damage proceeds and to correlate this with safety-instrument response, to determine if there is sufficient time to reduce power before extensive damage occurs. Since wall effects are

important, tests on small-scale (37 to 61 pin assemblies) simulations are desirable to permit extrapolation of wall effects. Tests with programmed power can verify that safety actions are effective. Should coherence effects be important, tests on a full-scale subassembly ($\lesssim$271 pins) will be required. For damage that proceeds too rapidly for safety action to be effective, small-scale tests can ascertain (from postfailure examination) whether damage propagation between subassemblies is possible. In these tests, dynamic pressure measurements can be made to indicate possible vapor explosions, and flow measurements can be made to indicate flow-interference effects. Depending on the time scale indicated by the smaller tests, a full-size subassembly test may be required to estimate the net reactivity effect.

(1) Test Conditions. Preliminary tests using "flattop" transients and a few pins can isolate the basic mechanisms of single- or few-pin failure and indicate the scope of postfailure phenomena. These tests can cover the initial failure and its immediate consequences to neighboring pins, but they are basically limited to the time scale of a few seconds. Prolonged operation at steady-state conditions is not necessary to produce temperature distributions representative of steady state: heating times comparable to one or two time constants for heat transfer from the fuel are adequate for realistic distributions.[178]

Two TREAT loop experiments of this type have already been performed on EBR-II driver fuel. The short-term consequences of a loss of sodium bond incident in a fresh fuel pin were mocked up by running two argon-bonded pins, each in a flattop power pulse of about 6-sec duration.[179,180] Two features of the experimental program associated with these experiments are particularly relevant to this discussion: First, a foundation was laid for the investigation, by characterizing failure modes and thresholds under relatively idealized conditions,[138] measurement of cladding penetration rates,[181] confirmation of the use of the rate data for actual fuel pins in-pile,[182] study of sample failures in a channel-like environment using sodium loops,[81] and calculations of probable pin behavior. The two loop experiments on unbonded pins were then run to confirm the adequacy of the assumed models to predict that the pins would in fact survive. Second, the possible range of accident conditions was studied, so that the problem could be attacked systematically. The question of greatest immediate interest is whether a pin that is unbonded or loses its bond early in irradiation due to a defect will cause immediate damage to its neighbors. This was checked experimentally first, and then additional work was undertaken to study (1) the effects of continued steady-state exposure of the pin after its meltdown inside its cladding by running in a special EBR-II capsule, (2) the potential for damage propagation from loss of bond near the end of irradiation, and (3) refinement of both experimental and analytical techniques.

(2) Longer and Larger Tests. For experiments on single pins run in loops during flattop power transients near design power levels, a

TREAT type facility can be used for excursions of the order of several seconds. For example, a 13% enriched UO_2 pin (low enough enrichment that the thermal-neutron flux depression inside the pin does not unduly distort transient temperature distributions) can be run at a level of about 200 W/g of fuel inside a Mark-II integral TREAT loop* for about 15 sec. With an advanced reactor control system that would make essentially all the reactor capacity available for a flattop, this duration could be doubled to about 30 sec. However, the neutron spectrum in the test section must be hardened if experiments are to be done with mixed-oxide pins which absorb thermal neutrons strongly (particularly pins with uranium enriched for steady-state irradiation in a test reactor, as would be the case for irradiated pins from EBR-II, etc.).[183] The degree of spectrum change needed increases for clusters, if cluster power depressions are to be tolerable (say, within 10%). A single fully enriched mixed-oxide pin, partially shielded from thermal neutrons, could be run in the Mark-II loop for somewhat shorter durations (~12 sec or ~24 sec if the advanced control system is available). These times would be cut approximately in half for partially shielded clusters of seven fully enriched oxide pins. With a thermal-neutron converter region installed between the loop and the core,[184] maximum flattop durations approaching the limits for the partially shielded single mixed-oxide pin case become feasible for clusters larger than seven pins. At present, however, prospects for performing experiments on clusters of more than 37 pins appear dim in a TREAT-type facility. Within this limitation and the flattop power-duration limitation of several seconds, short-time propagation phenomena can be studied in such a facility.

c. Large-scale Reactivity Insertions

TREAT is the current major vehicle for a program of tests of the behavior of fuel under these conditions. Postfailure examination of TREAT tests shows dispersal of fuel with much fuel on the sides of the test section, and the possibility of extensive movement of molten fuel into blanket regions. The absence of decay heat in TREAT, however, leads to chilling the molten fuel prematurely so that postfailure examination may give a misleading picture of the final disposition of the fuel. Molten UO_2 is inviscid, and decay-heat simulation is required, particularly with regard to the possibility of secondary criticality. TREAT experiments also indicate strong wall effects. In the presence of proper environments, the fuel motion is expected to be even greater and perhaps more coherent. We need to determine wall effect and the effect of axial blankets on the flow of molten fuel.

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*The "second-generation" integral TREAT sodium loop designed to contain the high pressures that can be calculated for sodium-vapor explosions resulting from oxide fuel meltdowns. (See Ref. 157).

The modification of TREAT to simulate decay-heat should therefore be considered. Wall effects may be assessed from other programs.

d. Fuel Melting from Simulated Loss of Cooling When Shut Down

Although complete loss of cooling is extremely unlikely, consideration has been given to this accident under shutdown conditions, since there would normally be fewer safeguards available to keep the fuel from melting from decay heat. Where decay-heat capability can be obtained, that facility should serve to observe the meltdown of an array of a few pins in the absence of sodium.*

2. Safety Instrument Testing

As has been indicated several times in preceding paragraphs and in Chapter IX, a facility for full-scale tests is the natural place to demonstrate the effectiveness of safety systems involving instruments such as boiling detectors, fission-product detectors, and flowmeters. There are two chief reasons for this: First, the facility would be large enough and have the appropriate environment to accommodate typical instruments. Second, although the facility would be designed for the transient type of operation associated with safety-system tests, irradiation facilities such as EBR-II and FFTF require steady power with a minimum of interruptions to carry out their programs.

3. Program Summary

We describe a program of work to study the dynamics of melting fuel and its associated coolant under a wide variety of accident conditions. These conditions include: the initiation of meltdown from slow blockage by propagating fuel-pin damage, if this is possible; loss of flow from pump failure with delayed scram; sudden loss of flow through blockage of subassembly inlet orifice; and both slow and fast power transients. These experiments are designed to determine the effects that limit fuel-pin damage propagation when such propagation does occur, the causes of propagation of damage from subassembly to subassembly using small-scale subassembly tests, and realistic simulation of meltdown in a power reactor by tests involving afull-scale subassembly. The tests involve the maintenance of power in the subassembly for periods of time ranging from small fractions of a second to many seconds or minutes and possibly longer. Decay-heat simulation is also required.

The primary objective of these tests is the determination of the maximum realistic reactivity-insertion rates for accident analysis,

*A possible out-of-pile experiment can be imagined using an irradiated element.

particularly of super-prompt-critical accident analysis. A secondary objective is the determination of the possibilities of secondary criticality and the problems involved in emergency cooling. The planned tests are large enough to account realistically for the effects of incoherence and of damage propagation.

Another objective of these tests is to detect the possibility of damage to other core structure and to determine if planned core protective actions are sufficient. The program should test the effectiveness of engineered safeguards, including safety instrumentation. These tests can be carried out under realistic simulations in a facility that can accommodate full-scale subassemblies.

Out-of-pile tests similar to, but more generally applicable and more realistically simulating, in-pile conditions than those reported in APDA-155 are required to indicate the probable sequence of events and time scales to be expected in the larger subassembly-meltdown experiments. The objective here is to determine the way to use loop instrumentation most effectively.

APPENDIX B

Fast-neutron Hodoscope

A. DeVolpi

1. Introduction

Since a major objective of a large-scale test is to obtain as much information as possible from subassembly meltdown, a preliminary evaluation is made of the fast-neutron hodoscope in the role of acquiring this data.

To accomplish the evaluation on a specific basis, certain ground rules were assumed; undoubtedly, just as the ground rules affect the design of the hodoscope, they perhaps even more seriously affect the reactor concept. A recurring dialog will be required in order to find a condition of mutual optimization.

In terms of application of a hodoscope, latitude remains with regard to the various deduced limitations and constraints. The fact that experimental facilities are considered before reactor design is considered improves prospects for maximum utility.

This appendix is divided into two stages: the first reviews the fast-neutron hodoscope designed for TREAT, and the second contains both extrapolations and new considerations in application to large-scale tests.

Conclusions are stated in full at the end.

Summarizing briefly, we find by extrapolation from the TREAT fast-neutron hodoscope that a "radiation hodoscope" for large-scale tests is feasible. This device would not only monitor massive fuel movements in a large subassembly, but would also have separate instrumentation to deduce sodium void occurrence.

2. Fast-neutron Hodoscope for TREAT

a. Major Elements of the Fast-neutron Hodoscope

The hodoscope is designed for maximum sensitivity to fast neutrons produced by fission in the test capsule. Background consists of fast neutrons from the core, which scatter in the test capsule, in addition to residual gamma-ray and thermal-neutron effects. Signal/background ratios and spatial resolution are enhanced by minimizing extraneous materials and by providing a clear slot through the reactor.

We may conceptually consider the hodoscope in two parts. The front section consists of the collimator, neutron detectors, and a preliminary electronic-sending unit; the back section accomplishes the final electronic processing and ultimate digital readout.

b. General Basis for Analysis

The overall design of the TREAT hodoscope is founded upon a succession of studies supported by experimental data. Using the reactor characteristics and cross-section data, an initial proposal was developed in 1963. That report was based also on data generated from an earlier gamma-ray scanning device.

In the course of time, a large amount of experimental information derived from both static and dynamic tests at TREAT has been used to support the calculated estimates, supplemented with other tests conducted at Argonne. Over a dozen reports, mostly on an internal level, have been issued with a summary of these findings. There are also several published articles that describe the entire system with emphasis on the novel electronic and digital photographic readout features. Section 5 of this appendix represents a chronological list of references.

c. Specific Hodoscope Features for TREAT

The central characteristics of the fast-neutron hodoscope must be described in terms of spatial and time resolution, scanning area, and signal/background ratio.

Design resolution is 0.15 in., which has been accomplished, although this characteristic is used only for horizontal differentiation; vertical resolution is extended to 0.86 in. Such resolution is observed with bare fuel pins surrounded by a capsule containing about an equal thickness of extraneous material. An example of this and other features is provided by Fig. 14, which displays the results obtained by scanning a single detector across the 4-in. reactor slot.

The ability to distinguish fissionable material from sodium coolant and capsule is represented by the signal/background ratio. For the bare pin in Fig. 14, the signal from the 6% enriched pin is seven times stronger than the container background. As interceding material is introduced, the ratio drops until, for a capsule filled with nearly 2 in. of sodium, it becomes as low as two or three (see Fig. 15). In addition, horizontal resolution broadens, typically to about 0.25 in.

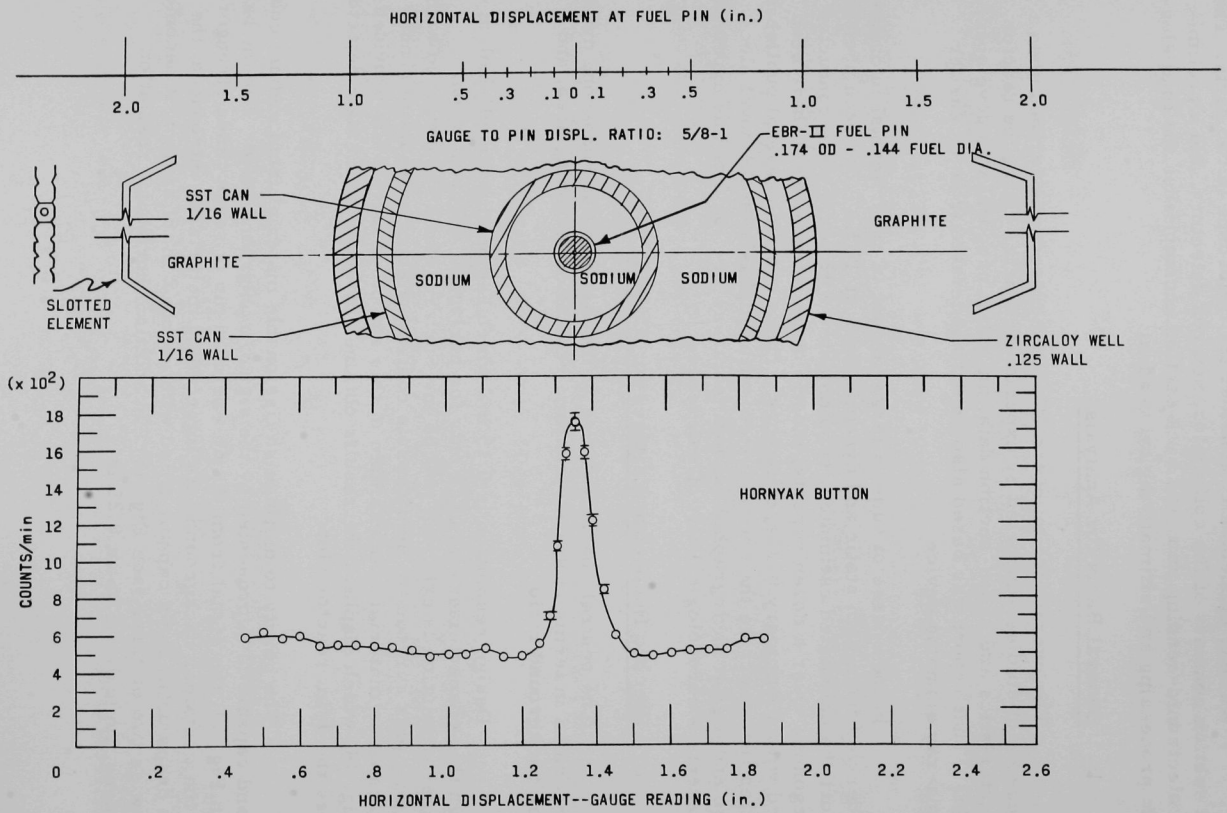

Fig. 14. Hodoscope Resolution--Bare Pins. ANL Neg. No. 112-4780.

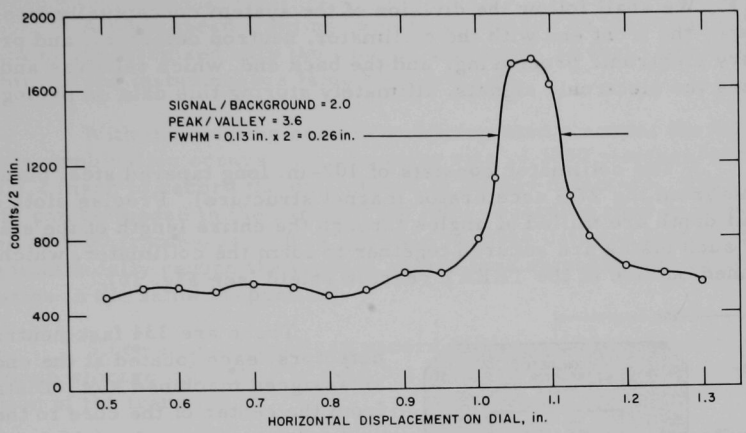

Fig. 15. Hodoscope Resolution--Pin in Sodium. ANL Neg. No. 113-1680.

Time resolution, which identifies ability to follow fuel movement, is intricately related to signal/background ratio, count rate, and readout response. In terms of accumulated data transfer to permanent storage, a time resolution of 1.2 msec is scheduled. With the present system, up to 24 sec of data may be recorded. Extending time resolution to 2.4 msec allows an equivalent increase in data-recording time.

Certain other characteristics are also important. The total area that the TREAT hodoscope can cover at any instant is 2 in. horizontally and 20 in. vertically; this is chiefly dependent on the number of parallel channels of detection. The TREAT hodoscope has 334 channels.

Each channel can tolerate a peak count rate of over 10^6 counts/sec. Such a high peak rate was built into the system to allow a wide dynamic counting range to follow the reactor from full power down to a small fraction. The system is capable of recording data at a sustained peak rate for the full time coverage.

Much of this information with regard to capability of the TREAT fast neutron hodoscope may be compressed by referring to a channel count sensitivity of 200 counts/(msec/cm^3) of fuel at 100 MW of TREAT reactor power for a 6% enriched pin viewed with 50% transmission of neutrons.

d. Specific Elements of the TREAT Hodoscope

Some of the devices used in the existing TREAT hodoscope should be reviewed before discussing the application of this instrument to

the STF. We shall follow the division of the system conceptually into two sections: the front end with the collimator, neutron detectors, and preliminary electronic processing; and the back end, which receives and digitizes the electronic signals, ultimately storing this data on photographic film.

The collimator consists of 102-in. long tapered steel plates (surplus from the ZGS accelerator magnet structure). Precise slots of 100-mil depth are milled at angles through the entire length of the slabs. Thirty such plates are secured together to form the collimator, which is positioned outside of the TREAT reactor shield (see Fig. 16).

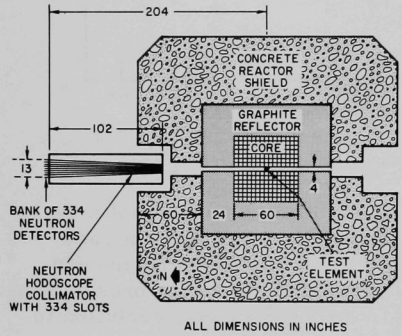

Fig. 16. Hodoscope Collimator at TREAT. ANL Neg. No. 113-1682.

There are 334 fast-neutron detectors, each located at the end of an assigned machined slot. Distance from the center of the core to the neutron detector is about 13 ft. The type of detector is the "Hornyak button," a scintillator working on the proton-recoil principle. Rather high inherent discrimination against gamma rays is obtained.

Each scintillator is coupled to a 3/4-in. photomultiplier tube; current pulses from the anode of the tube go directly to the back end of the system.

The collimator is capable of vertical and horizontal movement. A standard neutron source is used to calibrate the neutron detectors.

Signals from the front end of the hodoscope system are transported along coaxial cables about 50 ft to the main electronic racks located in an air-conditioned room.

The first rf-shielded rack receives the low-level pulses and carries out the functions of integration, amplification, level discrimination, and shaping. The output then consists of a digital pulse, which is associated primarily with high-energy neutrons. These pulses are forwarded to another rack where they are temporarily scaled and stored in integrated circuit binary registers. There are 334 such parallel channels.

The registers have a 4096-count capacity. Upon external command, the contents of each register can be shifted to a bank of lights, which simultaneously displays the binary total for up to 56 scalers.

A high-speed framing camera is focused on the lamp bank. The camera is synchronized with the readout command, so that each frame takes a picture of the instantaneous lamp bank contents.

With the existing camera and integrated circuits, the following readout combination occurs: At a framing rate of 5000 pictures/sec, it takes 1.2 msec to record the data stored in all 334 detectors. Since 2800 ft of film can be loaded in the camera, the total recording time is about 24 sec at 5000 frames/sec. It is possible to store data for longer periods and automatically reduce the camera speed, thus extending the time of observation in the same proportion.

Since each channel can record 4096 counts in 1.2 msec, the maximum storage capability is over 3×10^6 counts/sec sustained for the duration of the transient.

After development, the film is scanned at the ANL Applied Mathematics Division installation designated CHLOE. This automatically controlled, computer-coupled, flying-spot scanner can translate the recorded images onto magnetic tape.

The magnetic-tape information is then processed by a high-speed computer to recover the original count-rate data, which can then be subjected to further operations to secure an interpretation of the original transient event.

e. Operating Experience with the TREAT Hodoscope

The TREAT hodoscope has participated in a number of transients to test certain features of the design. At first, about 50 channels of the electronic system were available to follow fuel movement, making use of a temporary analog readout system. As a result of these experiments, (1) the count-rate capacity of each channel was gradually extended, (2) the capacity of the system has been expanded to the full quota of 334, and (3) a much more effective digital readout system was installed.

An analysis of test results is contained in RPD-EPM No. 54. One of the most informative meltdowns was Transient 917, in which a sodium-bonded EBR-II pin was destroyed in a transparent capsule. Because of the sodium cloud that formed upon cladding rupture, a major portion of the parallel color-photographic data was obscured. Nevertheless, enough data remained from various transducers to demonstrate a strong correlation with events interpreted from the hodoscope, which was able to derive useful information during the sodium-vapor expulsion stage.

Sufficient hodoscope data have been obtained from two meltdowns produced in the Mark-I sodium loop to warrant analysis. Transient 1007

involved a single 6% EBR-II pin. There are no optical data for the sodium loops; thus comparisons must be made with pressure transducers, thermocouples, flowmeters, and postmortem examination. Not only were the various correlations reasonable, but this was the first case in which evidence of sodium voiding was found by the hodoscope in the form of a reduction in "background."

Another interesting experiment in which the hodoscope obtained useful data was Transient 1048, in which the Mark-I loop contained all seven pins. In this case, the cluster stands out rather strongly with a resolution (full width at half maximum) of about 0.5 in. Because of the prominence of the fuel with regard to the background, an effectively large signal-to-background ratio of about four occurred. Once again, there were points of proper correspondence between the hodoscope and the other instrumentation. There was also evidence of sodium voiding for this transient. Incidentally, all the fuel was melted down.

As mentioned, the overall quality of hodoscope data has so far been limited by both an insufficient number of channels and by a temporary system of readout. Since these limitations have been remedied, the first full-scale measurements are now being planned.

3. Radiation Hodoscope for Large-scale Tests

The discussion that follows is predicated in part on certain ground rules, which are subject to readjustment. We shall focus on a single subassembly confined by a stainless steel tube of 6-in. diameter. We assume the subassembly will consist of many enriched pins of any type, with sodium coolant constituting the remaining filler.

The problem being faced, it is understood, differs from TREAT tests in two major ways: first, in the larger quantity of fissile material; and, second, in the massive failures expected.

Accordingly, an instrument more elaborate than the TREAT fast-neutron hodoscope is proposed. We shall temporarily classify it by a more general term "radiation hodoscope," for reasons that should become apparent in the following sections.

a. General

To observe the instantaneous position of fissionable material in the test capsule, reliance is placed upon unattenuated (and unscattered) neutrons generated from fission within the capsule. Neutrons from the driver must undergo scattering directly into the line of sight of the hodoscope; if these neutrons are initially below the hodoscope detection threshold, or if in

scattering they lose sufficient energy to drop below the threshold, then the electronic processing system can effectively exclude them from the count. At present, the threshold is set in the vicinity of 1-2 MeV. Gamma rays make a less important contribution to the background because the neutron detector discriminates against them well.

In terms of the signal-to-background ratio, the best conditions are obtained when there is little interceding material and when there is a clear slot through the reactor. Since, for realistic test conditions, it may be necessary to introduce a substantial amount of material around the test object, there will probably not be any latitude to sacrifice the clear-slot requirement.

Intervening material affects results in three ways:

(1) Neutrons of interest from the fuel are absorbed or scattered out, thus reducing the count rate.

(2) Neutrons that undergo small angle scattering can enter the "wrong" collimator slot, thus degrading spatial resolution.

(3) Neutrons from the reactor core can be scattered into the collimator, thus creating the primary source of uniform background.

Material in the line of sight behind the object of interest is primarily responsible for contributing to the third category. Tests have shown that placement of graphite moderator behind or in the vicinity of the test capsule is deleterious. In general, use of low-atomic-mass materials should be minimized in the collimator field of view.

Specific limits on the amount of material surrounding the fuel assembly cannot be established. The limits are partly qualitative, being connected with ultimate decisions about desired information. Some quantitative and semiquantitative guidelines can be established, though.

The degree of enrichment of the fuel will be an important factor, so will the flux pattern across the subassembly--peak, flat, or depressed.

TREAT experiments have shown that a cross section of 4 in. of material--chiefly sodium, Zircaloy, and steel--will still yield a signal/background of two. Whether lesser figures of merit can be tolerated would require a more extensive study.

The effect of steel inserted between the test capsule and the collimator has already been examined. Figure 17 depicts a set of distributions obtained from experiments at TREAT. About 1 in. of steel reduced

the signal/background by a factor of two with attendant degradation of spatial resolution. Recognizing the possible necessity of pressure vessels or other containment blocking the hodoscope view, this loss could be accepted if the design cannot be optimized for the best instrumental conditions. On the other hand, if the collimator and detectors can be placed within the pressure vessel, then this latter approach should be given preference. Using tungsten as the shielding material may make it possible to shorten the collimator to a few feet.

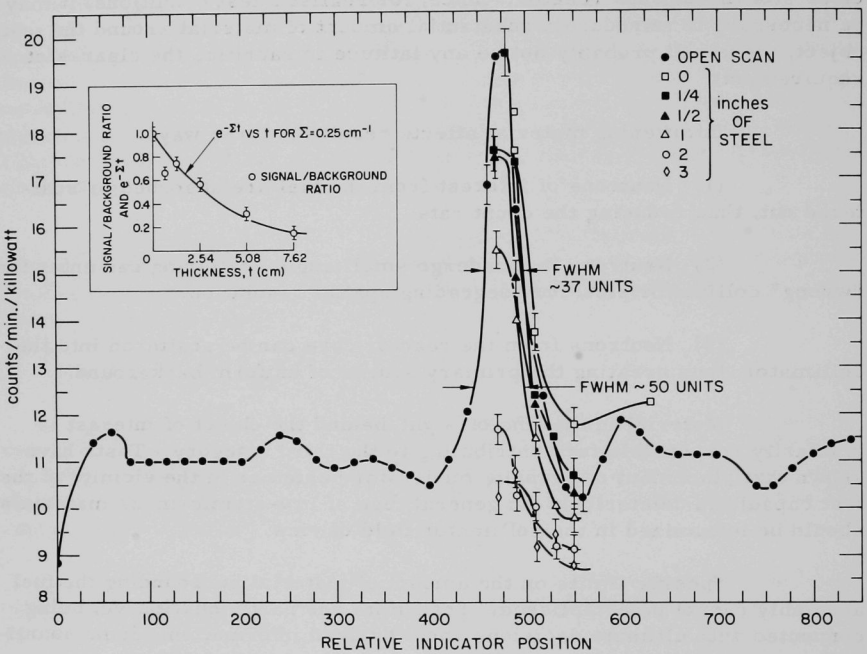

Fig. 17. Hodoscope Resolution as Affected by Steel

Some general characteristics have been outlined in a rather broad fashion: a clear slot and test assemblies designed with consideration of hodoscope limitations. In later sections, some other relevant factors will arise to justify the present constraints.

b. Occultation Effect

One major limitation of the hodoscope, inherent in the concept, can be attributed to occultation. Since a subassembly containing many pins is of primary concern and since each pin is a source of genuine fission

neutrons (depending on the flux and enrichment pattern), there is what one may call an occultation effect: a blocking of view caused by adjacent pins.

Suppose in a given transient only the central pin melts down. The movement of this fuel is then obscured by all the remaining pins, not just those in the line of sight. Clearly the amount of information that can be derived from this type of transient is dependent on the number of surrounding pins, the pin-arrangement pattern, the flux weighting, the signal/background ratio, the reactor power level, and spatial resolution.

Fortunately, it can be assumed that such single-pin meltdowns are of less interest than more massive failures. If many or all of the pins in a capsule are destroyed, the hodoscope will then be operating in the favorable mode of a large proportion of fissile material with a rather small background from extraneous matter. Accordingly, decisions with regard to the type of failure to be monitored must be directly fed back into the design criteria for the hodoscope.

To place this section in perspective, one should note for subassemblies being considered that in- and out-scattered neutrons produce degrading effects comparable with that caused by occultation.

c. Sodium-void Detection

The consequences of sodium voiding or bubble formation are important in the study of meltdown phenomena. It has already been observed in TREAT experiments that--inadvertently--there is some prospect of detecting the void effect. The word "inadvertently" is inserted because the original hodoscope design enhanced fission-neutron detection at the expense of background from materials like sodium. However, with sodium loops the signal/background ratio has been unavoidably reduced to two; or, to put it another way, the background mostly from sodium is about half of the fission-source strength. Consequently, the voids appear as reductions in relative background.

Advantage may be taken of this effect in two ways. First, the observation of voiding appears to be a consequence of reduced in-scattering of neutrons from the core. It should be possible to emphasize this effect with a separate detector which is more sensitive to lower-energy neutrons. Second, some examination of a gamma-ray detector to be used in conjunction with the hodoscope may be warranted.

It is likely that both of these detectors may be employed simultaneously; that is, doubling the capacity of the electronic processing system can enable concurrent detection of fission-fuel effects and sodium voiding to be accomplished.

There is another beneficial effect to be derived from a secondary data channel. Aside from voiding effects, the remaining material tends to scatter radiation uniformly and at a rate proportional to the reactor power. Thus, a "background" detector would help in continuously generating the correct nonfission component for subtraction from the fission-neutron channel; this alone is a valuable improvement in system capability.

d. Spatial Resolution and Scanning Area

The achievable spatial resolution is a characteristic intimately connected to signal/background ratio and to geometrical properties of the hodoscope. Another limit is imposed by interceding material.

The smaller the solid angle intercepted by the neutron detectors, the better will be the spatial resolution. The solid angle is chiefly a function of the dimensions of the collimating hole, the length of the collimator, and the distance between the test capsule and the detector.

If it is desired to allow for an area coverage of 12 in. across and 3 ft vertically, then, assuming limiting resolution is 0.25 in., a matrix of 48 x 144 = 6912 detectors could be applied. Possibly 1-in. vertical resolution would be acceptable, reducing the total number to 48 x 36 = 1728 detectors. This is already over five times the number used in TREAT. Since we can expect another detector behind the fission-neutron monitor, we are then talking about a tenfold increase in detectors. Should reactor design considerations or other factors reduce the horizontal scanning area requirements, then the number of desired detectors can be correspondingly reduced. Moreover, perhaps broader horizontal resolution is sufficient.

e. Time Scale and Resolution

Although current time resolution (1.2 msec for 24-sec excursion) may be adequate, some of the constraints can be discussed should other options be desired. The actual readout time for each scaler is 0.2 msec; this duration is dominated by the framing camera running at 5000 frames/sec. The particular camera in use (HYCAM) goes up to 10,000 frames/sec, and is likely that faster cameras can be adapted. Film-capacity problems can be overcome by brute-force methods (starting a second camera when the film load of the first camera nears depletion); thus, duration of observation is probably not a limiting factor.

Since most of the digital integrated circuit systems now have 20 MHz logic, there is no bound caused by this part of the circuitry. It may be possible--depending strictly on the type of detector and necessary linear processing--to provide for 10^7 counts/sec input rate.

Consequently, the major limitation for increases in time resolution is probably the financial cost. A second more fundamental limitation will arise from the minimum number of counts obtained in a given time increment. If only three counts are accumulated in this increment, one of which should be background (for signal/background = 2), then the statistics of random events begins to obscure the usefulness of the data.

f. Operating Conditions

Of highest value to the hodoscope is operation of reactor power at a sustained "flattop." If the driving flux does not perturb the experimental object, it greatly improves counting statistics to maintain the maximum fission rate in the test assembly as long as possible.

Whether or not stereoscopic viewing provides needed or optional data must be compared with further restrictions applied to the reactor design. For this consideration, a second hodoscope would be situated at right angles to the first. If molten fuel is constrained by the subassembly walls to move primarily in a vertical direction--up or down--then not much is to be gained from a second observation point. However, if detailed oscillations of great interest are expected, and they cannot be deduced from a two-dimensional picture, then allowance will have to be made for a second instrument rotated 90° about the axis of the subassembly. There is little point to detectors located along the subassembly axis, unless some special requirements develop.

Some mention has been made of the deleterious effects of thick materials placed between the subassembly and the hodoscope. Unless the collimator must be positioned within a vessel, it is generally best to move it some distance (10 or 20 ft, depending on other conditions) from the origin. The collimator can even be made an integral part of the reactor shield.

Occasions may arise to conduct in-core tests with fuel configurations that preclude use of the hodoscope. As situated at TREAT, we know of no present limitation on the regular program caused by having the hodoscope at the North facility. Presumably, similar arrangements will be made for STF; however, if there is any tradeoff between resolution degraded by the introduction of permanent barriers, this requirement should be carefully considered at the earliest design stage.

g. Supplementary Remarks Concerning Sensitivity

In Section 2.c of this appendix, the sensitivity of the present TREAT hodoscope was given as 200 counts/(msec/cm^3) of 6% enriched fuel*

*This pin is operating at a linear power of approximately 20 kW/ft.

for 100 MW of TREAT power, assuming 50% transmission loss. This applies to the detectors now in use. With a signal/background ratio of 2, we thus obtain about 50 counts from a 0.15-in.-diam pin and 25 counts from background every millisecond. It may be possible to improve the efficiency, if necessary; more important, though, better spatial resolution may be achievable by investigating threshold-neutron detectors.

h. Illustration of Radiation Hodoscope Response

Some coarse estimates may be made of the radiation hodoscope response to meltdown of a subassembly. Figure 18 is the subassembly pattern chosen for illustration. A total of 261 pins are enclosed in the 14-cm-diam capsule. The pins are up to 0.25 in. in diameter of UO_2 with 15% enrichment. Volume ratios are roughly 40% fuel, 10% stainless steel, and 50% sodium.

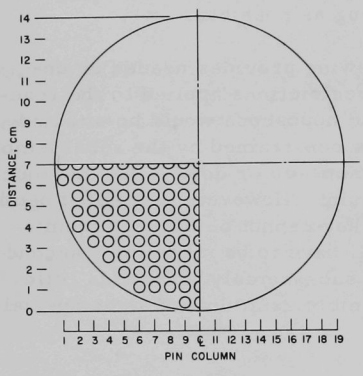

Fig. 18. Illustrative Large Scale Test Pin Distribution. ANL Neg. No. 113-1684.

For comparison with TREAT data, we compute an equivalent uranium or plutonium metallic-fuel content of 18.5% in this subassembly. For single-pin tests in the Mark-I sodium loop used at TREAT, the fuel/nonfuel ratio is only 0.3%. With seven pins clustered in the Mark-I loop, the ratio is 2.2%. These ratios are important in estimation of relative background from neutrons scattered by nonfuel portions of the capsule. Clearly, the STF subassembly pattern heavily loaded with fissile material in a tight lattice spacing can have significantly less in-scatter; nevertheless, the TREAT signal/background ratio of two and resolution of 0.25 in. have been applied in the following estimates, which should thus represent a worst case.

For the time being, a "removal" coefficient of 0.25 cm^{-1} has been taken as a basis of neutron loss through the subassembly. This number, derived from the data in Fig. 17, is experimentally observed for steel under certain limited detection conditions. Although the actual mixture in the subassembly is sodium and fuel, it is temporarily assumed that a homogeneous cross section of steel is adequate to represent neutron scattering. This too is probably a pessimistic assumption. In particular, restricting detection to supra-threshold neutrons may improve system definition.

The numbers being applied here are consistent with a TREAT hodoscope scan made of a 9-cm-diam container holding 37 EBR-II pins at 8% enrichment. A signal/background ratio of two was obtained (and peak/valley ratio of three). About seven pins were in the line of sight at the center of the cluster.

Subject then to these constraints, Figs. 19-23 depict a visual interpretation of instantaneous fuel position at one level of the hodoscope, which views a cross-section slice of the subassembly. The ordinate is the pin column (referring to Fig. 18); the abscissa is in relative count units, each unit being approximately equivalent to the response from a single bare pin irradiated in TREAT.

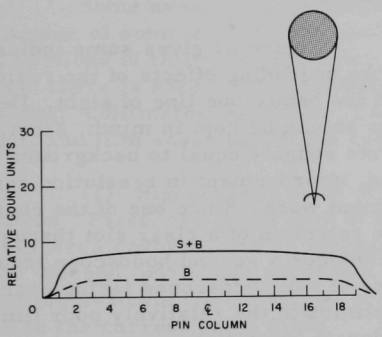

Fig. 19. Washout Effect from Occultation. ANL Neg. No. 113-1675.

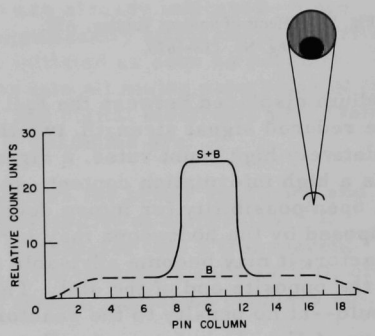

Fig. 20. Best Attainable Signal. ANL Neg. No. 113-1681.

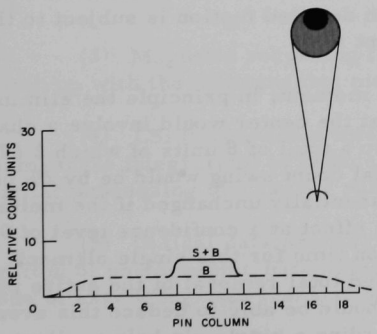

Fig. 21. Degrading Effects of Residual Sodium. ANL Neg. No. 113-1677.

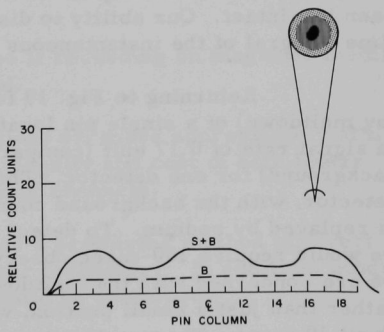

Fig. 22. Signal from Fifty percent Fuel Collapse. ANL Neg. No. 113-1678.

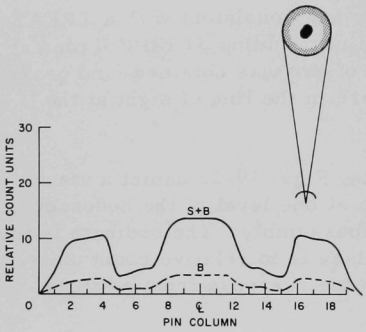

Fig. 23. Effects of Sodium Voiding. ANL Neg. No. 113-1676.

Figure 19 shows the washout effect caused primarily by occultation. The background is labeled B, and the gross count S + B. Figure 20 demonstrates the best attainable signal (under the above conditions) for a case in which all fuel has collapsed into a single column leaning against the front wall of the subassembly. Among other things, this picture indicates that the radiation hodoscope may require an order-of-magnitude extension in count-rate capacity.

Figure 21 gives some indication of the degrading effects of the residual sodium displaced between the fuel and the hodoscope line of sight. Despite the reduced signal strength, two things should be kept in mind: First, at relatively high count rates, a signal rate roughly equal to background level has a high information content; second, improvement in resolution remains an open possibility for future development work. Since one of the constraints imposed by the hodoscope may involve retention of a clear slot through the reactor, it may become advisable to consider a second hodoscope located at the opposite end of the slot. The use of two hodoscopes facing each other would--at no penalty to the reactor--eliminate the relatively poor limitations of Fig. 21 in favor of rather moderate limitations.

Figure 22 has been prepared to assist in the evaluation of response to less massive fuel accumulations. For this example, half the fuel pins have been collapsed toward the center and the remainder have been left intact. Our ability to discern detailed motion is subject to the time integral of the instantaneous count rate.

Returning to Fig. 19 for a moment, in principle the elimination (by meltdown) of a single pin located at the center would involve a change in signal rate of 0.17 unit (compared to a total of 8 units of which 2.67 is background) for one detector. The total count swing would be by 2% per detector, with the background count essentially unchanged if the melted pin is replaced by sodium. To detect this effect at a confidence level of 0.67, we would require 100-msec observation time for this single element. If we are concerned with the meltdown and total removal of the entire pin, rather than just a small portion, we should be able to deduce this event in about 10 msec. These estimates regarding a single pin help to illustrate the interaction between time and position information that applies to all examples.

Figure 23 illustrates some of the changes expected from major sodium voiding. This situation is the same as in the previous figure, except

that the sodium has been displaced by low-density vapor. Not only does the signal from the fuel stand out more clearly, but the background detector simultaneously follows the reduced scattering effect. In these figures, the background level has been depicted in its approximate proportion for the fast-neutron detector. Since it is possible to have parallel background detection, the precision in background may actually be better than the counting statistics derived from the fast-neutron channel. As a result, void effects can probably produce signal changes comparable to that obtained from equal volumes of fuel.

i. Directions for Early Development

Some avenues of development are already indicated. Since the outcome of some of this work could significantly affect design considerations, portions of this program should be initiated as soon as possible. In essence, there is need for efforts directed into all major categories of the hodoscope: collimator, detectors, linear and digital electronic processing, readout, and film scanning. Some specific examples are:

(1) High-energy neutron detectors for better efficiency and resolution. First, one might look at the possibility of dispersing ^{235}U microspheres covered with an inorganic scintillator into a transparent binder. Second, one might substitute a fission threshold such as ^{238}U, or alternately evaluate the efficiency of ^{238}U fission ionization counters operated in the current pulse mode.

(2) Sodium-void detectors. The ^{235}U detector mentioned above may be adequate for this purpose; otherwise, scintillators sensitive to neutrons or primarily to gamma rays may provide the best response to sodium.

(3) Magnetic recording to see if recording on magnetic media can catch up with the photographic method.

(4) LSI (large-scale integration) or, more appropriately MSI (medium-scale integration) to reduce the density and cost of both linear and digital processing of a larger number of channels.

(5) Optical data processing as a promising method of direct integration of photographic data using laser technology. In addition, other methods of expediting data translation should be examined.

4. Conclusions

An instrument that can follow massive movements of molten fuel from a contained subassembly of enriched pins is feasible. To obtain

reasonable resolution, a clear slot through the reactor is needed. The device that could accomplish these objectives represents an extension of concepts satisfactorily applied to the fast-neutron hodoscope at TREAT. For the program of full-scale tests, a larger "radiation hodoscope" is envisioned. This dual-function device would have peaked sensitivity for fission neutrons produced in the fuel and, in addition, separate channels for observation of sodium-void effects. To minimize the occulting effects caused by intermediate fuel pins in field of view, the location of hodoscope collimators at opposite ends of the clear slot is probably a reasonable compromise without further reactor design constraints.

Spatial resolution down to 1/4 in. may be possible; time resolution better than 1 msec can be achieved. Scanning area and duration of observation are compatible with experimental needs. A program of development starting at the earliest possible stage is advisable to ensure a system that is optimized with respect to the needs of data acquisition and facility design.

5. List of Related Publications

A. DeVolpi, K. G. A. Porges, and C. Rush, Subnanosecond RC Risetime from Ionization Counters, Bull. Am. Phys. Soc. 9, 46 (1964).

K. G. Porges, A. DeVolpi, and P. W. Polinski, note, Fast Cancellation of Gamma Radiation Background Pulses in Hornyak Buttons with an Active Circuit, Rev. Sci. Instr. 35, 1602 (Nov 1964).

A. DeVolpi and K. G. Porges, letter, Electron Excitation of ZnS(Ag), Nucl. Instr. Methods 36(2), 354 (Oct 1965).

A. DeVolpi and C. H. Freese, Fast Neutron Transient Detection Hodoscope for Reactor Fuel Meltdown Studies, IEEE Nuclear Science Symposium, San Francisco, October 1965, IEEE Trans. NS-13(1), 623-629 (Feb 1966).

A. DeVolpi, High-Speed, Multi-channel Data Recording for Nuclear Physics Experiments Using Fast-Framing Cameras, Proc. 7th Intern. Congress on High Speed Photography, Zurich, Switzerland, September 1965, Verlag, Dr. Othman Helwich, Darnstadt, pp. 516-521 (1967).

A. DeVolpi, R. Pecina, C. H. Freese, E. Rolnicki, D. Travis, and G. Larsen, Extended Capability for Fast Neutron Hodoscope Used at TREAT, Proc. IEEE Nuclear Science Symposium, IEEE Trans. Nucl. Sci. NS-15(1), 77-86 (1968).

A. DeVolpi, Digital Data Recording Using High-Speed Framing Cameras, Proc. 8th Intern. Congress on High-Speed Photography, Stockholm, June 23-29, 1968.

A. DeVolpi, The Recording of Transient Nuclear Data at 10^9 Counts/Second, Intern. Symp. Nuclear Electronics, Versailles, France, September 10-13, 1968.

APPENDIX C

Equation of State of Core and Associated Materials

D. Miller

1. Introduction

The primary source of a stress (or pressure) wave during a hypothetical fast-reactor accident is the deposition of energy in the core at a higher rate than it can be removed under steady-state design conditions or during transients defined by the normal operational variations of power, coolant flow, or coolant temperature.

The equation of state (EOS) is the general terminology for the relations among the thermodynamic variables, pressure P, temperature T, volume V (or density, $\rho = 1/V$), and energy E. It is also used to describe the material or constitutive relation between the mechanical properties such as stress σ, strain ϵ, and the thermodynamic variables. The following special names are given to different forms of the equation of state:

Thermal equation of state: $$P = P(V,T), \tag{1}$$

Caloric equation of state: $$E = E(V,T), \tag{2}$$

and

Mechanical equation of state: $$\sigma = \sigma\left(V, T, \frac{dV}{dt}\right), \tag{3}$$

where t is time.

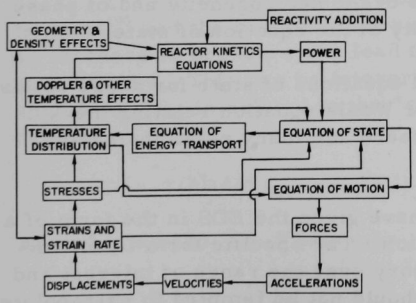

Fig. 24. Feedback Loop for Stress and Temperature Interactions with Reactor Dynamics. ANL Neg. No. 116-30.

The EOS is of importance in reactor safety studies because it links the chain-reaction kinetics to the equations of motion and of energy transport via the Doppler effect, thermal expansion, and mechanical distortions. Figure 24 indicates these interactions.

Because of the wide range of conditions that might be encountered in a hypothetical accident, consideration must be given to a wide variation of properties needed for its analysis. The materials of principal interest are the

sodium coolant, the structural materials (generally iron alloys), and fuel and fertile materials. Fuel and fertile materials may be metallic or ceramic, each with special characteristics. Although extensive experimental data are available for many of these materials, theory and empiricism must be used to extend the data and apply them.

The thermophysical properties necessary for a description of the phenomena include the conditions of phase transformations, such as the solid-solid phase transformations (as are found in iron alloys[185] and UO_2[186]), melting, and vaporization. These are characterized by temperatures and pressures, such as the melting points or boiling points, vapor pressures, and the associated volumetric and energy changes.

Other important properties include the heat capacities C_p and C_v (for constant pressure and volume, respectively*), or the corresponding enthalpies; the thermal-expansion coefficient $\alpha_p = (1/V)(\partial V/\partial T_p)$; the compressibility $\beta_T = (-1/v)(\partial v/\partial p)$; and its reciprocal, the bulk modulus of compression $B_T = 1/\beta_T$, which links the thermal and caloric equations of state to interrelated mechanical properties such as Poisson's ratio ν, the shear modulus G, and the elastic modulus E.

For severe transients, the properties listed above must be supplemented by the thermodynamic critical constants, the yield stress Y and its dependence on strain ϵ, temperature, composition, strain rate, and prior history, and dissociation energies of compounds and ionization potentials.

Description of the mechanical behavior should also include compaction and crushing[188,189] (the closing of pores and cracks), spallation[190] (the opening of cracks under dynamic tensile conditions), swelling or creation of porosity due to temperature-induced diffusion of fission gases,[191] and other nonlinear phenomena. Indeed the details of the heterogeneity and of phase transitions cause the principal complexity of the equation of state.

There are no simple generalized equations of state for dense solids and liquids such as the classical Van der Waals equation relating the coexisting liquid and vapor phases, but there are many relations having utility.

For convenience, many authors have given the EOS in the form of a polynomial or some other analytic function. The specific form is not important, as long as it fits the data or theory over the range of interest and is thermodynamically consistent. One should not be tempted to extrapolate these empirical equations outside their range of applicability.

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*At temperatures greater than about 3000°K, the electronic contribution to the heat capacity $c_v^e = c_e T$ becomes important[187] because of its temperature dependence. The coefficient c_e is usually determined from heat-capacity measurements at very low temperatures.

2. The Pressure-Volume-Energy Relationships

The path by which the final state is reached is important in considering the behavior of a material that can be raised to a high pressure and temperature by the addition of heat or by compression.

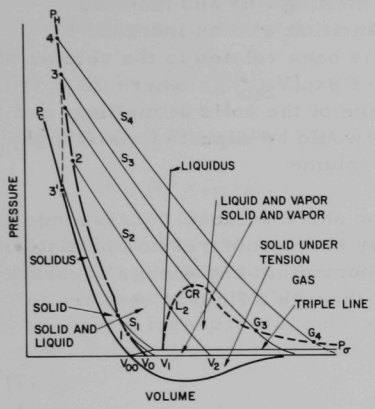

Fig 25. Plot of Pressure-Volume Equation of State. ANL Neg. No. 113-1644.

Figure 25 shows some relations between pressure, volume, and the temperature-dependent phase transformations. The curve labeled P_c is the "cold" compression curve at T = 0; it depends only on the volume through the elastic properties of the lattice at absolute zero. This curve can be derived from an appropriate microscopic model of forces between atoms or experimentally determined from compression measurements with appropriate corrections for temperature.

At any finite temperature there is a thermal-pressure contribution P_{th}, which consists of the effect of lattice motion P_L and of the electrons P_e. In equation form, this can be written as

$$P(V,T) = P_c(V) + P_{th}(V,T) = P_c + P_L + P_e. \tag{4}$$

The difference between points 3 and 3' in Fig. 25 represents P_{th}.

The curve P_c extends into the negative region, representing tension,[190] and the minimum corresponds to an unstable point beyond which an increase in pressure can lead to an increase in volume as the tension goes to zero. This can be interpreted as failure in tension and is related to spallation. P_c is related to the "cold" energy E_c by the thermodynamic definition

$$P = T(\partial P/\partial T)_V - (\partial E/\partial V)_T. \tag{5}$$

At absolute zero, Eq. 5 becomes

$$P_c = (\partial E_c/\partial V)_{T=0}. \tag{6}$$

The zero point of the energy can be taken at the minimum of P_c and considered as the cohesive energy or heat of sublimation to the atomic species.[190,192]

The [illegible] of material failure in tension and in expanded states is

The curve P_σ is the boundary of the saturated liquid and vapor region and consists of the liquid range from the triple point (the intersection of the liquidus line and the solidus line) to the critical point CR and the vapor region from the critical point to infinite volume. The solidus line is the melting curve that intersects the sublimation curve at the triple point. The change in volume at the intersection of the melting line and the cold-compression curve is typical of a phase transition with an increase in volume. The volume increase in melting has been related to the volume of the solid at melting by Stishov[193] as $\Delta V_m = f \exp(V_{Sm}/g)$, where ΔV_m is the volume change on melting, V_{Sm} is the volume of the solid at melting, and f and g are constants for each material. As would be expected, most high-pressure transitions display a decrease in volume.

The curve P_H, called the Hugoniot or shock adiabat, corresponds to the locus of all states that can be reached by shock compression of material initially at volume V_0. Equations of mass momentum and energy conservation for a shock wave of velocity U moving through a fluid at rest are, respectively,[187]

$$U(V_0 - V_1) = uV_0, \tag{7}$$

$$(P_1 - P_0)\, V_0 = uU, \tag{8}$$

and

$$E_1 - E_0 = \tfrac{1}{2}(P_1 + P_0)(V_0 - V_1), \tag{9}$$

where u is the material velocity and the subscripts 0 and 1 represent initial and final states, respectively. These relations may be legitimately applied to a solid only if the shock strength is such that shear stresses are negligible and the material responds in a "hydrodynamic" manner. The effect of deviatoric stresses in applying these equations below the Hugoniot elastic limit introduces errors.

An additional useful relation involving the enthalpy H can be derived with the same assumptions as

$$H_1 - H_0 = \tfrac{1}{2}(P_1 - P_0)(V_0 + V_1). \tag{10}$$

Over a wide range of shock compressions, within a single phase, the shock velocity and particle velocity are linearly related;[187,194] that is,

$$U = C + su. \tag{11}$$

The constant C is often identified with the sonic velocity C_0 at zero pressure as defined by

$$C_0 = \sqrt{-(dP/dV)_S} = \sqrt{B_S/\rho}\,,$$

where the subscript s indicates constant entropy. More correctly, it should be the velocity of the first plastic wave, but the difference is generally small; C_0 can be used if no experimental data are available.[195] The constant s can be related to other thermophysical properties.[195,196]

To calculate the pressure at a given volume from P_C (or from P_H), the Grüneisen ratio $\gamma = \gamma(V)$ defined by

$$\gamma = V(\partial P/\partial E)_V = -(\partial \ln T/\partial \ln V)_S \tag{13}$$

is used in an integrated form termed the Grüneisen equation of state.[187,194]

$$P - P_i = \frac{\gamma}{V}(E - E_i), \tag{14}$$

where the subscript i indicates some initial state. Since generally only small offsets from the cold-compression curve or Hugoniot are involved, the major errors involve the volume dependence of γ and the correct value following a high-pressure phase transformation. Lacking an appropriate microscopic model, the relationship independently derived by Rice[198] and Richter,[199]

$$\gamma = (V/V_0)\Big/\left(1 - V/V_0 + \frac{1}{\gamma_0}\right), \tag{15}$$

is recommended.

Equation 15 reduces to the commonly used form

$$\frac{\gamma}{V} = \frac{\gamma_0}{V_0} = \text{const} \tag{16}$$

at small compressions when $V/V_0 \cong 1$. The values of γ_0 can also be calculated from the thermodynamic identity

$$\gamma = \frac{V\alpha B_S}{C_P} = \frac{V\alpha B_T}{C_V}. \tag{17}$$

Agreement between statically and dynamically determined equations of state is good,[200,201] provided deviatoric stress components are considered.

Returning to Fig. 25, material at points off the cold compression curve will expand adiabatically to lower pressures if permitted to do so rapidly. For example, consider points 1-4 on the Hugoniot. Expanding along adiabat S_1 from point 1 will yield a volume V_1 at ambient pressure. This volume is larger than the initial volume V_0 by the amount of thermal expansion due to the waste heat corresponding to the area bounded by the Hugoniot,

the curve S_1, and the ambient pressure. A rarefaction wave due to reflection at an interface could intersect the tensile curve to the left of the minimum and would thus contract to a solid.

From point 2, expansion along S_2 will intersect the liquid curve at point L_2, and its final volume will be V_2. Whether melting would take place would depend on the relatively unknown details of the kinetics of phase transformations. The transit time of the stress wave can be short compared to the melting resulting in superheated solid, which might then melt. For adiabats of entropy greater than that at the critical point S_{cr}, such as S_3, the expansion will produce vapor, some of which will condense. If the expansion is along an adiabat of high entropy such as S_4, the expansion could produce supercooled vapor which would require nucleii to condense.

To construct the curves in Fig. 25, the cold compression curve can be given to adequate accuracy by the Murnaghan equation,[197]

$$P_c = \frac{B_{T0}}{B'_T}\left[\left(\frac{V_0}{V}\right)^{B'_T} - 1\right], \tag{18}$$

where B_{T_0} and B'_T are the coefficients in the expansion of the bulk modulus

$$B = B_{T0} + B'_T P. \tag{19}$$

In general, the temperature dependence is small.

The approximation

$$B'_T = B'_S \cong 2\gamma + 1 \tag{20}$$

is useful when accurate compressibility measurements to determine B'_T are not available.

Using the fact that there is a second-order contact between the Hugoniot and the cold-compression curve, one can find that

$$B'_T \cong B'_S = 4s - 1. \tag{21}$$

Combining Eqs. 20 and 21 yields

$$s = \frac{\gamma + 1}{2} \tag{22}$$

for use in Eq. 11. The Hugoniot is given by

$$P_H = \frac{C^2}{V}(1 - V/V_0)\Big/\left[1 - s\left(1 - \frac{V}{V_0}\right)\right]^2. \tag{23}$$

Table II lists measured or estimated values of γ, B_T', B_S', and s (Ref. 202) for a number of substances of reactor interest. Estimates for the thermodynamic critical properties of similar substances[105] are available. Also available are methods of estimating the vapor-liquid coexistence curves from limited data[106] and methods of estimating and extrapolating the thermal-expansion coefficient, compressibility, heat capacities, and heat of vaporization.

TABLE II. Constants in Equation of State

Material	ρ_0, g/cm³	B_T^0, kbar	B_T'	B_S, kbar	B_S'	θ, °K	γ	s	4s - 1
U	19.04	987	8.38			222	1.91	1.504	5.02
UO_2	10.97	500[a]	4.75[a]	470		154	1.88	1.438[a]	4.75
UC	13.61	1700[a]	5.22[a]	1636		320	2.11	1.555[a]	5.22
Pu	19.79	457	8.65			178	2.56	1.780[a]	6.12
Li	0.545	123	3.6			390	1.18	1.154	3.62
Na	0.971	74	3.95	67.5	3.85	150	1.17	1.242	3.97
K	0.87	37	3.85	30.8	3.77	93	1.30	1.188	3.75
Fe	7.84	1780	5.0	1667	5.04	467	1.60	1.58	5.32
W	19.35	3080	4.2			310	1.54	1.27	4.08
Ta	16.68	2005	5.3[a]	1897		230	1.68	1.214	3.86
Al	2.785	789	4.27	764	5.08	435	2.18	1.39	4.56
Cu	9.02	1310	5.20	1370	5.51	315	2.06	1.514	5.06
Hg (liq)	13.55	245.4	8.71	214	8.82	92	1.51	2.068	7.27
NaCl	2.02	234.2	5.35	247	5.27	326	1.54	1.541	5.164
SiO_2 (fused)	2.20	371	6.33	365	6.15	490	0.746	1.56	5.24
BeO	3.01	2430	5.04	2520		1280	1.97	1.36	4.44
MgO	3.65	1691	3.95	1717	3.915	946	1.60	1.223	3.892

[a]Estimated.

The temperature along the Hugoniot is found by integrating the thermodynamic identity

$$T\,dS = C_V\,dT + T(\partial P/\partial T)_V\,dV, \tag{24}$$

where S is the entropy.

The specific heat C_V is taken to be a function of volume and temperature in the following way. The temperature dependence of C_V is assumed in the Debye form,[203] and the Debye temperature θ_D is made volume-dependent using the relation

$$\gamma(V) = -\partial\ \ln \theta_D/\partial\ \ln V. \tag{25}$$

A code for performing the calculations and reducing the experimental Hugoniot data has been reported by Rogers.[204]

3. Temperature Dependence of Elastic Properties

Since the reactor accidents might initiate from operating conditions [illegible] under accident, the temperature dependence of the

physical properties is important. Anderson[205] examined the theoretical basis for the empirical relation of Wachtman et al.[206] In the form

$$B_S(T) = B_{00} - bT \exp(-T_0/T), \tag{26}$$

where B_{00} is the bulk modulus at absolute zero and T_0 is about half the Debye θ. He found the high-temperature slope b of the bulk modulus to be given by

$$b = 3R\gamma\delta/V_0, \tag{27}$$

where δ is the second Grüneisen ratio, independent of temperature, and is defined as

$$\delta = -\frac{1}{\alpha}\left(\frac{\partial \ln B_S}{\partial T}\right)_P .$$

Birch[207] showed that δ can be obtained from acoustic measurements as

$$\delta = (\partial B_T/\partial P)_T = B_T',$$

where

$$B_T' = B_S' + K,$$

where K is a small dimensionless constant about 1% of B_S'. Values of δ vary between 3.5 and 8 for substances of reactor interest.

Tottle[208] gave some information on the temperature dependence of Young's modulus of UO_2. Sung and Turnbaugh[209] estimated the temperature dependence of the bulk and shear moduli, but made no comparison to experiment.

4. The Effect of Material Strength

The strength of the material is important in determining whether the stress waves are one-dimensional or whether deviatoric components must be included. The point at which the behavior becomes hydrodynamic is taken as the yield stress Y. A dynamic value of Y determined in shock-compression experiments is termed the Hugoniot elastic limit (HEL). Experiments[210] in the ceramic material $A\ell_2O_3$ showed that the HEL is large and that shear stresses of magnitudes of about 30-40 kbar persist to shock-pressure levels of at least 300 kbar. Similar effects might be expected for UO_2.

A study[211] of the effects of stress gradient $d\sigma/dx$ on spallation gave the correlation

$$\sigma_S = Y + A\left(\frac{\Delta\sigma}{\Delta x}\right)^B, \tag{28}$$

where σ_S is the dynamic stress at spallation

Y is the static yield stress,

$\Delta\sigma/\Delta x$ is the stress gradient as calculated,

and

A and B are constants ($B \cong 1/2$).

The dependence of Y on strain rate is still an experimental and theoretical frontier, although data exist for some materials such as iron, steel, and aluminum.

5. The Effect of Heterogeneity

If the composition and geometry of a heterogeneous material are known and the elastic properties of each component are known, it should be possible, in principle, to calculate the pressure-volume relations of any mixture. In practice, this is most successfully applied to systems of regular or random geometry of simple, known properties.

Complications ensue with the presence of mixtures of phases such as liquids and vapors, which do not transmit shear and may not be at temperature equilibrium. Although in principle the equations of conservation of mass, momentum, and energy can be solved numerically to include the rate of approach to equilibrium with a sufficiently fine mesh, it is not currently practical to do so. Indeed, the total volume and energy of a mixture at the final point in <u>any</u> process in which temperature equilibrium is not reached is not sufficient to determine the final state.[212]

To describe a system with one or more dispersed phases in a continuous phase, the following four features must be considered:

1. The geometry of the dispersed phase(s)
   - a. Size and size distribution of the particles
   - b. Concentration and concentration distribution of the particles
   - c. Shape of the particles
   - d. Orientation of the particles
   - e. Topology of the particles

2. The composition of the continuous phase
3. The state of matter of the dispersed phase(s)
4. The composition of the dispersed phase(s)

6. Static Effects of Heterogeneity

The overall problem may be simplified by considering separately (A) systems with one continuous phase and (B) systems with two or more continuous phases, which are far more complex. Considering system A, the shape of the dispersed phase can often be idealized as spherical, cylindrical, or planar, although additional anistropy could be included by treating the shape as spheroidal or ellipsoidal. The simplest treatment for dilute dispersions is the representative cell. Here interactions between dispersed components are neglected and the system idealized, such as the effect of a spherical inclusion in a finite spherical cell.

For the elastic moduli, several theories have been proposed for spherical dispersions in a matrix and are listed in Table III along with the systems to which they apply and comments on their accuracy. Some of these theories are approximate; others[213] set exact upper and lower bounds on the properties. The bounds can be averaged suitably to give a good estimate of the true behavior.

TABLE III. Theories of Elastic Moduli of Composite Systems

Reference	Applicable Systems	Accuracy
Hashin[215]	Rigid spheres in soft matrix	Accurate--valid only at low concentrations
Reiner[216] and Hashin[215]	All	Accurate--low concentrations only
Kerner[217]	All	Approximate--high concentrations
Mackenzie[218]	Pores in a matrix	Accurate--low concentrations only
van der Poel[219]	All	Approximate--high concentrations
Guth[220]	Rigid spheres in soft matrix	Approximate--high concentrations
Hashin[221]	All	Sets upper and lower bounds

Hashin[214] showed that, for the effective Young's modulus E, these can be generalized in the form

$$\frac{1}{E} = \frac{1}{E_-} - \frac{1}{\lambda}\left(\frac{1}{E_-} - \frac{1}{E_+}\right), \tag{29}$$

where λ is a parameter that contains the ratio of the phase moduli and the geometry and

$$\frac{1}{E_-} = \frac{X_1}{E_1} + \frac{X_2}{E_2}$$

and

$$E+ = X_1E_1 + X_2E_2. \tag{31}$$

Here E- is the lower bound for equal Poisson's ratio, E+ is the upper bound for equal Poisson's ratio, and X_1 and X_2 are the volumetric fractions. The subscripts 1 and 2 denote the respective phases. Closer bounds for extreme values of the ratio of moduli require specification of the geometry.[214]

7. Application to Compressibility

Two widely used methods of averaging are the Ruess and Voight methods. For compressibility of a random elastic aggregate, Hill[222] showed that these averages are the upper and lower bounds for the true average compressibility β,

$$\beta_V < \beta < \beta_R. \tag{32}$$

The subscripts V and R indicate Voight and Ruess, respectively.

In the Ruess averaging method, stress is considered to be uniform through the aggregate, and

$$\beta_R = \sum X_i\beta_i, \tag{33}$$

where X_i is the volume fraction of material i. The difficulty with this model is that under pressure the distorted parts cannot fit together. This method is similar to that first proposed by Goranson et al.[223] and applied to shock compression and rederived periodically by others.

In the Voigt model, strain is assumed to be uniform throughout the aggregate, and

$$\frac{1}{\beta_V} = \sum \frac{X_i}{\beta_i}. \tag{34}$$

The error in this model is that the forces between adjacent regions are not in equilibrium.

The effect of spherical pores on compressibility can be given by

$$\beta = \beta_0(1 - k'p), \tag{35}$$

where the subscript 0 refers to the completely dense material, p is the porosity, and k' is an empirical constant, generally having the value 2 to 4. The constant k is related to the theoretical values derived by Hashin[221] for the bulk modulus $B = 1/\beta$

$$B = B_0\left(1 - \frac{3(1-\nu_0)(1-\eta)}{2(2\nu_0) + (1+\nu_0)\eta}C\right), \tag{36}$$

where

ν_0 is the Poisson's ratio of the compact matrix,

$\eta = B_1/B_0$ is the ratio of particle modulus to matrix modulus,

and

C is the volume concentration.

Similar relations are available for the other moduli. Poisson's ratio is relatively insensitive to pressure and temperature change although it goes to 1/2 at melting.

When spherical porosity is the included phase, Eq. 36 reduces to

$$B = B_0\left(1 - \frac{3}{2}\,\frac{1-\nu_0}{1-2\nu_0}p\right) \tag{37}$$

or

$$B = B_0(1 - kp). \tag{38}$$

Differentiating Eq. 38 produces

$$\frac{dB}{dP} = \frac{dB_0}{dP}(1 - kp) - B_0k\frac{dp}{dP} - B_0p\frac{dk}{dP}. \tag{39}$$

Both the second and third terms on the right side of Eq. 39 are small and can be neglected in the elastic region, and the equation simplifies to

$$\frac{1}{B}\frac{dB}{dP} = \frac{1}{B_0}\frac{dB_0}{dP}. \tag{40}$$

The physical implication of this equation is that the pressure coefficient of this elastic modulus (and others) determined on nonporous polycrystalline aggregates represents the corresponding quantity of the porous polycrystalline aggregate. Any departure, if observed, would correspond to the neglect of the two terms. This has immediate application to the use of Eq. 18.

Rossi[224] proposed a model for low concentrations that includes the shape of a second-phase inclusion. The effect of an oblate spheroidal inclusion was found to be more profound than that of prolate spheroids or spheres. This explains the high values of k' and k found for some porous ceramics.

Walsh[189] considered the effects of pores and cracks on compressibility of rocks and found that cracks have a much greater effect on compressibility than do spherical pores. He found that the pressure to close an elliptical crack P_c was

$$P_c = aE,$$

where a is the ratio of the minor axis to the major axis of the cracks. For many materials, E is of the order of 1000 kbar and crack porosity disappears at a few kbar; hence, a is of the order 10^{-3}. In both static and dynamic compression, the effects of porosity persist to high pressures (for porous iron greater than 20 kbar).

Static and dynamic compressions of rock materials[188] have shown hysteresis curves attributed to closing and opening of pores and cracks, and these have been incorporated in computer codes.[225] Several times the static yield strength is required to completely collapse the pores.

8. Dynamic Effects of Heterogeneity

The correct form of the shock Hugoniot for porous materials and its prediction from the Hugoniot of fully dense material are still a current research problem. Thouvenin[226] proposed a method to calculate the shock velocity in porous materials from a simplified theory called the "plate-gap" model. Here the porous solid is replaced by an array of parallel plates and gaps appropriate to the solidity κ,

$$\kappa = \rho_{0p}/\rho_0,$$

where the subscripts 0p and 0 relate to the initial density of the porous material and solid material, respectively. The shock velocity of the porous material U* is given by

$$\frac{1}{U^*} = \frac{\kappa}{U} + \frac{1-\kappa}{2u}. \tag{41}$$

Thouvenin omitted several steps, including assumptions, but showed good agreement for porous aluminum, copper, nickel, and uranium.

Thouvenin's model was extended by Heyda[227] who eliminated one of Thouvenin's assumptions by using the shock polar[228] of the solid material to find the material velocity in the crushed-up state U* and the pressure from conservation of momentum

$$P^* = \rho_{0p}U^*u^* = F(2u - u^*) = \rho_0(2u - u^*)[C + s(2u - u^*)], \tag{42}$$

and the final density from conservation of mass

$$\frac{\rho^*}{\rho_0} = \frac{\kappa U^*}{U^* - u^*}. \tag{43}$$

Agreement with experiments on aluminum, copper, lead, iron, nickel, and tungsten was generally very good, except for high porosities.

Thouvenin's approach has been criticized by Hofmann et al.[212] who claim it neglects the internal energy dependency[229] of the pressure in the porous material. This criticism was based on a calculation for aluminum that used an elastic-plastic code and assumed the Hugoniot jump conditions hold. Resolution of the disagreement between a fit to experiment and this calculation remains an open question.

Experimental information and estimated values for the equation of state are available[230] for U, Pu, UO_2, UC, Fe, Na, and some mixtures. The relations of Hashin fail for Young's modulus of porous cermets of UO_2-Mo.[231] This elastic modulus lies below the bounds for the boundary condition of perfect cohesion between phases but well above the values expected for phases with no cohesion, indicating imperfect wetting during sintering. The boundary conditions therefore must match the actual condition of the heterogeneous material.

The elastic moduli of UO_2 are not only functions of temperature, pressure, and porosity, but also of stoichiometry.[232]

9. Equation of State Used in Existing Computer Codes

The variety of expressions used in computer codes for accident analyses or for the propagation of elastic-plastic stress waves is almost as broad as the variety of codes. A few samples will be reviewed critically to indicate their applicability and potential inaccuracy.

Among the simplest expressions is the threshold equation of state, which is related to the Mie-Grüneisen equation.[233,234] This equation has the drawback[105] that the threshold energy is not defined by any clear-cut criteria, and this equation cannot be expected to be more than a simple curve fit over a narrow range. A modification of this equation using a vapor-pressure curve extrapolated many orders of magnitude beyond the experimental range has also been utilized.[234,235]

Empirical equations of state, such as the van der Waals have been used to describe vapor-liquid equilibria, but neglect the solid phase. Energy versus temperature and density relations based on application of the law of corresponding states comparing oxide fuels and metals to the corresponding properties of inert gases and organic materials have been used by Meyer and Wolfe[236] and others,[237-239] but no experimental or theo[illegible] given. The errors introduced by this formulation ha[illegible]

Little[240] used purely thermodynamic relations based on simple fits to a variety of approximate equations of state. Details of solid, liquid, and vapor phases are not included, and the code first predicts melting and considers purely hydrodynamic behavior.

Careful consideration of the thermodynamic and elastic-plastic equations of state were incorporated into the one-dimensional code PUFF.[241] Here attention was given to providing for the deviatoric components of the stress tensor and the effects of tensile and expanded states, and the possibility of a polymorphic phase transition is covered by allowing cusps in the curve fit to the compressed state Hugoniot.

Perhaps the most sophisticated and realistic equation-of-state treatment is given in the one-dimensional tensor code SOC[225] written for the IBM-7030 computer. This code includes details of phase transitions and opening and closing of cracks and pores with elastic-plastic deformation. Because of the extremely large size of this code and the general unavailability of the very large computer for which it was written, its principal application has been in describing underground explosions.

In general, these codes do not indicate concern for satisfying the requirements for thermodynamic consistency as pointed out by Jankus[242] and Wagner et al.[243] One such criterion is given in Eq. 5.

10. Experimental Determination of the Equation of State

The methods of static and dynamic compression generally used to determine the thermal and mechanical equations of state have recently been supplemented by acoustic methods. Precision measurement of the sonic velocity over a modest range of pressure and temperature enables extrapolation over a far wider range with some confidence by using a combination of theory and empiricism.[244,245]

Problems of phase changes in solid or fluid materials and macroscopic heterogeneity still represent major unknown areas. Bases for predicting the yielding of solids under dynamic conditions is also rudimentary. Phase changes in a homogeneous material can be studied by a variety of techniques, but the effects of heterogeneity are not as easily investigated. Of primary interest to reactor safety is the generation and propagation of stresses producing deformations that can change the reactor power. The development of stresses in the homogeneous fuel region is reasonably well understood, and bounds can be estimated for the effects of porosity. Phase changes generally cannot be predicted a priori, but the effects of pressure and temperature on the phase transition can be estimated once a phase transition is located.

Due to geometric sensitivity and acoustic impedance matching as well as elastic-plastic deformation phenomena, the details of stress propagation and attenuation in heterogeneous media are fraught with complications. Because of these complications, integral reactor experiments involving the destructive pulsing of major core segments are unlikely to yield useful equation-of-state information.

11. A Proposed Experiment

The best method of using reactors to generate equation of state information would probably be to use a pulsed reactor, such as TREAT, to drive an experiment based on a single fuel material to a very high temperature with a known energy input (which can be obtained by counting $^{137}C_s$ or other isotopes), measuring the heat content with a calorimeter to get the caloric EOS, and measuring the pressure generated by the vapor or by constraint of a massive cladding.

The melting point of tungsten at 3400°C and that of the mixed carbides of zirconium and hafnium at about 4000°C set limits on the temperature that can be contained for long periods of time, even if material compatability is not a problem. For very short periods of time, the finite time for a thermal wave to penetrate and melt through a thick wall would allow transient measurements to be made above the melting point of a thick-walled container. As a check on the effect of interaction between fuel and container, novel techniques are available. For instance, a depleted UO_2 crucible could be used to hold highly enriched UO_2 heated well above the melting point by fissions induced by the flux from a pulsed reactor. Since only a phase transition and no chemical interaction would take place, the effects of interaction in other container materials could be evaluated. The UO_2 crucible could be encased in a thin tungsten jacket to hold the assembly and be dropped into the calorimeter.

For the thermodynamic EOS a pressure measurement is needed, and the thrust from a vapor jet escaping through a small hole in the crucible could be measured. The pressure, aside from small corrections due to the momentum carried by the vapor jet, would be the thrust divided by the area of the hole. Performing the experiment in a vacuum and with the pressure of an inert gas would provide checks on the method. The temperature would be inferred from the calorimeter measurements mentioned above and the energy input, although optical pyrometer measurements might be developed for the temperature.

In the case of a highly constrained solid or fuel, the pressure must be inferred from the elastic deformation of the cladding at times short enough that thermal effects would not change the mechanical properties of the cladding. Complete constraint of a UO_2 sample in the maximum permitted TREAT transient is estimated to lead to pressures greater than 2 Mbars, more than enough to deform any container.

12. The Role of Integral Tests

If integral tests are to be performed because of the desire to establish overall damage limits, their great expense will always provide the temptation to add additional measurements to "milk" the event for all possible information.

Because of the complexities alluded to in the previous discussion, any integral test should be given special analysis and justification of effectiveness. Since the cost of such experiments is likely to be much larger and the possibility of decoupling phenomena to produce an easily understood result is smaller, alternative experiments should be considered before resorting to the inherently more complicated integral test.

In an integral destructive test of a segment or complete core, the aforementioned kinds of measurements of thermal and mechanical EOS probably could not be made in a precise and unequivocal manner. Although individual elements might be instrumented to provide information from a core that is to be destroyed for other purposes, these tests probably cannot be as useful as tests in an environment specifically designed to produce the information.

The potential difficulty of fitting equation-of-state parameters to integral tests can be illustrated by reference to the destructive test of the thermal reactor KIWI-TNT.[246,247] This cylindrical gas-cooled reactor was almost ideal for analysis and experiment because of its geometry, homogeneity, and accessibility. Because detailed information on the equation of state of graphite was available, total fissions and kinetic energy were calculated[247] before the experiment.

The calculated number of fissions was about three times as high as the experimental value, and the kinetic energy was about that predicted. It was necessary to modify the expansion and temperature coefficient of reactivity and the graphite equation of state to fit the data. A comparison[248] has been published about the relative changes in these influences of the equation of state, but the inadequacy of the computer model[247] was not examined.

The computer code RAC used in the analysis of the KIWI-TNT event treats the reactor as a series of concentric cylinders, but calculates interactions between regions as if they were slabs. It does not include the volumetric changes associated with phase transformations, although it treats the energetics. The equation of state has been adjusted on the basis of experiment to fit all these data, but there is no proof that this modified equation would fit another experiment on a similar core and no way is given to make the adjustments for another system a priori.

Unfortunately, the most valid experience on equation of state and ability to predict accurately the effects of composition, geometry, and structure is classified under weapons technology.

The numerical methods applied to these systems were used for bomb calculations,[249] and their success was attributed to the high temperatures and homogeneity of the systems considered. In addition, it was still found necessary to adjust the calculational procedures to fit the experiments.

Among the few experimental tests of coupled neutronics, mechanical movement, and equation-of-state calculations are studies of the GODIVA reactor.[249,250] GODIVA was a near-solid (few parting surfaces), unreflected sphere of 93.5% enriched-uranium metal, which could be pulsed by inserting a "burst" rod. Experimental results were the burst power profile as a function of mass, volume, and excess reactivity. The results were analyzed with a coupled "neutronics-dynamics" code[249] using the empirical equation of state

$$P = a + bT + cp,$$

where a, b, and c are constants and tensions of up to 300 kbar permitted. This equation of state is also used in AX-1 code.[251] The neutronics were treated with a one-dimensional, one-velocity, group-transport code normalized to experiment. The results were found to be "rather sensitive to the equation of state."[249]

An example of an analysis of the sensitivity of calculations to assumptions about the equation of state is given in the preliminary accident-analysis study[252] of FFTF. This study showed the sensitivity of the calculations to some of the parameters assumed.

13. Status of Equations of State of Core Materials

The status of the equation of state of homogeneous core materials is summarized in this section. Emphasis is placed on fuel materials, since the equation of state of iron is well known, although complex,[253] and work on iron alloys is still under review. The equation of state of sodium is known or can be estimated within engineering accuracy (±20%) in the coexistence region of liquid and vapor, and empirical equations[254] are expected to be useful in the gas region above the critical point, although the existence of a "Mott-transition" seen in mercury and cesium near the critical point[255] makes applying equations that work for a nonmetal speculative.

It is difficult to ascertain the accuracy of estimated values where no measurements or well-established theory or even empiricism exist. In many cases, long extrapolation of fragmentary data is required. The status will be summarized under three headings whose topics sometimes overlap:

Phase data--including transformation temperatures, pressures, volumes, and energies in the solid, liquid, and vapor phases, including ionization and dissociation.

Thermodynamic data--including heat capacities (electronic and vacancy effects) and enthalpies, thermal-expansion coefficients, and compressibility or bulk modulus.

Mechanical data--including elastic and shear moduli, Poisson's ratio, yield stress, etc.

a. Phase Data

(1) Uranium Metal. The high-temperature, high-pressure phase diagram has been studied,[256,257] and the triple point between the (α-β-γ) phases was determined. The complex set of phases makes the solid behavior unclear, and the effect of pressure on the melting point of 1406°K at 1 atm is small but known only over a narrow region. The heat of fusion is known as 1990 ± 130 cal/mole.[258]

The vapor pressure of uranium has recently been determined[259] to 2340°K, and thus a boiling point of 4690°K is determined that is 600°K higher than that previously estimated.[260] This indicates the problems of making accurate measurements at high temperatures and low pressures and of extrapolating low-pressure data over a long temperature range. The liquid density is considered known from the melting point to 1900°K,[261] although recent measurements over the narrow temperature range of 1410-1513°K are not in agreement.[262] Liquid density data, extrapolation of ideal gas density and the rectilinear diameter relation are used to estimate the critical properties and the vapor-liquid density coexistence curve.[105]

(2) Uranium Dioxide. The phase diagram of UO_2 as a function of temperature, density, and pressure is not known (although the temperature composition diagram of UO_2 is known at atmospheric pressure), and anomalies seen in heat capacity[263] and shock compression studies[177] indicate a fluorite-type transition. The change in volume on melting is poorly known,[264] although the heat of fusion now appears to be well established at 18.2 kcal/mole,[263,265] after early estimates and measurements ranging from 3.2 to 25.3.[266] The melting point for stoichiometric oxide is 3115 ± 15°K,[263,265] although it does depend on oxide content.

The vapor pressure has been studied extensively[267,268] and found to be sensitive to stoichiometry and impurities. Measurements are needed above 10^{-3} atm and well above the melting point to give confidence in the heat of vaporization and the extrapolation of the vapor pressure to the boiling point and beyond as functions of temperature.[105] The critical properties have been estimated, and the liquid and vapor densities were estimated to about ±30%.

(3) Uranium Carbide. The phase diagram of uranium carbide is less well known than that of UO_2 except for the melting point, a crude measurement of the volume change on melting,[269] and some low-pressure measurements of vapor pressure. The vapor pressure[270,271] is based on measurements at low pressures and is sensitive to stoichiometry and impurities such as oxygen. A solid phase transition is known in UC_2.

(4) Plutonium Metal. The phase diagram of plutonium has been studied extensively due to the variety of solid phases and the low melting point of 912°K. Volume changes[272-275] and vapor pressure are well known,[276,277] although the latter has been measured only up to a fraction of an atmosphere. Extrapolation gives a boiling point of 3550°K. Heats and volume changes of transition are known to adequate accuracy. The critical properties and coexistence curve have been estimated.[232]

(5) Plutonium Oxide. The melting point is known to be 2663 ± 20°K, and the boiling point has been estimated as 3433-3620°K from the extrapolation of low-temperature vapor pressure.[278,279] The heat of fusion has been estimated as 15.2-16.8 kcal/g atom by using regular solution theory with melting-point data on UO_2-PuO_2 mixtures.[280,281] Since the heat of fusion predicted for UO_2 by the same technique is in good agreement with that measured calorimetically, the value is believed to be within 5% of the correct value. Critical properties are not available.

(6) Plutonium Carbide. Very little information is available on plutonium carbide, other than its melting temperature of 1950°K and a limited amount of vaporization data. Its properties are expected to be dependent on stoichiometry in a manner similar to uranium carbides. When properties are unknown, the comparable values for uranium carbide could be used with errors of less than 15%. Critical properties have been estimated using an incorrect vapor pressure relation,[282] but are incorrect.

(7) Miscellaneous. Melting points and some vapor-pressure data are available for mixed uranium and plutonium oxides, carbides, and nitrides. Some information is available on some U-Pu-based alloys such as "fissium." Since these alloys are predominantly uranium, the phase changes generally are similar to those of uranium near its melting temperature, although some of the solid transitions are smeared out in the alloys. There is drastic shortage of data for molten phases, and it is recommended that an extensive program be implemented to fill this need for fuel materials of significant promise.

b. Thermodynamic Data

(1) Uranium Metal. The heat content of uranium has been determined to 1579°K,[258] high enough to establish the heat of fusion and the heat capacity of liquid but not the temperature dependence of the latter. It is

recommended that additional measurements of C_p should be made to 3000°K. Heat capacities and other functions are available for the ideal gas condition,[283] with sufficient energy levels included to be reliable to at least 6000°K and perhaps to 10,000°K. The ionization potential for the first level is accurately known, and this process is unimportant below 50,000°K. Other elemental substances are similarly treated.

No measurements of sonic velocity or liquid compressibility are available, and these should be measured over a wide enough range to establish the temperature dependence. Since the melting point is sufficiently low, this should not be very difficult by existing techniques.

The heat of vaporization could be extrapolated from a single point and the estimate of critical temperature using the empirical relation

$$\Delta H_v/RT_c = A_H(1 - T/T_c)^{B_H}.$$

Lacking additional information, B_H can be taken as 0.375 without introducing serious errors, and A_H can be fitted to the single point. This form has been found to fit the data for a wide variety of substances with only a small range of exponents.

(2) Uranium Dioxide. The heat content has been measured by two groups[263,265] to above the melting point with good agreement giving the heat of fusion and the liquid heat capacity. The latter can be used to extrapolate enthalpy and energy to within about 10% of the critical point before the critical singularity introduces significant errors. The effect of the singularity might be calculated from the lattice gas model of the critical point.[284]

Liquid and vapor densities and the critical constants have been estimated by several investigators.[105,236-239] These can be used to fit an empirical equation of state to get the liquid compressibility as a function of temperature. The temperature dependence of the thermal-expansion coefficient can be obtained from the liquid branch of the coexistence curve or from the empirical equation of state.

Ideal-gas heat capacities and thermodynamic properties could be calculated from vibrational frequencies of the molecule.[285] The frequencies are not yet available, but could be estimated. Deviations from the ideal state could be calculated from the empirical EOS or from the corresponding states table.[286] Heats of vaporization can be estimated as explained under uranium metal.

(3) Uranium Carbide. High-temperature enthalpies of uranium carbide measured by two workers, one to 1500°K and the other to 2500°K, are in good agreement in the region of overlap. However, heat capacity values near the melting point differ by about 10%.[271,287]

For $UC_{1.93}$, the enthalpy is known to 2600°K and the heat of transformation from α to β form is given as 2.57 kcal/mole after about a 50% increase in heat capacity from 1000°K to the transition at 2038°K.

Due to a lack of information on the liquid phase, the critical properties and coexistence curves have not been estimated. Thermal-expansion data are available for the solid phase.[269]

(4) Plutonium Metal. The thermodynamic properties of plutonium have been studied extensively[288] with a wealth of complex transitions up to the melting point. Expansion coefficients and compressibility are known to adequate accuracy in the solid phase, but only expansion coefficients in the liquid phase. Because of the low melting point, it would be desirable to extend the range of measurements to allow comparison with other metallic liquids.

(5) Plutonium Oxide. The enthalpy of solid PuO_2 has been measured to 1400°K for a near-stoichiometric sample,[289] and the derived heat capacity shows no abnormality. The thermal-expansion coefficient is available[290] to 1273°K, and solid compressibility can be estimated from mechanical properties. No data are available for properties above the melting point. Stoichiometric effects are expected to be important for solid properties. It is strongly recommended that experimental data be obtained above the melting point.

(6) Plutonium Carbide. The heat content of plutonium carbide has been measured to 1300°K,[291] and the derived heat capacity does not show the increase that UC and UC_2 display above 1000°. For comparison, the room-temperature heat capacity of PuC is just a little over half of that of PuO_2 and somewhat smaller than that of UC.

(7) Miscellaneous. Some enthalpy data exist for the important mixed oxides of uranium and plutonium to almost 2500°K,[292] along with thermal-expansion coefficients and mechanical properties from which compressibility data can be estimated. Additional data exist on alloys[293] and compounds such as nitrides, sulfides, selenides, arsenides, and phosphides.[294,295] There are no significant data in molten phase, except for a few plutonium-based alloys with low melting points which have been studied for molten fuel reactors. These gaps should be filled.

c. Mechanical Properties

(1) Uranium Metal. The directional elastic properties of single-crystal α-U,[296] as well as polycrystalline uranium, have been measured by sonic techniques to 923°K. These can be combined with thermal-expansion coefficients,[297] heat capacities, and densities to give the temperature and

volume dependence of the Grüneisen coefficient. Shock-compression data by French,[298] English,[299] Swedish,[300] and American[301] investigators are available and yield the solid equation of state. Some data are available on the dynamic yield stress of uranium metal and alloys.

(2) Uranium Oxides. An extensive body of experimental information is available on the mechanical properties of UO_2 including some shock-compression data.[186] These include the room-temperature elastic constants of single-crystal UO_2[302,303] and the temperature dependence (mostly near room temperature) and the effects of porosity and stoichiometry in these constants.[49, 304] Data extends to about 1300°K.[25] Poisson's ratio varies little with temperature. The Hugoniot elastic limit is less than 5 kbar.

(3) Uranium Carbide. Some data exist on the single crystal[305] and polycrystalline[306] elastic moduli of UC and their temperature dependence from near room temperature. Measurements have been made on both hypo- and hyper-stoichiometric UC to 873 and 1273°K, respectively. The latter material has a lower decrease in Young's modulus with temperatures. Good agreement is found between compressibility obtained by static compression and ultrasonically.[307]

(4) Plutonium Metal. Although there is a good deal of experimental data on solid plutonium[308] in its various phases, the accuracy and agreement between investigators is not satisfactory. Much of this can be ascribed to the self-heating and radiation damage from the spontaneous α decay of the samples yielding a dependence on prior sample history. Most of this work should be repeated using well-characterized samples. There is undoubtedly a good deal of work on mechanical properties of plutonium and its alloys that remains unavailable due to weapons interest.

(5) Plutonium Oxide. No published values of the mechanical properties of PuO_2 were found in this brief survey. Comparable properties of UO_2 or ThO_2 should be used until data are available.

(6) Plutonium Carbide. Properties similar to those of uranium carbide should be used.

(7) Miscellaneous. There is a sprinkling of miscellaneous mechanical property data for uranium and plutonium compounds, mixtures, and alloys available in the literature. A literature search on each individual system of potential interest is beyond the scope of this study.

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REFERENCES

1. D. Okrent, *Design and Safety in Large Fast Power Reactors*, Atomic Energy Review, Vol. VII, 2, pp. 135-223, IAEA, Vienna (1969). This review contains an extensive bibliography of work in fast-reactor design and related safety questions.

2. H. K. Fauske, *Liquid Metal Boiling in Relation to LMFBR Safety Design*, presented at Tenth National Heat Transfer Conference, Philadelphia, Pennsylvania, August 1968, published in C.E.P. Symposium Series, Vol. 65(92) (1969).

3. R. E. Holtz and R. M. Singer, "On the Superheating of Sodium and the Generation of Pressure Pulses," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

4. Le Gonidec *et al.*, "Experimental Studies on Sodium Boiling," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

5. G. Grass *et al.*, "Measurements of the Superheating and Studies About Boiling Phenomena in Liquid Metals," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

6. G. C. Pinchera *et al.*, "Experimental Boiling Studies Related to Fast Reactor Safety," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

7. R. E. Holtz and R. M. Singer, *On the Initiation of Pool Boiling in Sodium*, presented at Tenth National Heat Transfer Conference, Philadelphia, Pennsylvania, August 1968, published in C.E.P. Symposium Series, Vol. 65(92) (1969).

8. J. C. Chen, *Incipient Boiling Superheats in Liquid Metals*, Trans. ASME J. Heat Trans., Vol. 90, Series C, No. 3 (Aug 1968).

9. *Reactor Development Program Progress Report, October 1968*, ANL-7513, pp. 132-133.

10. R. E. Holtz, *The Effect of the Pressure-Temperature History upon Incipient-boiling Superheats in Liquid Metals*, ANL-7184 (June 1966).

11. J. B. Heineman, "Forced Convection Boiling Sodium Studies at Low Pressures," in *Proceedings of the Conference on Application of High Temperature Instrumentation of Liquid-metal Experiments, September 28-29, 1965*, ANL-7100, pp. 189-194.

12. J. A. Edwards and H. W. Hoffman, "Superheat with Boiling Alkali Metals," in *Proceedings of the Conference on Application of High Temperature Instrumentation to Liquid-metal Experiments, September 28-29, 1965*, ANL-7100, pp. 515-534.

13. B. S. Petukhov, S. A. Kovsler, and V. M. Zhukov, "Study of Sodium Boiling Heat Transfer," in *Proceedings of the Third International Heat Transfer Conference*, Chicago, Illinois, Vol. V, p. 60 (Aug 1966).

14. *Atomics International Quarterly Technical Progress Report, January-March 1967*, NAA-SR-12395.

15. G. C. Pinchera *et al.*, *Sodium Boiling Researches Related to Fast Reactor Safety*, Trans. ANS *11*(2), 691 (Nov 1968).

16. R. M. Singer and R. E. Holtz, *On the Role of Inert Gas in Incipient Boiling Liquid Metal Equipments*, Intern. J. Heat Mass Transfer *12*(9), 1045-1060 (Sept 1969).

17. K. T. Claxton, "The Influence of Radiation on the Inception of Boiling in Liquid Sodium," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

18. D. Smidt, "Sodium Boiling in Fast Reactors. A Review with Reference to the Ispra Conference," in *Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, November 7-9, 1968*, ANL-7520, Part I, pp. 12-16.

19. H. K. Forster and N. Zuber, J. Appl. Phys. *25*, 474 (1954).

20. S. A. Zwick and M. S. Plesset, J. Math. & Phys. *33*, No. 4, 308 (1955).

21. L. A. Skinner and S. G. Bankoff, Phys. Fluids *8*, No. 4, 1417 (1965).

22. P. Griffith, *Bubble Growth Rates in Boiling*, M.I.T. Technical Report No. 8, NP-6188, D-1 (1956).

23. T. Theofanous *et al.*, *The Rate of Bubble Growth in Superheated Liquid Sodium*, Trans. ANS *11*(2), 692 (Nov 1968).

24. K. H. Spiller, G. Grass, and D. Perschke, *Uberhitzung und Einzelblasenejektion bei der Verdampfung von Stagnierenden Flüssigmetallen*, Atomkernergie *12*, No. 3/4, 111-114 (1967).

25. R. C. Noyes, J. G. Morgan, and H. H. Cappel, *TRANSFUGUE-I, A Digital Code for Transient Two-Phase Flow and Heat Transfer*, NAA-SR-11008 (July 1965).

26. R. C. Noyes, H. Lurie, and A. A. Jarrett, "The Development and Growth of In-core Voids due to Boiling during Fast Reactor Transients," in *Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, October 11-14, 1965*, ANL-7120, pp. 881-889.

27. D. R. MacFarlane, *Transient Sodium Boiling Calculations-II*, Nucl. Eng. Design *6*, 103-114 (1967).

28. D. Logan *et al.*, "Boiling Liquid Metals and Two-Phase Flow Investigations," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

29. W. G. Meinhardt, *Calculating Vapor Growth in a Superheated Liquid Metal Within a Heated Channel*, Trans. ANS *9*(2), 566 (Nov 1966).

30. W. Peppler *et al.*, "Sodium Boiling and Fast Reactor Safety," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

31. G. Grass *et al.*, *Das Sieden von Flussigen Alkalimetallen*, Zeitschrift Atomkernergie, 12 Jahrg., pp. 101-110 (1967).

32. *Reactor Development Program Progress Report, October 1968*, ANL-7513, pp. 131-132.

33. A. W. Cronenberg, H. K. Fauske, and S. G. Bankoff, *Simplified Model for Sodium Coolant Expulsion and Reentry*, Trans. ANS *12*(1), 364 (June 1969).

34. R. E. Henry and H. K. Fauske, *Propagation Velocity of Pressure Waves in Two-Phase Mixtures and the Dependency on Flow Regimes*, Trans. ANS *11*(1), 364 (June 1968).

35. R. E. Henry, M. C. Grolmes, and H. K. Fauske, "Propagation Velocity of Pressure Waves in Gas-Liquid Mixtures," in *Proceedings of the International Symposium on Research in Cocurrent Gas-Liquid Flow*, Sept 1968, Waterloo, Canada.

36. R. M. Singer, *The Expulsion of Liquid from a Rapidly Heated Channel*, ANL-7337 (May 1967).

37. H. K. Fauske, "Two-Phase Compressibility in Liquid Metal Systems," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

38. S. J. Board, *Experimental Observations of Vapor Bubble Growth in Various Constrained Geometries*, Central Electricity Generating Board Report No. RD/B/N1418 (July 1969).

39. A. M. Judd, "Boiling and Condensation of Sodium in Relation to Fast Reactor Safety," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, France, Sept. 19-22, 1967.

40. K. H. Spiller, D. Perschke, and G. Grass, *Überhitzung und Einzelblasenejektion von Stagnierenden Natrium*, Atomkernergie *13*, No. 4, 245-252 (1968).

41. R. E. Holtz and R. M. Singer, *On the Superheating of Sodium at Low Heat Fluxes*, ANL-7383 (Nov 1967).

42. T. Theofanous, L. Biasi, H. S. Isbin, and H. K. Fauske, *A Theoretical Study on Bubble Growth in Constant and Time-Dependent Pressure Fields*, Chem. Eng. Sci. *24*, 885-897 (1969).

43. T. Theofanous, L. Biasi, H. S. Isbin, and H. K. Fauske, *Non-Equilibrium Bubble Collapse--A Theoretical Study*, Chem. Eng. Prog. Symp. Series (in print).

44. K. Y. Narasimham, G. S. Rosenberg, and M. W. Wambsganss, Jr., *Coolant-structure Coupling during a Rapid Insertion of Energy*, ANL-7502 (Oct 1968).

45. L. J. Hamilton and V. E. Schrock, *Propagation of Rarefaction Waves Through Two-Phase, Two-Component Media*, Trans. ANS *11*, No. 1, 367 (1968).

46. H. K. Fauske, Argonne National Laboratory, private communication (Dec 1968).

47. R. D. Carlson and R. E. Holtz, *Design and Testing of a High-heat-flux Electron-bombardment Heater*, ANL-7431 (March 1968).

47a. J. H. Field, *Experimental Studies of Transient Effects in Fast Reactor Fuels: Series I: UO_2 Irradiations*, GEAP-4130 (1962).

48. J. E. Hanson, J. H. Field, and S. A. Rabin, *Experimental Studies of Transient Effects in Fast Reactor Fuels: Series II: Mixed Oxide (PuO_2-UO_2) Irradiations*, GEAP-4804 (1965).

49. J. E. Hanson and S. A. Rabin, *Experimental Studies of Transient Effects in Fast Reactor Fuels: Series III: Pre-Irradiated Mixed Oxide (PuO_2-UO_2) Irradiations: Interim Report--Steady-State Irradiations*, GEAP-4819

50. J. E. Hanson and J. H. Field, *Experimental Studies of Transient Effects in Fast Reactor Fuels: Series III: Pre-Irradiated Mixed Oxide (PuO_2-UO_2) Irradiations: Final Report--Transient Irradiations*, GEAP-4469 (1967).

51. G. R. Thomas and J. H. Field, *Response of Axially Restrained Pellet Fuel to Transient Overpower*, Trans. ANS *10*, 707 (1967).

52. G. R. Thomas and J. H. Field, *Destructive Examination of an Axially-Restrained Pelleted Mixed-Oxide Specimen That Failed During Transient Overpower Conditions*, Trans. ANS *11*, 697 (1968).

53. C. E. Dickerman, L. E. Robinson, E. S. Sowa, and S. B. Skladzien, *Studies of Fast Reactor Fuel Element Behavior under Transient Heating to Failure: II. In-pile Experiments on UO_2 Samples in the Absence of Coolant*, ANL-6845 (Jan 1965).

54. J. H. Gittus, *Contribution to the Theory of Deformation and Fracture*, TRG-Report-1293(S) (1966).

55. J. H. Gittus, *Heterogeneity in the Dislocation Assembly: Its Effect Upon the Probability of Fatigue Failure*, TRG-Report-1294(S) (1966).

56. J. H. Gittus, *A Review of the Likely Fatigue Behavior of Irradiated Stainless Steel, Etc.*, TRG-Report-1353(S) (1966).

57. L. E. Robinson, C. E. Dickerman, and F. L. Willis, *Initial (7-Pin) EBR-II Mark I Scale-Up Cluster Fast Reactor Safety Integral Loop Experiments in TREAT*, Trans. ANS *10*, 697 (1967).

58. E. P. Hicks and D. C. Menzies, "Theoretical Studies on the Fast Reactor Maximum Accident," in *Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, October 11-14, 1965*, ANL-7120, pp. 671-685.

59. S. G. Lipsett, *Explosions from Molten Materials and Water*, Fire Technology *2*, 118 (1966).

60. G. Long, *Explosions of Molten Aluminum in Water--Cause and Prevention*, Metal Progr. *71*, 107 (May 1957).

61. P. E. Hess and K. J. Brondyke, *Study of Explosions of Molten Aluminum in Water*, paper presented at Metals Congress, Detroit (Oct 1968).

62. W. Nelson and E. H. Kennedy, *What Causes Kraft Dissolving Tank Explosions?* Paper Trade Journal *140*(29), 50 (July 1966).

63. J. A. Sallack, *An Investigation of Explosions in the Soda Melt Dissolving Operation*, Pulp and Paper Magazine of Canada *56*(10), 114 (1955).

64. J. Dugone and D. D. Wieland, *Fuel Plate Experience During the SPERT-1 Destructive Test Series with an Aluminum-clad, Plate-type Core*, IDO-16885 (July 1963).

65. R. Zeissler, *Non-destructive and Destructive Transient Tests of the SPERT-1D, Fully Enriched, Aluminum-plate-type Core: Data Summary Report*, IDO-16886 (Nov 1963).

66. J. R. Huffman, W. E. Nyler, and F. Schroeder, *Quarterly Technical Report SPERT Project, January, February, March, 1963*, IDO-16893 (May 1963).

67. *Kinetic Studies of Heterogeneous Water Reactors, Annual Summary Report 1965*, STL 372-30, TRW Systems, p. 62 (Dec. 30, 1965).

68. *Kinetic Studies of Heterogeneous Water Reactors, Annual Summary Report 1966*, STL 372-50, TRW Systems, p. 24 (Dec 1966).

69. *Reactor Development Program Progress Report, February 1969*, ANL-7553, pp. 113-116.

70. R. O. Ivins and L. Baker, "Calculation of the Pressure History of a Water-cooled Reactor during an Excursion," in *Chemical Engineering Division Semiannual Report, January-June 1964*, ANL-6900, pp. 270-280 (Aug 1964).

71. F. E. Brauer, N. W. Green, and R. B. Mesler, *Metal-Water Explosions*, Nucl. Sci. Eng. *31*(3), 551 (1968).

72. L. C. White, *Heat Transfer from a Sphere to Liquid Sodium during Forced Convection*, ANL-7296 (Jan 1967).

73. D. Swift and P. Krause, "Fuel-Coolant and Cladding-Coolant Interaction Studies," in *Chemical Engineering Division Semiannual Report, January-June 1965*, ANL-7055, pp. 199-206 (Oct 1965).

74. D. Swift and J. Pavlik, "Fuel-Coolant and Cladding-Coolant Interaction Studies," in *Chemical Engineering Division Semiannual Report, July-December 1965*, ANL-7125, pp. 187-193 (May 1966).

75. J. C. Hesson *et al.*, "Interactions of Fuel, Cladding, and Coolant," in *Chemical Engineering Division Semiannual Report, July-December 1967*, ANL-7425, pp. 159-163 (May 1968).

76. *Reactor Safety Development during 1969*, internal report.

77. *Reactor Development Program Progress Report, November 1968*, ANL-7518, pp. 123-125.

78. T. A. Doyle, R. B. Nicholson, J. B. Nims, E. M. Page, and J. T. Hagstrom, *Two Facility Approach to Investigations of Fast Reactor Meltdowns*, APDA-151 (Jan. 15, 1963).*

79. D. Okrent and H. H. Hummel, *Reactivity Coefficients in Large Fast Power Reactors*, ANS monograph (1970).*

80. H. A. Bethe and J. H. Tait, *An Estimate of the Order of Magnitude of the Explosion when the Core of a Fast Reactor Collapses*, UKAEA-RHM (56)/113, 1956.

81. C. E. Dickerman, R. Purviance, L. E. Robinson, W. Stephany, and F. L. Willis, *Summary and Analysis of TREAT Sodium Loop Experiments on Behavior of Single EBR-II Mark I Pins Under Accident Conditions*, Nucl. Eng. Design *7*, 442 (1968).

82. P. R. Huebotter, *Study of Fast Reactor Meltdown Accidents Using Simulant Materials*, APDA-155 (Aug. 31, 1963).

83. D. V. Gopinath and C. E. Dickerman, *Calculations of Coherence of Failure for Hypothetical Meltdown Accidents in an EBR-II-like Reactor*, ANL-6844 (March 1964).

84. C. N. Kelber, "Coherence Effects in Super-prompt Critical Excursions," in *Reactor Physics Division Annual Report, July 1, 1967 to June 30, 1968*, ANL-7410, pp. 334-335 (Jan 1969).

*These references contain substantial comments on the problem of fuel meltdown.

85. D. B. Sherer, W. A. Sangster, and F. W. Sciacca, "Alternate Core Design Parameters and Economic Trade-offs for a 1000-MWe Sodium-cooled Fast Reactor," in *Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, November 7-9, 1968*, ANL-7520, Part II, pp. 358-368.

86. P. M. Murphy, D. B. Sherer, and A. S. Gibson, "The Effect of Using Different Safety Criteria and Selected Safety Studies for a 1000-MWe Sodium-cooled Fast Reactor," ibid., pp. 345-357.

87. R. B. Nicholson, *Methods for Estimating the Energy Releases in Hypothetical Reactor Meltdown Accidents*, APDA-150 (Dec 1962).

88. *Liquid Metal Fast Breeder Reactor Design Study*, Vol. II, Fig. 3.3.3(a), GEAP-4418 (1964).

89. R. A. Daane, *Fast Reactor Safety Studies Report No. 2*, NDA 14-55 (1954).

90. W. J. McCarthy, Jr. and D. Okrent, "Fast Reactor Kinetics," Chapter 10 of *Reactor Physics and Control*, edited by T. J. Thomson and J. G. Beckerle, M.I.T. Press (1964).

91. P. Greebler and B. A. Hutchins, "The Doppler Effect in a Large Oxide Reactor--Its Calculation and Significance for Reactor Safety," in Proceedings of IAEA Seminar, *Physics of Fast and Intermediate Reactors*, Vol. III, 212 (1962).

92. R. Michel and L. L. Kintner, *Revised Report on Containment of Fermi Reactor Plug*, Allis-Chalmers Report 46-15-130 (June 6, 1961).

93. P. Azzoni, A. Carpini, L. DeCicco, and J. Randles, *A Contribution to the Theory of Uncontrolled Prompt Critical Excursions in Liquid Metal Coolant Fast Reactors*, J. Nucl. Energy *23*, pp. 315-330 (1969).

94. W. J. McCarthy, R. B. Nicholson, D. Okrent, and V. Z. Jankus, *Studies of Nuclear Accidents in Fast Power Reactors*, Second Geneva Conference, Paper 2165 (1958).

95. A. F. Henry and A. V. Vota, *WIGL2--A Program for the Solution of the One-Dimensional, 2-Group, Space-Time Diffusion Equations Accounting for Temperature Xenon and Control Feedback*, WAPD-TM-532 (Oct 1965).

96. K. Ott and D. Meneley, "Accuracy of Quasi-Static Treatment of Spatial Reactor Kinetics," in *Proceedings of the Brookhaven National Laboratory Conference on Industrial Needs and Academic Research in Reactor Kinetics*, BNL-50117, p. 497 (1968).

97. *Liquid Metal Fast Breeder Reactor Program Plan*, Vol. 9, Physics, WASH-1109 (1968).

98. R. W. Hardie and W. W. Little, Jr., *1DX--A One Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections*, BNWL-954 (March 1969).

99. N. M. Greene and C. W. Craven, Jr., *XSDRN: A Discrete Ordinate Spectral Averaging Code*, ORNL-TM-2500 (July 1969).

100. C. E. Till, R. A. Lewis, and R. N. Hwang, "ZPR-6 Doppler Measurements and Comparisons with Theory," in *Proceedings of the International Conference on Fast Critical Experiments and Their Analysis, October 10-13, 1966*, ANL-7320, pp. 319-333.

101. *Southwest Experimental Fast Oxide Reactor Development Program First Quarterly Report, April-June 1964*, GEAP-4594. See also the many subsequent SEFOR reports.

102. N. Hirakawa, *MARS, A Two-Dimensional Excursion Code*, APDA-198 (1967).

103. J. C. Carter *et al.*, "SAS1A Computer Code as a Tool for Fast Power Breeder Reactor Safety Analysis," described in the *Reactor Physics Division Annual Report, July 1, 1967 to June 30, 1968*, ANL-7410, pp. 278-282 (Jan 1969).

104. *Liquid Metal Fast Breeder Reactor Program Plan*, Vol. 10, Safety, WASH-1110 (1968).

105. D. Miller, "A Critical Review of the Properties of Materials at High Temperatures and Pressures Significant for Fast Reactor Safety," in *Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, October 11-14, 1965*, ANL-7120, pp. 641-653 (1965).

106. D. Miller, A. B. Cohen, and C. E. Dickerman, "Estimates of Vapor and Liquid Density and Heat of Vaporization of Alkali Metals to the Critical Point," in *Proceedings of the International Conference on Safety of Fast Reactors*, Aix-en-Provence, Sept. 19-22, 1967.

107. A. Judd and D. Schoeberle, *The Himpan Equation of State for Sodium and Cesium*, ANL-7637 (to be published).

108. Y. Chang, J. Gvildys, and S. H. Fistedis, *Two-dimensional Hydrodynamic Analysis and Code for Primary Containment*, ANL-7498 (Nov 1969).

109. G. Cinelli, Jr., J. Gvildys, and S. H. Fistedis, *Inelastic Response of Primary Reactor Containment to High-energy Excursions*, ANL-7499 (Nov 1969).

110. *Containment of Fragments from a Runaway Reactor*, Technical Reports 5, 6, and 7; SRIA-43, -113, -117; AEC R&D Report, Stanford Research Institute (Sept. 18, 1963, Dec. 2, 1963, and Feb. 15, 1964).

111. *Scaling Laws for Fragments from Explosives*, BRL-TR-7413 (1962).

112. D. E. J. Samuels, "Design Factors Influencing the Containment of Internal Explosion in Fast Reactors," in *Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, October 11-14, 1965*, ANL-7120, pp. 734-740 (1965).

113. J. F. Proctor, *Adequacy of Explosion-Response Data in Estimating Reactor-Vessel Damage*, Nucl. Safety *8*, 6, pp. 565-572 (Nov-Dec 1967).

114. W. R. Wise, Jr. and J. F. Proctor, *Explosion Containment Laws for Nuclear Reactor Vessels*, NOLTR-63-140 (Aug. 16, 1965).

115. *Comparison of Two Sodium-cooled 1000-MWe Fast Reactor Concepts, Task I Report of 1000-MWe LMFBR Follow-on Work*, GEAP-5618 (June 1968).

116. R. E. Aronstein, E. B. Ash, D. F. Casey, S. Golan, and R. J. Lancet, "Atomics International 1000-MWe LMFBR Follow-on Study Progress Report," in *Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, November 7-9, 1968*, ANL-7520, Part II, pp. 171-214.

117. W. K. Ergen (Ed.), *Emergency Core Cooling: Report of Advisory Task Force on Power Reactor Engineering Cooling*, USAEC (no reference number

118. T. H. Row, *Interactions of Molten UO_2 and Refractory Material*, ORNL-TM-1828 (June 1967).

119. G. W. Keilholtz and G. C. Battle, Jr., *Fission Product Release and Transport in Liquid Metal Cooled Fast Breeder Reactors*, Nucl. Safety *9*, 6, pp. 494-509 (Nov-Dec 1968).

120. A. W. Castleman, Jr., I. N. Tang, and R. A. Mackay, "Fission Product Behavior in Sodium Systems," in *Alkali Metal Coolants, Symposium Proceedings, Vienna, 1966*, pp. 729-742, IAEA, Vienna (1967); paper also available as BNL-10727 (CONF-661110-1) (Nov. 2, 1966).

121. W. P. Kunkel and G. Schlesinger, "Fission Product Retention and Plating Behavior in Liquid Sodium Systems," in *International Symposium on Fission Product Release and Transport Under Accident Conditions, April 5-7, 1965, Oak Ridge, Tennessee*, CONF-650407, Vol. 1, pp. 73-93 (1965).

122. A. A. Jarrett (Ed.), *SRE Fuel Element Damage*, NAA-SR-4488, pp. IV-C-1 to IV-C-5 (Nov. 30, 1959).

123. W. P. Kunkel, D. M. Elliott, and A. S. Gibson, *High Temperature Experiments on Fission Product Retention in Sodium*, NAA-SR-9287 (Feb. 1, 1965).

124. R. S. Hart and C. T. Nelson, *Fission Product Retention Characteristics of Sodium at High Temperatures*, NAA-SR-Memo-8712 (June 1, 1964).

125. W. P. Kunkel, *Fission Product Retention in Sodium--A Summary of Analytical and Experimental Studies at Atomics International*, NAA-SR-11766 (Aug. 8, 1966).

126. W. P. Kunkel *et al.*, *LMFBR Fission Product Contamination and Control Studies--A Status Report*, AI-AEC-12687 (June 30, 1968).

127. Atomics International, AEC Unclassified Programs, *Quarterly Technical Progress Report, July-September 1966*, NAA-SR-12175 (Nov. 25, 1966).

128. Atomics International, AEC Unclassified Programs, *Quarterly Technical Progress Report, October-December 1966*, NAA-SR-12296 (Oct. 25, 1967).

129. C. T. Nelson, R. L. Koontz, and L. Baurmash, *Characteristics of Vaporized UO_2 Aerosols in Air*, Trans. ANS *10*(2), 698-699 (Nov 1967).

130. F. L. Horn, *Some Aerosol Properties of Vaporized Fast-Reactor Fuels*, Trans. ANS *10*(2), 699 (Nov 1967).

131. C. Descoups and J. Rygaert, "Behavior of Fission Products in the Atmosphere Surrounding a Sodium Cooled Fast Reactor-Accidental Coolant Fires," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, Sept. 19-22, 1967.

132. H. A. Morewitz, M. Silberberg, G. N. Lauben, R. P. Johnson, and R. N. Cordy, "Sodium Burning, Aerosol Generation and Control of Fission Product Release During an Accident," in *Proceedings of the International Conference on the Safety of Fast Reactors*, Aix-en-Provence, Sept. 19-22, 1967.

133. Atomics International, *Annual Technical Progress Report--AEC Unclassified Programs--Fiscal Year 1966*, NAA-SR-12010, p. 241 (Aug 1966).

134. *Liquid Metal Fast Breeder Reactor Program Plan*, Vol. 4, Instrumentation and Control, WASH-1104 (Aug 1968).

135. G. F. Popper and A. E. Knox, *FARET In-core Instrument Development*, ANL-7161 (July 1966).

136. J. S. Hochheiser, *Liquid Metal Temperature Measurement (Sodium) State-of-Art Study*, LMEC-MEMO-68-10 (May 1968).

137. D. L. Rall and S. W. Bovarney, *Development of Temperature Instrumentation for EBR-II Fuel Pins*, ATL-188 (1959).

138. C. E. Dickerman, E. S. Sowa, D. Okrent, J. Monaweck, and L. B. Miller, *Studies of Fast Reactor Fuel Element Behavior under Transient Heating to Failure: I. Initial Experiments on Metallic Samples in the Absence of Coolant*, ANL-6334 (Aug 1961).

139. C. E. Dickerman, E. S. Sowa, J. H. Monaweck, and A. Barsell, *In-Pile Experiments on Meltdown of EBR-II Mark I Fuel Elements in Stagnant Sodium*, Nucl. Sci. Eng. *18*, 319-328 (1964).

140. D. F. Schoeberle, J. Heestand, and L. B. Miller, *A Method of Calculating Transient Temperatures in a Multiregion, Axisymmetric, Cylindrical Configuration. The ARGUS Program, 1089/RE248, Written in FORTRAN-II*, ANL-6654 (Nov 1963).

141. L. D. Montgomery, *A New Technique for Experimental Investigation of High-temperature Thermal Diffusivity of Uranium Dioxide under Transient Conditions*, ANL-7116 (Nov 1965).

142. R. C. Strohm, *Instrumentation for the Power Burst Facility (PBF) Fuel Rod Reaction Force Test*, Instrument Development Branch Report 1967, IDO-17269, pp. 55-61 (Aug 1968).

143. C. F. Reinke, L. E. Robinson, and R. O. Ivins, "High Temperature Thermometry--Service and Related Experience," Paper 22, *High Temperature Thermometry*, WASH-1067 (March 1966).

144. L. E. Robinson, C. E. Dickerman, R. Carlander, C. August, and C. Mueller, *Behavior of Uranium Sulfide Fast-Reactor-Type Fuel Specimens Under Transient Heating in TREAT*, Nucl. Appl. *1*, 168 (1965).

145. C. E. Dickerman, G. H. Golden, and R. Stewart, "Transient In-pile Tests on W-UO_2 Cermet Rocket Fuel Samples," in *Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966*, ANL-7210, pp. 244-245 (Dec 1966).

146. A. Smaardyk, C. J. Divona, R. H. Olp, and E. Hutter, *Design and Analysis of Instrumented Subassembly System for EBR-II*, ANL-7423 (June 1968).

147. S. S. Fam, L. C. Lynnworth, and E. H. Carnevale, *Ultrasonic Thermometry for LMFBR Systems--Phase I Report*, NYO-3906-4 (Sept 1968).

148. L. C. Lynnworth *et al.*, *Ultrasonic Thermometry for Nuclear Reactors*, IEEE Trans. Nucl. Sci. *NS-16*, No. 1, 177-183 (Feb 1969).

149. C. E. Dickerman, G. H. Golden, and L. E. Robinson, *In-Pile Photographic Studies of EBR-II Mark I and Fermi Core A Sample Meltdown*, Nucl. Sci. Eng. *14*, 30 (1962).

150. C. E. Dickerman and L. E. Robinson, *Photographic Experiments on Meltdown of Irradiated Metallic Fuel Pins*, Trans. ANS *7*, No. 1, 137 (1964).

151. C. E. Dickerman, L. E. Robinson, E. S. Sowa, J. H. Monaweck, and C. August, *TREAT Meltdown of Irradiated Metallic Fast-Reactor Fuel in an Inert Atmosphere*, Nucl. Appl. *3*, 461 (1967).

152. B. P. Hildebrand, *Acoustic Holography for Nuclear Instrumentation*, Trans. ANS *11*, No. 2, 633-634 (Nov 1968).

153. C. E. Dickerman, R. T. Purviance, and A. B. Cohen, *Analyses and Significance of Initial EBR-II Mark I Cluster Fast-Reactor Safety Integral Loop Experiments in TREAT*, Trans. ANS *10*, No. 2, 698 (1967).

154. M. C. Glass and G. F. Popper, *The Calibration and Stability of a 1200°F Permanent Magnet In-Core Sodium Flowmeter*, IEEE Trans. Nucl. Sci. *NS-15*, No. 1, 37-40 (Feb 1968).

155. M. C. Glass and G. F. Popper, *The Design and Performance of a 1200°F Magnetic Flowmeter for In-Core Application in Sodium-Cooled Reactors*, IEEE Trans. Nucl. Sci. *NS-14*, No. 1, 342-347 (Feb 1967).

156. J. F. Boland, R. D. DeForest, R. O. Ivins, F. S. Kirn, H. Lawroski, and R. C. Liimatainen, *Sensitivity of Pressure Transducers to Transient Neutron and Gamma Radiation*, Trans. ANS *5*, No. 1, 196 (1962).

157. L. E. Robinson, "Design of the Mark-II Integral Transient Reactor Test (TREAT) Facility Sodium Loop," in *Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966*, ANL-7210, pp. 245-250 (Dec 1966).

158. K. A. Dietz (Ed.), "SNAPTRAN Instrumentation Environmental Testing, Installation and Test Results," in *Step Project Quarterly Technical Report, July 1965-September 1965*, IDO-17165, 28 (1966).

159. V. T. Berta, *SNAPTRAN Experimental Measurements and Instrumentation*, IDO-17205 (1967).

160. L. E. Robinson and R. T. Purviance, "Pressure Transducers for TREAT Sodium Loops," in *Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965*, ANL-7110, pp. 279-280 (Dec 1965).

161. L. E. Robinson and R. T. Purviance, "Transient Response of Stand-off Pressure Transducer Assemblies on the Transient Reactor Test (TREAT) Facility Integral Sodium Loops," in *Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966*, ANL-7210, pp. 254-258 (Dec 1966).

162. *Reactor Development Program Progress Report, December 1967*, ANL-7403, p. 30.

163. J. R. Folkrod, *Design and Performance Characteristics of a Miniaturized Null Balance Fission Gas Pressure Transducer and Controller*, ANL-7573 (to be published).

164. *Handbook of Reliability Engineering*, U.S. Bureau of Naval Weapons (June 1964).

165. *Minuteman Electronic Part Failure Rates*, D2-14134, Boeing Co. (July 1966).

166. *Minuteman Electronics Parts Failure Mode Percentage*, D2-15358-1, Boeing Co. (July 1964).

167. *Reliability Stress and Failure Rate Data for Electronic Equipment*, MIL-HDBK-217A (Dec 1965).

168. W. L. Headington, M. E. Stewart, and J. O. Zane, *Fault Tree Analysis of the PBF Transient Rod Drive System*, IDO-17274 (Nov 1968).

169. O. B. Monteith, *Preliminary Reliability Analysis for FFTF Safety Circuits*, BNWL-661 (March 1968).

170. I. M. Jacobs, *Reliability of Engineered Safety Features as a Function of Testing Frequency*, Nucl. Safety *9*, No. 4, 303-312 (July-Aug 1968).

171. H. Kartluke, *Ultrasonic Instrumentation for Detection of Incipient Boiling in Liquid Metals*, NYO-3622-20 (July 1968).

172. D. E. Wiegand, *Summary of an Analysis of the Eddy-Current Flowmeter*, IEEE Trans. Nucl. Sci. *NS-15*, No. 1, 28-36 (Feb 1968).

173. D. E. Wiegand and C. W. Michels, *Performance Tests on an Eddy-Current Flowmeter*, IEEE Trans. Nucl. Sci. *NS-16*, No. 1, 192-195 (Feb 1969).

174. D. E. Wiegand, *The Eddy-current Flowmeter: An Analysis Giving Performance Characteristics and Preferred Operating Conditions*, ANL-7554 (Aug 1969).

175. H. W. Slocomb, *Liquid Metal Pressure Measurement (Sodium) State-of-the-Art Study*, LMEC-MEMO-68-7 (March 1968).

176. T. J. Crocker, *FFTF Sodium Service Pressure Measurement: State-of-the-Art Report*, BNWL-882 (Dec 1968).

177. J. E. Hanson, J. H. Field, and S. A. Rabin, *Experimental Studies of Transient Effects in Fast Reactor Fuels: Series II. Mixed Oxide (PuO_2-UO_2) Irradiations*, GEAP-4804 (1965).

178. C. E. Dickerman, *Use of Present TREAT Core as a Fast-Flux Loop-Meltdown Facility*, Trans. ANS *9*, No. 2, 551 (1966).

179. F. L. Willis, C. E. Dickerman, P. Henault, R. Purviance, J. F. Boland, and R. Noland, *TREAT Experimental Study of Survival of EBR-II Fuel Pin in a Loss-of-Bond Incident*, Trans. ANS *11*, No. 2, 697 (1968).

180. C. E. Dickerman, R. R. Smith, F. L. Willis, A. B. Cohen, and J. Regis, *Analysis of Transient Behavior of Unbonded EBR-II Fuel Pin Incident*, Trans. ANS *11*, No. 2, 698 (1968).

181. C. M. Walter and L. R. Kelman, *Penetration Rate Studies of Stainless Steel by Molten Uranium and Uranium-Fissium Alloy*, J. Nucl. Mater. *6*, 281 (1962).

182. C. M. Walter and C. E. Dickerman, *TREAT Study of the Penetration of Molten Uranium and U-5 wt % fs Alloy Through Type 304 Stainless Steel*, Nucl. Sci. Eng. *18*, 518 (1964).

183. *Reactor Development Program Progress Report, August 1968*, ANL-7487, p. 106.

184. H. P. Iskenderian, *Study on Conversion of TREAT Facility for Testing of Fast Reactor Subassemblies*, Trans. ANS *11*, No. 2, 643 (1968).

185. T. R. Loree, C. M. Fowler, E. G. Zukas, and F. S. Minshall, *Dynamic Polymorphism of Some Binary Iron Alloys*, J. Appl. Phys. *37*, 1918-1927 (1966).

186. Bruce Goplen, LASL, private communication.

187. L. V. Al'tschuler, *Use of Shock Waves in High-Pressure Physics*, Soviet Physics-Usp. *8*, 52-91 (1965) (English translation).

188. D. R. Stephens and E. M. Lilley, *Static P-V Curves of Cracked and Consolidated Earth Materials to 40 kbar*, UCRL-14711 (March 28, 1966).

189. J. G. Walsh, *The Effects of Cracks on the Compressibility of Rock*, J. Geophys. Res. *70*, 381-389 (1965).

190. R. G. McQueen and S. P. Marsh, *Ultimate Yield Strength of Copper*, J. Appl. Phys. *33*, 654-665 (1962).

191. G. W. Greenwood and M. V. Speight, *An Analysis of the Diffusion of Fission Gas Bubbles and Its Effect on the Behavior of Reactor Fuels*, J. Nucl. Mater. *10*, 140-144 (1963).

192. H. C. Rodean, *Relation for Condensed Materials Among Heat of Sublimation, Shock Wave Velocity and Particle Velocity*, J. Chem. Phys. *49*, 4117-4127 (1968).

193. S. M. Stishov, *Relation Between the Volume Jump During Melting and the Volume of a Solid at the Melting Point*, JETP Letters *6*, 97-99 (1967).

194. M. H. Rice, R. G. McQueen, and J. M. Walsh, *Compression of Solids by Strong Shock Waves*, Solid State Phys. *6*, 1 (1958).

195. L. M. Barker, C. D. Lundergan, and W. Hermann, *Dynamic Response of Aluminum*, J. Appl. Phys. *35*, 1203-1212 (1965).

196. H. C. Rodean, "Thermodynamic Relations for Shock Waves," in *Proceedings of the Fifth National Congress of Applied Mechanics*, American Society of Mechanical Engineers, New York, pp. 89-96 (1966).

197. A. L. Ruoff, *Linear Shock-Velocity-Particle-Velocity Relationship*, J. Appl. Phys. *38*, 4976-4980 (1967).

198. M. H. Rice, *Pressure-Volume Relations for the Alkali Metals from Shock-Wave Measurements*, J. Phys. Chem. Solids *26*, 483-492 (1965).

199. H. P. Richter, *Numerical Problems with Grüneisen Ratios*, UCRL-70670 Rev. 1 (Jan. 22, 1968).

200. D. E. Munson and L. M. Barker, *Dynamically Determined Pressure-Volume Relationships for Aluminum, Copper and Lead*, J. Appl. Phys. *37*, 1652-1660 (1966).

201. J. R. Asay and A. H. Guenther, *Ultrasonic Studies of 1060 and 6061-T6 Aluminum*, J. Appl. Phys. *38*, 4086-4088 (1967).

202. D. Miller, "Physical Properties of Solids and Liquids under Extreme Conditions," in *Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967*, ANL-7310, pp. 335-339 (Jan 1968).

203. L. V. Al'tschuler, A. A. Bakanova, and R. F. Trunin, *Shock Adiabats and Zero Isotherms of Seven Metals at High Pressures*, Soviet Physics JETP *15*, 65-74 (1962).

204. F. J. Rogers, *GERED--A Code for Equation of State Calculations at Moderate Pressures*, UCRL-50500 (Sept. 9, 1968).

205. O. L. Anderson, *Derivation of Wachtman's Equation for the Temperature Dependence of Elastic Moduli of Oxide Compounds*, Phys. Rev. *144*, 553-557 (1966).

206. J. B. Wachtman, W. E. Tefft, D. G. Lam, Jr., and C. S. Apstein, *Exponential Temperature Dependence of Young's Modulus for Several Oxides*, Phys. Rev. *122*, 1754-1760 (1961).

207. F. Birch, *Thermal Expansion at High Pressures*, J. Geophys. Res. *73*, 817-819 (1968).

208. C. R. Tottle, *Mechanical Properties of Uranium Compounds*, ANL-7070 (Nov 1965).

209. P. Sung and J. E. Turnbaugh, *Calculation of the Temperature-Dependent Shear Modulus of* UO_2, J. Nucl. Mater. *24*, 107-108 (1967).

210. T. J. Ahrens, W. H. Gust, and E. B. Royce, *Material Strength Effect in the Shock Compression of Alumina*, J. Appl. Phys. *39*, 4610-4616 (1968).

211. R. S. Thurston and W. L. Mudd, *Spallation Criteria for Numerical Computations*, LA-4013 (Sept 1968).

212. R. Hofmann, D. J. Andrews, and D. E. Maxwell, *Computed Shock of Porous Aluminum*, J. Appl. Phys. *39*, 4555-4562 (1968).

213. Z. Hashin and S. Shtrikman, *A Variational Approach to the Theory of the Elastic Behavior of Multiphase Materials*, J. Mech. Phys. Solids *11*, 127-140 (1963).

214. Z. Hashin, *On The Parametrization of the Elastic Moduli of Two-Phase Materials*, Trans. ASME *87E*, 953-954 (1965).

215. Z. Hashin, *The Moduli of an Elastic Solid Reinforced by Rigid Particles*, Bull. Res. Council of Israel *5C*, 46-59 (1955).

216. M. Reiner, *Deformation, Strain and Flow*, H. K. Lewis & Co., Ltd., London, p. 214 (1960).

217. E. H. Kerner, *The Elastic and Thermo-Elastic Properties of Composite Media*, Proc. Phys. Soc. (London) *69B*, 808-813 (1956).

218. J. H. Mackenzie, *The Elastic Constants of a Solid Containing Spherical Holes*, Proc. Phys. Soc. (London) *63B*, 2-11 (1950).

219. C. van der Poel, *Time and Temperature Effects on Deformation of Asphaltic Bitumens and Bitumenous-Mineral Mixtures*, Rheologia Acta *1*, 198-202 (1958).

220. E. Guth, "Hydrodynamical Theory of the Viscosity of Suspensions," in *Proceedings of the Fifth International Congress on Applied Mechanics*, Cambridge, Massachusetts, 1938, Wiley, New York, p. 448 (1939).

221. Z. Hashin, *Elastic Moduli of Heterogeneous Materials*, J. Appl. Mech. *29*, 143-150 (1962).

222. R. Hill, *The Elastic Behavior of a Crystalline Aggregate*, Proc. Phys. Soc. (London), Section A *65*, 349-354 (1952).

223. R. W. Goranson, D. Bancroft, B. L. Burton, T. Blechar, E. F. Houston, E. F. Gittings, and S. A. Landeen, *Dynamic Determination of the Compressibility of Metals*, J. Appl. Phys. *26*, 1472-1479 (1955).

224. R. C. Rossi, *Prediction of the Elastic Moduli of Composites*, J. Amer. Ceram. Soc. *51*, 433-439 (1968).

225. F. G. P. Sidel, *SOC: A Numerical Model for the Behavior of Materials Exposed to Intense Impulsive Stresses*, UCID-5033 (July 9, 1965).

226. J. Thouvenin, *Effect of a Shock Wave on a Porous Solid*, Fourth ONR Symposium of Detonation, ORN-ACR-126 (1965).

227. J. F. Heyda, *Plate-Gap Model of a Porous Solid and Its Application to Impact by Reduced Density Projectiles*, NASA CR-1140 (Aug 1968).

228. R. Courant and K. O. Friedrichs, *Supersonic Flow and Shock Waves*, Interscience, New York (1948).

229. D. T. Morgan, M. Rockowitz, and A. L. Atkinson, *Measurement of the Grüneisen Parameter and the Internal Energy Dependence of the Solid Equation of State for Aluminum and Teflon*, AFWL-TR-65-117 (Oct 1965).

230. D. Miller, Argonne National Laboratory, unpublished work.

231. N. Claussen, *Isothermal Youngs Modulus of UO_2 and UO_2-Mo Cermets*, (in German) English trans., AERE-TRANS-1094 (Sept. 1967).

232. R. J. Forlano, A. W. Allen, and R. J. Beals, *Elasticity and Anelasticity of Uranium Oxides at Room Temperature*, ANL-7101 (Dec 1965).

233. H. A. Bethe and J. H. Tait, *An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses*, British Report RHM(56)/113 (1956).

234. R. B. Nicholson, *Methods for Determining the Energy Release in Hypothetical Reactor Meltdown Accidents*, APDA-150 (Dec 1962); or Nucl. Sci. Eng. *18*, 207 (1964).

235. N. Hirakawa, *MARS: A Two-Dimensional Excursion Code*, APDA-198 (June 1967).

236. R. A. Meyer and B. E. Wolfe, *High Temperature Equation of State of Uranium Dioxide*, Trans. ANS *7*(1), 111 (June 1964).

237. G. L. Ball *et al.*, *Two-Dimensional Hypothetical Accident Investigation of a Mixed-Oxide Core Design for the Fast Test Reactor*, APDA-209 (June 1968).

238. D. C. Menzies, *The Equation of State of UO_2 at High Temperatures and Pressures*, UKAEA Report TRG-Report-1119 (Feb. 10, 1966).

239. E. J. Robbins, *Limits for the Equation of State of Uranium Dioxide*, UKAEA Report TRG-Report-1344 (Oct. 6, 1966).

240. W. W. Little, Jr., *MAX--A One-Dimensional Maximum Hypothetical Accident Code in FORTRAN-IV*, BNWL-612 (March 1968).

241. R. H. Brodie and J. E. Hormuth, *The PUFF 66 and P PUFF 66 Computer Programs*, AFWL-TR-66-48 (May 1966).

242. V. Z. Jankus, "A Modified Equation of State for Hydrodynamic Calculations in AX1 Numerical Program," in *Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, October 11-14, 1965*, ANL-7120, pp. 686-691.

243. N. H. Wagner, N. B. Brodie, and R. B. Bjork, "Impact of a Porous Aluminum Projectile on Aluminum at 20 and 72 km/sec," in *Proceedings of the Seventh Hypervelocity Impact Symposium, Tampa, Florida, November 17-29, 1964: Vol. III. Theory*, Martin Co. Report AD-463229 (Feb 1965).

244. O. L. Anderson, *The Use of Ultrasonic Measurements Under Modest Pressures to Estimate Compression at High Pressure*, J. Phys. Chem. Solids *27*, 547-565 (1966).

245. D. H. Chung and G. Simmons, *Pressure and Temperature Dependences of the Isotropic Elastic Moduli of Polycrystalline Alumina*, J. App. Phys. *39*, 5316-5326 (1968).

246. W. R. Stratton, D. M. Peterson, and P. M. Altomare, *Analysis of the KIWI-TNT Experiment*, Trans. ANS *8*, 126-127 (1965).

247. C. G. Chezem and W. R. Stratton, *RAC - A Computer Code for Reactor Accident Calculations*, LAMS-2920 (1963).

248. L. D. P. King, *Safety Neutronics for ROVER Reactors*, LA-3358 (1965).

249. R. B. Lazarus, W. R. Stratton, and T. H. Hughes, "Coupled Neutronic-Dynamic Problems," *Computing Methods in Reactor Physics*, Ed. by H. Greenspan, C. N. Kelber, and D. Okrent, Gordon and Breach, New York, pp. 513-533 (1968).

250. W. R. Stratton, T. H. Colvin, and R. B. Lazarus, "Analysis of Prompt Excursions in Simple Systems and Idealized Fast Reactors," in *Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958*, Vol. 12, pp. 196-206, United Nations, New York (1958).

251. D. Okrent, J. M. Cook, D. Satkus, R. B. Lazarus, and M. B. Wells, *AX-1, A Computing Program for Coupled Neutronics-Hydrodynamics Calculations on the IBM-704*, ANL-5977 (May 1959).

252. D. E. Simpson, Battelle Northwest Laboratory Report (Private Communication).

253. D. Miller, "Preliminary Report on the Equation of State of Iron and Steel," in *Reactor Physics Division Annual Report, July 1, 1967 to June 30, 1968*, ANL-7410, pp. 308-311 (Jan 1969).

254. J. Himpan, *The Definitive Form of the New Thermal Equation of State with Its Material Constants for Over 100 Different Substances* (in German), Montashefte für Chemie *86*, 259-268 (1955).

255. F. Hensel and E. U. Franck, *Metal-Non-Metal Transition in Dense Mercury Vapor*, Rev. Mod. Phys. *40*, 697-703 (1968).

256. N. Asami, M. Yamada, and S. Takahoshi, *Pressure-Temperature Phase Diagram of Uranium Metal*, Nippon Kinzoku Gakkai (J. Japan Institute of Metals) *31*(4), 389-394, in Japanese (1967).

257. W. Klement, Jr., A. Jayaraman, and G. C. Kennedy, *Phase Transformations in Uranium at High Pressure*, Phys. Rev. *129*, 1971-1975 (1963).

258. L. S. Levinson, *Heat Content of Molten Uranium*, J. Chem. Phys. *40*, 3584-3585 (1964).

259. A. Pattoret, J. Drowart, and S. Smoes, *Mass Spectrometric Determination of Heat of Sublimation of Uranium*, Trans. Faraday Soc. *65*, 98-112 (1969).

260. E. G. Rauh and R. J. Thorn, *Vapor Pressure of Uranium*, J. Chem. Phys. *22*, 1414-1420 (1954).

261. A. V. Grosse, J. A. Cahill, and A. D. Kirshenbaum, *Density of Liquid Uranium*, J. Amer. Chem. Soc. *83*, 4665-4666 (1961).

262. *Reactor Fuels and Materials Development Plutonium Research: January-June 1967*, MLM-1445 (May 13, 1968).

263. L. Liebowitz, L. W. Mishler, and M. G. Chasanov, *Enthalpy of Solid Uranium Dioxide from 2500°K to Its Melting Point*, J. Nucl. Mater. *29* (1969).

264. J. A. Christiansen, *Thermal Expansion and Change in Volume on Melting for Uranium Dioxide*, J. Amer. Ceram. Soc. *46*, 607 (1963).

265. R. A. Hein, P. N. Flagella, and J. B. Conway, *High Temperature Enthalpy and Heat of Fusion of UO_2*, J. Amer. Ceram. Soc. *51*, 291-292 (1968).

266. L. N. Grossman and A. I. Kaznoff, *Determination of the Enthalpy of Fusion of Uranium Dioxide*, J. Amer. Ceram. Soc. *51*, 59-60 (1968).

267. A. Pattoret, J. Drowart, and S. Smoes, "Etudes Thermodynamiques par Spectrometrie de Mass sur le Systeme Uranium-Oxygen," in *Thermodynamics of Nuclear Materials, 1967*, IAEA, Vienna, pp. 613-634 (1968).

268. R. J. Ackerman, P. W. Gilles, and R. J. Thorn, *High Temperature Thermodynamic Properties of Uranium Dioxide*, J. Chem. Phys. *25*, 1089-1097 (1956).

269. H. K. Richards, "Physical Properties of Uranium Monocarbide at the Melting Temperature and in the Liquid State," in *Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967*, ANL-7310, pp. 339-342 (Jan 1968).

270. E. K. Storms, "A Mass Spectrometric Study of the Vapor Pressure of Uranium (g) and UC_2 (g) over Various Compositions in the U-C System," in *Thermodynamics*, Vol. I, Proc. Symp. Vienna, 1965, IAEA, Vienna, pp. 309-343 (1966).

271. C. E. Holley, Jr. and E. K. Storms, "Actinide Carbides: A Review of Thermodynamic Properties," *Thermodynamics of Nuclear Materials, IAEA, 1968*, Paper SM 98/102, pp. 397-425.

272. P. W. Bridgeman, *Compression and the α-β Phase Transition of Plutonium*, J. App. Phys. *30*, 214-217 (1959).

273. D. R. Stephens, *The Phase Diagram of Plutonium*, J. Phys. Chem. Solids *24*, 1197 (1963).

274. R. G. Liptai and R. J. Friddle, *The Effects of High Pressure on the Beta-Alpha Transformation in Plutonium*, J. Nucl. Mater. *20*, 48-56 (1966).

275. C. Z. Serpan, Jr. and L. J. Wittenberg, *The Density of Liquid Plutonium*, Trans. Met. Soc. AIME *221*, 1017-1020 (1961).

276. T. E. Phipps, G. W. Sears, R. L. Seifert, and O. C. Simpson, "Vapor Pressure of Plutonium," in *Proceedings of International Conference on Peaceful Uses of Atomic Energy, 1955, 7*, 382, United Nations, N. Y. (1956).

277. R. A. Kent and A. Leary, *Mass Spectrometric Studies of Plutonium Compounds at High Temperatures*, LA-3902 (Feb. 29, 1968).

278. T. E. Phipps, G. W. Sears, and O. C. Simpson, *The Volatility of Plutonium Dioxide*, AECD 2609 (May 1949); also, J. Chem. Phys. *18*, 724 (1960).

279. R. N. R. Mulford and L. E. Lamar, "Volatility of Plutonium Oxide," in *Plutonium, 1960*, p. 411, Clever-Hume, London, Ed. by E. Grison and W. B. H. Lord (1961).

280. T. D. Chikalla, *The Liquidus for the System UO_2-PuO_2*, J. Amer. Ceram. Soc. *46*, 323 (1963).

281. L. F. Epstein, *Ideal Solution Behavior and Heats of Fusion from the UO_2-PuO_2 Phase Diagram*, J. Nucl. Mater. *22*, 340-349 (1967).

282. C. J. Kubit, R. G. Palm, and W. E. Gunson, *High-Temperature Thermodynamic Equation of State of Plutonium Carbide*, Trans. ANS *8*(1), 29 (1965).

283. R. C. Feber and C. C. Herrick, *Ideal Gas Thermodynamic Functions of Lanthanide and Actinide Elements*, LA-3184 (Feb. 5, 1965).

284. M. E. Fisher, *Renormalization of Critical Exponents by Hidden Variables*, Phys. Rev. *176*, 257-272 (1968).

285. G. N. Lewis, M. Randall, K. S. Pitzer, and L. Brewer, *Thermodynamics*, McGraw-Hill, New York, 2nd ed. (1961).

286. O. A. Haugen, K. M. Watson, and R. A. Rogatz, *Chemical Process Principles: Vol. II. Thermodynamics*, Wiley, New York, 2nd ed. (1959).

287. E. K. Storms, *The Refractory Carbides*, Academic Press, New York (1967).

288. F. L. Oetting, *The Chemical Thermodynamic Properties of Plutonium Compounds*, Chem. Reviews *67*, 261-297 (1967).

289. O. L. Kruger and H. Savage, *Heat Capacity and Thermodynamic Properties of Plutonium Dioxide*, J. Chem. Phys. *49*, 4540-4544 (1968).

290. E. R. Gardner, T. L. Markin, and R. S. Street, *The Plutonium-Oxygen Phase Diagram*, J. Inorg. Nucl. Chem. *27*, 541 (1965).

291. O. L. Kruger and H. Savage, *Heat Capacity of Plutonium Monocarbide from 400 to 1300°K*, J. Chem. Phys. *40*, 3324 (1964).

292. A. E. Ogard and J. A. Leary, "High Temperature Heat Content and Heat Capacity of Uranium Dioxide and Uranium Dioxide - Plutonium Dioxide Solid Solutions," *Thermodynamics of Nuclear Materials with Emphasis on Solution Systems*, IAEA, Vienna, Paper SM 98/32, pp. 651-664 (1968).

293. H. Savage, *The Heat Content and Specific Heat of Some Metallic Fast-Reactor Fuels Containing Plutonium*, J. Nucl. Mater. *25*, 249-259 (1968).

294. J. B. Moser and O. L. Kruger, *Thermal Conductivity and Heat Capacity of the Monocarbide, Monophosphide and Monosulfide of Uranium*, J. App. Phys. *38*, 3215-3222 (1967).

295. B. J. Seddon, *Physical Properties of Some Plutonium Ceramic Compounds: A Data Manual*, UKAEA Report TRG-Report-1601 (R) (Dec. 18, 1967).

296. E. S. Fisher, *Temperature Dependence of the Elastic Moduli in Alpha Uranium Single Crystals, Part IV (298° to 923°K)*, J. Nucl. Mater. *18*, 39-54 (1966).

297. L. T. Lloyd and C. S. Barrett, *Thermal Expansion of Alpha-Uranium*, J. Nucl. Mater. *18*, 55-59 (1966).

298. J. Viard, "Adiabatique Dynamique de l'Uranium," *International Colloquium on Detonation Waves*, p. 383, Gif S/Yvette, August 28-September 2, 1961, Edition du Centre National de la Recherche Scientifique, Paris, 1961 (in French) (English translation UCRL-Trans-1213(L)).

299. I. C. Skidmore and E. Morris, "Experimental Equation of State Data for Uranium and Its Interpretation in the Critical Region," *Thermodynamics of Nuclear Materials*, p. 173, IAEA, Vienna (STI/PUB/58) (1962); see also E. Morris, *The Equation of State of Uranium in the Critical Region*, British Report AWRE-O-61/62 (1962).

300. P. A. Persson and I. Persson, *Determination of State Data for Uranium at High Pressures by Means of Shock Wave Experiments*, UCRL-Trans-1173-L (from FOA-2 Report A2299-222, Sept 1964, of the Research Institute of National Defense, Stockholm, Sweden).

301. R. G. McQueen, S. P. Marsh, and W. J. Carter, *The Determination of New Standards for Shock Wave Equation-of-State Work*, LA-DC-8410 (CONF-670914-10) July 1967.

302. J. B. Wachtman, Jr. *et al.*, *Elastic Constants of Single Crystal UO_2 at 25°C*, J. Nucl. Mater. *6*, 39-41 (1965).

303. O. G. Brandt and C. T. Walker, *Temperature Dependence of Elastic Constants and Thermal Expansion for UO_2*, Phys. Rev. Letters *18*, 11-13 (1967).

304. R. A. Wolfe and S. F. Kaufman, *Mechanical Properties of Oxide Fuels*, WAPD-TM-587 (Oct 1967).

305. L. J. Graham and R. Chang, "The Elastic Properties of Uranium Carbide Between 4.2 and 300°K," "Compounds of Interest in Nuclear Reactor Technology," *Nuclear Metallurgy*, Vol. X, Ed. by J. T. Waber, P. Chiotti, and W. N. Miner, American Institute of Mining, Metallurgical and Petroleum Engineers, pp. 409-422 (1964).

306. R. Chang and L. J. Graham, *Low-Temperature Elastic Properties of ZrC and TiC*, J. App. Phys. *37*, 3778-3783 (1966).

307. C. P. Kempter, *The Isothermal Compressibility of Uranium Monocarbide*, J. Less-Common Metals *10*, 294-295 (1966).

308. J. A. Cornet and J. M. Bouchet, *Les Properties Elastiques du Plutonium en Phase ε Cubique Centree*, J. Nucl. Mater. *28*, 303-310 (1968) (in French).

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