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Preliminary Report
Study of Fast Reactor
Safety Test Facilities

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I. Introduction

In August, 1974 the Los Alamos Scientific Laboratory (LASL) agreed to undertake a study for the Division of Reactor Safety Research, AEC (now NRC) of the needs for new Fast Reactor Safety Test Facilities. The study was to include three closely related tasks.

(1) A review of current understanding of fast reactor safety was to be made, with the aim of identifying important areas of uncertainty which can not be adequately resolved using analysis, out-of-pile experiments, and/or existing in-pile facilities. One objective of the review, which is addressed in this report, is to delineate the characteristics of new in-pile safety test facilities (STF) which would be useful in resolving significant safety questions. A longer-range objective, not addressed in any detail in this report is the definition of a program of analysis and experiments, using out-of-pile methods together with existing and new in-pile facilities for advancing the understanding of fast reactor safety in a timely and economical manner.

(2) Conceptual designs were to be developed of one or more new in-pile facilities having characteristics identified in (1). An ANL study⁽¹⁾ had already identified one particular type of facility, namely a thermal driver plus converters, as an attractive option for adiabatic burst operation, or for a short term (≤ 2 min) steady-state operation plus burst. Moreover, the detailed design of such a facility was being undertaken by ANL. We therefore decided to concentrate on the conceptual design of a different type of facility which could produce prototypic powers in test fuel for a period of days to weeks and then impose a burst on such fuel. In the ANL terminology⁽¹⁾ this would be called a class III facility.* In addition,

* In the ANL report, an adiabatic or heat capacity system was designated a class I facility, while a system with additional capability to impose prototypic power levels on test fuel for short times (≤ 2 min) was called a class II facility. In addition, the class II facility was able to produce decay heat power levels in fuel for a few hours.

some consideration has been given to use of a sub-critical neutron multiplier driven by an accelerator for heating the test fuel.

(3) Advanced data acquisition techniques were to be examined for possible incorporation in new facilities. In particular, experimental methods for measuring motion of fuel, clad, and sodium were to be considered. The early consideration of such techniques is dictated by two considerations. First of all, facility design must be significantly modified in order to accomodate instruments such as a fast neutron hodoscope or a flash x-radiography system for measuring fuel motion. In addition, the value of a facility will strongly depend on the quality of experimental information which can be obtained during a test.

This is a preliminary report of the conclusions which we have reached during the first six months of our study, that is through February 1975. The report has been reviewed by NRC staff and slightly revised on the basis of this review. The review also underlined the importance of the early and clear definition of a number of key experiments, including how they would be performed and what would be learned therefrom. Such experimental definition is especially important when multiple facilities are proposed as in this report, inasmuch as each facility can be designed to perform well a limited class of experiments. LASL staff members who have made important contributions to this report include : Chapter III, IV and V (J. Boudreau, M. Stevenson, J. Kerrisk); Chapter VII (R. Palmer, T. McLaughlin, M. Battat, E. Cort); Chapter VIII (A. Blair, V. Starkovich, W. Stein, J. Yarnell). In addition, contributions and helpful criticism were received from our consultants (P. Bleiweis, I. Catton, J. Jackson, W. Kastenberg, and D. Okrent) and we have profited from suggestions of W. Davey, D. Hall, W. Stratton, and M. Tillery.

George I. Bell

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I.

- (1) Summary Status Report on ANL Findings and Recommendations on LMFBR Safety Testing Needs and Acquisition of New In-Pile Testing Facilities - June 28, 1974.

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II. Summary of Conclusions and Recommendations

A fast reactor safety program should be based on three important components. (1) The development, testing, and verification of reliable performance of plant protective systems, including fault detection, scram, and decay heat removal systems, is essential. (2) A program of design and testing of energy and radioactivity containment should be included. In this area, we recommend initiation of a study of containment capability vs economic cost. (3) An ability to understand a wide range of postulated core disruptive accidents (CDA), their initiation, progression, and consequences, should be developed. In addition, reactor designers must, in their consideration of design options, be responsive to safety considerations, for example, striving to minimize the probability of occurrence of faults which are difficult to detect and/or which may lead to particularly undesirable consequences.

In this report consideration is given to the level of understanding of postulated CDAs and to the role which new in-pile safety test facilities should play in improving this understanding. We wish to emphasize however, that in-pile safety testing must be accompanied by a vigorous program of out-of-pile experiments which are designed to investigate basic phenomena believed to be important in accident situations, together with a program of theoretical analysis to interpret the experimental findings and to incorporate their understanding with general scientific knowledge in computer programs for accident analysis.

The level of detail which it is fruitful to include in CDA analysis is not yet clear. Historically, the attempt was first made to obtain upper bounds on accident consequences by gross and hopefully conservative calculations. As the ability to perform more detailed and mechanistic calculations has advanced, in many instances, the calculated accident progression has become more benign. We therefore believe that it is important to be able to analyze the

progression and consequences of a range of postulated CDAs in considerable detail. Such an ability would make possible the more rational designation of performance standards for plant protective and containment systems. In addition it would assist the reactor designers in identifying and eliminating potentially hazardous features.

The viewpoint from which this report is written emphasizes realistic evaluations of accident sequences and phenomena. This leads into detailed discussions of the needs involved in achieving the understanding and capability to make realistic evaluations. We would hope that, as the understanding grows, particular phenomena may be found to dominate accident sequences, that these lead to relatively benign consequences, and that the phenomena are independent of initial conditions. This would lessen the need for extensive detail in analytical and experimental investigations. However, in planning for new safety test facilities we cannot afford to assume that this optimistic situation will prevail. The planning of a thorough experimental-analytical research program is necessary. As more is learned, areas of the program may lessen in priority, but we do not feel the understanding is sufficient at this time to establish these areas.

In the mechanistic approach to accident analysis, some well defined initiating fault, usually involving failure of the plant protective system, is postulated and the sequence of events, which may include progressive fuel melting and core disruption, is followed in some detail. From our review of CDA analysis, for a variety of initiating events, we have identified a number of overriding generic questions which we feel will remain for all fast reactor sizes and types.

- (a) When do assumed initiating faults lead to partial core disruption and under what circumstances can growth of the disrupted region be arrested?

- (b) How is ultimate neutronic shutdown achieved?
- (c) What, if any, mechanical damage is done to the primary system?
- (d) What is the character and quantity of radioactive accident products, if any, which may be released?
- (e) What is the long term disposition and coolability of any core debris?

We have identified four general classes of in-pile experiments which would be valuable for resolution of these key questions and have attempted to delineate new facility characteristics which would be required for performing each class of experiments. This has led to the partial definition of four different types of facilities. We do not wish to imply that most of the experiments could not be done in a single, elaborate, and costly facility. However, we do strongly suggest that consideration be given to a number of simpler facilities which are specifically designed for performance of a limited class of experiments. Characteristics of the four facility types follow.

Facility A

This facility is required for few pin experiments related to pin failure and initial cladding and fuel motion. Pin failure thresholds and characteristics, such as the axial location of failure, are required for prototypic pins having a wide variety of irradiation, thermal, and fabrication histories. In addition, early stages of clad relocation and fuel sweepout, if any, could be studied in some detail. Sustained steady state operation, at prototypic power densities in the test fuel, is required for preconditioning of fuel, preirradiated elsewhere. On top of this steady power, a short burst capability sufficient to melt and preferably vaporize FTR irradiated fuel is desired. A test volume to accommodate 19 or 37 full length fuel pins is acceptable and instrumentation to resolve fuel and cladding motion at the single pin level is needed. Simulations of both protected and unprotected transients could be carried out.

Preliminary scoping calculations have been performed to identify candidate facilities having these capabilities. A promising candidate uses a driver with

BeO-UO₂ fuel, having a Be/U ratio ≈ 50 , a U-235 enrichment $\approx 50\%$, and either stainless steel clad with sodium coolant or possibly niobium clad with gas coolant. Alternatively, UC-ZrC-graphite (Rover) fuel is a possibility with C/U ≈ 50 . However the latter driver would have a larger volume and higher heat removal requirements. In either case a thin converter region for flattening the power density in the test fuel may be desirable.

While optimization studies remain to be done, it appears possible that such a facility could have a burst capability of ≈ 2000 J/gm for prototypic FTR fuel (≤ 19 pins) and ≈ 8000 J/gm for fully enriched fuel with the burst imposed while the test fuel is operating at prototypic power densities ≈ 200 W/gm. The steady state power level in the driver fuel might be as low as 50 MW for tests of enriched fuel or 200 MW for prototypic fuel. The Doppler effect would provide the inherent shutdown mechanism.

We also believe that a flash x-radiography system or neutron and gamma ray hodoscope or holograph system can be developed to resolve fuel or clad motion with a resolution of about 1 mm in space and 1 msec in time. Individual pins in a 19 or 37 pin array should be resolvable.

Facility B

This would be designed for performing experiments related to gross fuel/steel motion within a subassembly and to a limited class of radioactive material transport phenomena. The facility should accommodate somewhat more fuel than a single FTR subassembly. It would be desirable to be able to run the test fuel at power for several minutes to an hour, prior to burst, in order to build in short-lived fission products and a source of decay heat. This would be useful for experiments involving fuel freezing and remelting or for the study of boiling fuel/steel mixtures. However, to some extent, these experiments could instead be performed using programmed fission heating rather than decay heat. Enriched fuel could be used and a burst capability of 6000 J/gm in the test fuel is desirable.

Several candidates can be identified for this type of facility. One would be an enlarged type A facility which would have ample capability for imposing a burst on enriched fuel. One difficulty for such a design would arise from reactivity changes caused by motion of large quantities of enriched fuel. In particular, compaction of the fuel from one subassembly or more might lead to a reactivity addition above that which could be compensated by the driver fuel Doppler coefficient. This difficulty could be avoided by use of partially enriched fuel, but at the expense of reduced burst capability and higher steady power levels. An alternative design would be the ANL Class I or II proposal. Such a machine would be limited in its capability to build in short lived fission products but would, because of its probable larger temperature coefficient, be less sensitive to reactivity changes resulting from fuel motion. In either facility, we believe that a hodoscope or flash x-radiograph system could give fuel and clad motion data which would be very useful in testing the understanding and analysis of subassembly scale motions. It is less clear that such large scale tests would be useful for gathering data on basic phenomena.

The relative capabilities and costs of these two type B candidates are not yet clear. Moreover, the relative priorities of experiments which might be better performed in one or the other have not yet been considered. If a type A facility were constructed, it might be desirable to provide space for testing more than 37 pins therein and to consider a type B machine of complementary capabilities, e.g., an adiabatic burst machine. However we do not feel that the options are sufficiently clearly defined to permit recommendation of priorities at this time.

Facility C

Experiments involving substantially more fuel than that in a single FTR subassembly would be performed in such a facility. The reactivity of such large masses of fuel, if fully enriched, would be substantial. Indeed, seven FTR subassemblies containing fully enriched fuel, and with a reflector would

be critical. Hence, there are at least two options which should be considered for such experiments. In one, a driver could be used to impose a transient on fuel of more or less prototypical enrichment, to minimize reactivity contributions of the test fuel. In the other, a small self driven assembly with enriched fuel, to minimize size and cost, could be used. We do not believe that the significance of these experiments is clear enough to permit a judgment to be made on these or other options at present. However we do recommend that serious consideration be given to the contribution which a few small well-instrumented self-driven transient experiments could make to the understanding of fast reactor safety. Highly idealized configurations should be considered as well as more prototypic, though small scale, assemblies of fuel and coolant. We have not identified any techniques which could measure detailed fuel/clad motion in such large masses of fuel, but we believe that gross motion, e.g., radial expansion velocities, could be measured.

Facility D

This facility would be specifically for post-accident heat removal studies and would require large masses of fuel, perhaps in pancake geometry, with relatively low levels of internal heat generation. No burst capability is required. One option would be to first irradiate fuel in a type A or B facility, to accumulate a decay heat source, to then melt it in the same facility allowing the molten fuel to accumulate for study in a separate PAHR test vehicle. This option would probably be very difficult to engineer and somewhat limited in the amount and geometry of the test fuel. A different option would rely on fission heating rather than decay heat. This could be done in or adjacent to a low power driver or in a subcritical neutron multiplier driven by an accelerator such as the 800 MeV proton linac at Los Alamos. The accelerator-multiplier option has the attractive feature that changes in test fuel reactivity could be easily compensated (within limits) by varying the fraction of the proton beam current which is dumped into the multiplier.

It is possible to define with confidence the sorts of experiments which would be performed in type A and type D facilities and to evaluate the impact which such experiments would be expected to have on the understanding of postulated CDAs. This is because the experiments could be designed to improve and test the understanding of conditions near the initiation or termination, respectively, of almost any postulated CDA. Thus in a type A facility, effects of a great variety of power, flow, and fuel conditions could be explored in considerable detail, while in a type D facility the simulated conditions would be relatively insensitive to initiating circumstances. On the other hand, type B and C facilities would be designed to explore conditions during accident progression, including the transition phase. Since our understanding of these conditions is quite incomplete, and in particular we do not yet know whether they may be dominated by a few general characteristics or are sensitive to details of initiation, it is difficult to define the relative importance of various experiments and facility characteristics. It is to be expected that understanding of these problems will improve as recent models of a dispersive transition phase are evaluated and as new computer codes are developed. Under these conditions it may be premature to define the characteristics of any elaborate, costly, and long lead time type B or C facility. Alternatively, attention might be focused on two types of integral tests: first, relatively quick and simple type C tests which are specifically designed to observe any dispersive transition phase phenomena, and second, prompt burst tests to normalize new and existing disassembly codes. Perhaps type A plus type D facilities, together with a few integral tests would suffice to make possible and to confirm an adequate understanding of the progression and consequences of postulated CDAs. However, at this time we recommend that the study of type B and C facility options and of the experiments to be performed in them be aggressively continued so that a more informed judgment may be made on their expected significance.

It is tempting to try to list a few "key experiments" to be performed in each facility, but we have resisted this temptation, partly because in such a rapidly changing field, the key experiments of today may not appear so crucial in a year or two, but largely because we view such facilities as experimental tools for providing both a data base and confirmation of analysis in an orderly and long term fast reactor safety program. However we do believe that it will be important to define in some detail, a number of the more important experiments to be expected in each facility. In this way a better judgement can be made of the importance of various facility design features and of the significance of compromises in facility number or capability.

III. Fast Reactor Safety Analysis: Perspective and Philosophy

Summary

The viewpoint on safety strategy adopted within this report is discussed in this chapter. Also some comments are made on the perspective in which the experimental program should be viewed. Our principal conclusions are:

(1) Some ability to predict the consequences of various postulated core disruptive accidents (CDA) is an essential element of the fast reactor safety program. The required degree of detail in the analysis is not clear; it will depend on the safety criteria adopted and, most importantly, on the results of the CDA studies themselves.

(2) Whether or not CDA's are taken as design basis accidents (DBA's), reactor containment capability should be based on a reasonable envelope of loadings arrived at through mechanistic analyses of postulated accidents. A study should be made of containment capability vs economic cost.

(3) Out-of-pile experiments are mandatory in the safety program for study of basic phenomena and of conditions early in a postulated CDA.

(4) The primary purpose of in-pile experiments is to test our understanding of CDA phenomenology, although in many cases, in-pile experiments are necessary to help develop this understanding.

(5) A vigorous theoretical analysis and methods development program is needed to predict the progression and consequences of postulated CDA's in a real reactor and to analyze in-pile and out-of-pile experiments.

(6) Instrumentation capabilities determine the utility of in-pile experiments.

A. Safety Strategy

Each reactor type has its own characteristics which are important in safety considerations. For example, the liquid metal cooled fast breeder reactor (LMFBR) is a low pressure system utilizing a coolant with excellent heat transfer properties, both favorable characteristics for safe operation. However it has long been recognized that fast reactors do not necessarily lose reactivity on voiding of coolant and that rearrangements (generally compactions) of core material may be postulated which lead to more reactive configurations. Therefore in considerations of fast reactor safety, emphasis has traditionally been given to core disruptive accidents (CDA).^{*} Particular interest attached to the analysis of postulated CDAs which might lead to explosive energy release and breach of primary containment and to a large extent the identification, analysis, and prevention of such accidents, if any, is still of primary concern. However, many of the more recent and detailed CDA analyses have indicated that gross core disruptions may be obtained without violent disassembly and in such cases an understanding of the disposition and coolability of the resulting core debris is important. We believe that some ability to predict the consequences of various postulated CDAs is an essential element of any fast reactor safety program. Even if the probability of occurrence of a particular CDA can be shown to be very small, we wish to be confident that its consequences are not unacceptably serious. There remains, however, an important question of how much detail is worth taking into account in the analysis, and this in turn is related to more general questions regarding CDAs.

A consensus has not yet been reached on the proper role of core disruptive accidents, hypothetical or otherwise, in an overall fast reactor safety and licensing philosophy. There is agreement that any fast reactor

* We use this term throughout this report to refer to those accidents involving substantial fuel failures and fuel melting. This encompasses partial or full core meltdown (without excursions) and also core disassembly transients.

should be designed and constructed such that the probability of a CDA is very small. Moreover, there is agreement that the fundamental objective of the safety studies is to insure the protection of the health and safety of the public but there is no general agreement on the most effective means of providing this protection. This is evidenced by the differing viewpoints presented recently by several U.S. speakers on this subject. (1), (2), (3), (4)

Although consideration of risk to the public in the form of radioactivity release includes certain non-core related accidents, those situations involving disruption of the core (generally meaning substantial fuel failures and fuel melting) pose the greatest potential risk. There are at least three identifiable variant positions with regard to the role of core disruptive accidents in LMFBR safety:

1. Reliance on protection (scram) and decay heat removal systems to assure that no identifiable and credible fault conditions lead to core disruption. This is based upon the postulate that if scram is assured, only very low probability initial fault conditions can possibly lead to core disruption in an LMFBR as long as shutdown cooling is also effective. Assurance of shutdown cooling to a requisite high reliability can be provided with backup decay heat removal systems if not with the primary heat removal system, since even piped system LMFBRs have a large coolant heat capacity.

Even if this position were to be adopted, it would be necessary to have calculations of CDA consequences, that is of system failure consequences, in order to rationally arrive at design reliability targets. For this purpose it might be possible to use rather crude, upper-limit calculations which would, however, lead to correspondingly low failure probability requirements. Thus, even in this position, it might be economical to work for a more accurate calculational capability which would lead to reduced estimates of CDA consequences. In addition there is always the problem of being confident that crude calculations

really represent upper limits, unless better calculations are available.

There are difficulties in this position. For one thing, demonstrating the necessary low probability of scram failure for anticipated transients is difficult. Moreover, this position allows little margin for overlooked fault conditions or for low probability events more severe than anticipated. It should be pointed out that this position, with no containment margins provided, has not been taken for any LMFBRs built or planned in the U.S. It may represent the USSR position.

2. A second position follows the first in reliance on engineered systems to prevent core disruption. However, this second position goes further by specifically providing energy and radioactivity containment margin in the design. The containment capabilities are provided without reference to a particular disruptive accident or without assurance that a particular disruptive accident would be contained.

The degree of containment could be chosen on economic grounds alone; for example, 10% of the plant investment might be devoted to containment. A more rational approach would be to investigate the degree of containment which could be achieved as a function of cost.* If realistic estimates were available of CDA probabilities and consequences it would then be possible to choose a level of containment which would protect the public from a range of relatively high probability, low consequence accidents and perhaps from all credible CDAs. At present such estimates are not in hand. However we believe that if this position of adopting a somewhat arbitrary containment margin is accepted, then it is important

* We are surprised that such a systematic study does not seem to have been done, and believe that it would be worthwhile.

to measure the design margin against the predicted consequences of postulated CDAs. In principle, comparison could be made with conservative, bounding calculations, but such comparisons would tend to force the designers towards providing a very generous margin of containment and towards substantial economic penalties. We therefore expect, without having seen any detailed study of the trade-offs, that at least over the long run and assuming many fast reactors are built it would be worthwhile to improve our understanding of CDA consequences so that containment margins could be reduced.

3. A third position takes as a design basis the assurance of containment of disruptive accidents evaluated assuming a mechanistic sequence of events. There are two variations of this which are significant:

a. Provision for an "inherently safe" design so that the system would be shutdown and cooled with minor disruption in the event of any whole-core accident involving fuel melting. This could include using devices such as fusible fuel pins as an emergency shutdown mechanism or could involve showing that the system would be shutdown and coolable through phenomena such as fuel sweepout. A core "inherently safe" for any situation involving fuel melting would be an attractive solution if it could be achieved. Efforts to this end should be encouraged; however, a position depending on successful achievement of this goal is premature at the present time.

b. Provision for engineered systems such that gross disruptive accident sequences including a core meltdown and disassembly would be contained with cooling of the debris assured. This

position is reasonable in its conservatism and straightforwardness. Difficulties arise in its application at the present time because of uncertainties in predicting accident progressions and severity with the current methods and understanding of the phenomena. In addition, the problem will remain of deciding precisely which accidents to consider in the design base.

Our viewpoint is that whether or not core disruptive accidents are taken as design basis accidents, the best approach is to seek a reasonable containment capability based on and enveloping the mechanistic analyses of various accident sequences. This implies that CDA sequences and phenomena need to be understood in some detail, although only insofar as the end results are sensitive to the details. As the understanding of CDAs and the experience with containment design improve it is to be hoped that this degree of containment will not be a severe economic penalty. Also, as operating experience is gained and protective system reliability becomes verified, the requirements on containment could be relaxed. In such a balanced approach, a fast reactor safety program should emphasize three basic elements. First of all, the development, testing and verification of the plant protective system (including fault detection, scram, and decay heat removal systems). Second, containment design and testing should be pursued. Finally, a vigorous program to improve capabilities for assessing consequences of postulated CDA initiating events is required. Substantial progress in any of these three areas will somewhat alleviate the requirements on the other two.

The viewpoint from which this report is written emphasizes realistic evaluations of accident sequences and phenomena. This leads into detailed discussions of the needs involved in achieving the understanding and capability to make realistic evaluations. We would hope that, as the under-

standing grows, particular phenomena may be found to dominate accident sequences, that these lead to relatively benign consequences, and that the phenomena are independent of initial conditions. This would lessen the need for extensive detail in analytical and experimental investigations. However, in planning for new safety test facilities we cannot afford to assume that this optimistic situation will prevail. The planning of a thorough experimental-analytical research program is necessary. As more is learned, areas of the program may lessen in priority, but we do not feel the understanding is sufficient at this time to establish these areas.

A final reason for detailed analysis of accident progressions is to enable the designer to avoid potentially hazardous design features, which might, for example, lead to autocatalytic configurations. As various options are considered for improving fast reactor performance, their safety implications must be considered as well.

In summary, we believe that some form of detailed analysis of postulated accident progressions will play an important role in fast safety studies but that the degree of detail will be determined by the safety criteria adopted and, most importantly, by the results of the studies themselves.

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B. Experimental Program Perspective

A program for understanding the progression and consequences of postulated CDAs should be based on a combination of theoretical analysis, out-of-pile experiments, and in-pile experiments.

In such a program, the role of the out-of-pile experiments is to clarify, whenever possible, the phenomena which analysis suggests may be significant in affecting the progression of a postulated accident. For example, in analyzing the course of a postulated CDA in an LMFBR, it is evident that an understanding of sodium boiling is required at an early stage. Such an understanding has been largely achieved through out-of-pile experiments⁽¹⁾ using electrically heated materials adjacent to liquid sodium. Another area in which out-of-pile experiments are vital, is in elucidating the phenomena which are involved in liquid-liquid interactions. An understanding of these interactions is required in order to answer such questions as, how rapidly can liquid fuel or steel transfer energy to liquid sodium or to sodium vapor? It appears that liquid-liquid interactions are quite complex and that therefore a variety of experiments⁽²⁾ using both convenient laboratory liquids as well as simulated reactor conditions are warranted, as well as an intensive program of theoretical analysis. As a final example of out-of-pile experiments, we note the structural dynamics tests⁽³⁾ in which the responses of simulated reactors and systems components to postulated dynamic loadings have been successfully studied.

In general, out-of-pile experiments are essential for studying conditions early in the development of a postulated CDA, such as sodium boiling and voiding, or in clarifying general phenomena, which are important independent of the details of a particular CDA configuration or condition. They are less useful in studying situations where fission heating and/or radioactivity of the fuel are important.

Two essential roles may be distinguished for theoretical analysis in the understanding of postulated CDAs. First, theoretical models must be constructed of the phenomena which may be important during a CDA. These models must be based on valid scientific and engineering considerations, together with the results of appropriate out-of-pile and in-pile experiments. In many instances models will have been developed precisely for the interpretation of such experiments. Second, theoretical models and computer programs must be constructed for predicting the initiation, progression, and termination of CDAs. It is neither possible nor desirable to include in these computer codes all of the details which are known to be present in a reactor. Therefore simplifications and approximations are made in formulating such codes. Many of the approximations can be tested by comparison with more accurate formulations or by comparison with out-of-pile experiments. Moreover, when the approximations cannot be well tested, the code designer will attempt to insure that they are conservative and lead to the prediction of more severe consequences than are actually expected. In addition, however, it is necessary to test the computer code predictions by comparison with in-pile experiments in which essential features of the accident situation under consideration are well simulated. Such a well tested computer code may then be used for the more confident prediction of other similar CDA situations.

In writing this report, we assume the existence of a vigorous program of out-of-pile experiments and theoretical analysis, without which a heavy investment in new in-pile facilities would be absurd. Indeed, we caution that if a decision is made to proceed with the construction of new in-pile facilities, this should not diminish the support given to these other essential activities.

The primary purpose of in-pile experiments is to test our understanding of CDA phenomenology. To this end, it is necessary that many of the experiments be realistic simulations of key conditions which may arise in a postulated CDA. Such experiments can be regarded as proof tests of our general understanding of the accident situation, or more particularly of the capabilities of some computer program for following the simulation. In addition, in-pile experiments designed to clarify general phenomena are also appropriate; such experiments can explore temperature and power density regions which are not accessible out of pile.

It is apparent that the value of an in-pile experiment depends strongly on the quality and the quantity of data which can be obtained for comparison with theoretical models. It is therefore imperative that consideration be given in the design of any new facility to the experimental data which can be obtained therein and to the impact which such data would be expected to have on our level of understanding.

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IV. Status of Accident Analysis and Experimental Needs

Summary

This chapter consists of an elaborate review of the state-of-the art in fast reactor safety analysis and experimentation. It is intended as supporting information for discussion in the remainder of the report and is included for the sake of completeness. Readers knowledgeable in the area may, obviously, skip this chapter. The following conclusions can be inferred.

1. The key considerations in LMFBR core disruptive accident analysis are:
 - a. When do assumed initiating faults lead to partial core disruption and under what circumstances can the growth of disruption be arrested? (coolability)
 - b. How is ultimate neutronic shutdown achieved?
 - c. What, if any, mechanical damage is done to the primary system?
 - d. What, if any, is the character and quantity of radioactive accident products which might be released?
 - e. What is the long term disposition and coolability of any core debris?
2. Uncertainties which currently exist in analytical and experimental programs support the need for a strong fast reactor safety program in both analysis and experiments.
3. Trends which are currently identifiable for other fuel types, reactor sizes and reactor types support the need for an experimental program based primarily on needs which have been identified for FFTF.

A. Introduction - Recent Developments in Accident Analysis

Early studies of postulated fast-reactor accidents emphasized the analysis of energetic prompt-critical excursions that were initiated by assumed fuel compaction or by postulated arbitrary increases in reactivity.^{1,2,3,4,5,6}

The approach of considering fuel compaction was based on the general notion that core melting could result from some type of power-flow mismatch and could lead to some form of fuel compaction. The specific modes of compaction considered, however, were quite arbitrary and were typically highly idealized geometries selected to produce what were thought to be conservative upper-bounds on the reactivity insertion rates. The early reactor designs were small and utilized metallic fuel; sodium voiding reactivities were negative.

Subsequent to these early studies, considerable effort has gone into developing the so-called "mechanistic" approach for analyzing accidents.⁷ In this approach, an accident sequence is analyzed from some postulated but well-defined initiating event up to the conclusion of the accident. The final result of the analysis is the consequence, or distribution of consequences, that can result from a given initiating mechanism. A mechanistic approach analyzes the step-by-step progression of accident sequences using cause and effect relationships. The first step in this approach is the delineation of those initiating events that are to be analyzed. Any identified and credible fault condition that leads to serious consequences is designed against as part of the overall safety effort. Thus, the events emphasized in CDA analysis are usually related to a postulated failure of the plant protective system.

It is very important throughout the following discussions to keep in mind that the probability of many events and the descriptions of the phenomena involved are often speculative. Judgement is a necessary part of the analysis of postulated unprotected accidents in LMFBR's, and the degree of exactness of the analysis and the extent of experimental verification are always questions to be kept in the forefront.

In recent years, three specific postulated accidents have been concentrated on. The first is referred to as a transient overpower accident (TOP). This accident is initiated by assuming a fixed rate of reactivity insertion

with failure to scram. The insertion rates considered can be estimated by study of specific phenomena or faults that could potentially add reactivity, or can be selected on the basis of serving as an envelope for unforeseen events. The second accident is an unprotected loss-of-flow accident (LOF). This accident usually assumes that the coolant pumps lose power and coastdown and that the plant protective system fails to scram the reactor. The third type of accident involves the propagation of local fault conditions. Both fuel-pin to fuel-pin and subassembly-to-subassembly propagation have been considered. Although a number of other accidents can be defined, the above three types are usually singled out with the expectation that other accidents will not tend to produce strongly different or more serious consequences.

Most of the state-of-the-art techniques in this area have been developed as part of the FFTF licensing effort. These various methods are described for the most part in supporting documents for the FFTF FSAR.^{8,9,10,11,12,13,14,15}

The current approach in mechanistic analysis can be broken into different phases shown in Fig. 1. The analysis is divided into these phases because quite different analysis techniques are required to solve different aspects of the overall problem. A brief description of each phase follows.

Fig. 1. The Mechanistic Approach to Accident Analysis.

Accident Initiation

The first phase of a mechanistic accident analysis involves a calculation of the core neutronics and thermal behavior up to the point of loss-of-subassembly-geometry. Since most LMFBR designs have used rigid subassembly ducts, the intact subassemblies are coupled only through inlet and outlet thermal-hydraulic conditions, possible duct-to-duct heat transfer, and reactor power and neutronics. Subassemblies with similar power, flow and irradiation conditions are generally lumped together into groups for the purpose of analysis. All the subassemblies within a given group, or "channel", are assumed to behave identically. Phenomena such as transient thermal-hydraulics, sodium boiling, fuel-pin mechanics and failure, cladding and fuel motion, and fuel-coolant interactions are treated with one-dimensional models. Multichannel codes of this type which were developed under USAEC sponsorship are the FREADM,¹⁶ MELT,¹⁷ and SAS¹⁸ systems. The purpose of these codes was to give a reasonable and conservative accounting of the reactivity effects in order to provide initial conditions for core disassembly codes; the phenomenological modeling has become more detailed in the latest versions.

Accident initiation codes have been specifically designed to analyze LOF and TOP accidents. Although the early stages of these accidents are quite similar, they diverge somewhat as the accident progresses. For example, consider the case of the Fast Test Reactor (FTR). During a flow-coastdown with failure to scram, the initiating phase of the accident is calculated to produce an initial disruptive sequence whose main elements have the following chronology.

An unprotected TOP, on the other hand, is calculated to produce its initial disruption as follows.

In the case of a LOF accident, the initial sodium boiling and voiding is calculated to precede melting of the cladding, which in turn precedes gross melting of the fuel for the FFTF case. Thus, gross fuel-pin melting and relocation occur in subassemblies previously voided of sodium. In the case of a TOP accident for FFTF the fuel pins, depending on the reactivity insertion rate, might rupture and eject molten fuel into the coolant channels before significant cladding melting or sodium voiding have occurred. Since the physical situations may differ markedly between these two accidents (at least for the initial fuel pins to disrupt), different models have been developed to describe the material motion.

There is potentially considerable overlap and variation in these two accident sequences depending on the specific reactor design and initiating conditions under consideration. For example, if the early stages of a LOF accident induce a rapid reactivity insertion, as might be true in a large LMFBR, much of the core could experience conditions similar to a TOP. Also, if the subassemblies should become voided and uncoolable due to the initial fuel ejection in a TOP accident, the continuation of the sequence could become very similar to a LOF.

Fairly detailed models have been developed to describe the initiating phase of accidents. Many aspects of these models have been or are in the process of being, verified by experiments.¹⁸ In general, this phase of the modeling is relatively well advanced, although not necessarily complete at this time.

As can be seen in Fig. 1, it is possible for postulated accident sequences to end in the initiating phase. This is indicated as an "early termination" in the diagram. This implies that the bulk of the core be permanently cooled in-place, i.e., be cooled in its normal pin geometry. Calculations^{8,9} done to date indicate that such in-place cooling is unlikely following an unprotected LOF in the FTR. This is because the accident is not ended neutronically until some fuel has been dispersed from the core region. Since fuel relocation in this case is preceded by sodium voiding, long-term coolability implies rapid restoration of cooling and continued flow of at least some sodium through the disrupted regions of the core. This does not currently appear to be very probable because of the extremely high temperatures that could exist in the disrupted regions.

There appears to be a higher probability of in-place cooling following a mild TOP accident in the FTR, but establishing this requires a detailed understanding of the extended motion of the dispersing fuel/sodium mixture.

Plugging of subassemblies as the hot core materials are displaced into the colder regions near the axial ends of the core is possible. The ability of the sodium to continue to flow and cool the debris in partially disrupted subassemblies must be examined carefully.

Core Disruption

If neutronic shutdown and permanent in-place cooling are not established, an accident will then proceed to a "core disruption" phase. This phase is characterized by subassembly deterioration and gross fuel motion. The objective of the analysis is to evaluate the continuing motion of the fuel to a stable, permanently subcritical configuration and provide initial conditions for damage evaluation and post-accident heat-removal assessments.

The form of the disruption, as well as the approach appropriate for analyzing the fuel motion, is strongly dependent on the nature of the accident sequence. If core-disruption results from an energetic prompt-critical neutronics burst, then the fuel motion can be calculated with a hydrodynamic disassembly model. A number of these models^{19,20,21} have evolved from the original approach developed by Bethe and Tait while others^{6,22,23,24,25} have been derived through weapons development and testing programs. Although these various models provide a number of different features and improvements to the basic formulation, they all rely on a hydrodynamic treatment of the reactor materials. Such a treatment is valid when the temperatures and pressures are sufficiently high that the reactor materials can be modeled as fluids. As a result, their applicability generally increases with the severity of the neutronics burst. However, in spite of this limitation, applications to excursions in solid (metallic) reactors and to experiments with reactors in which a solid is transformed to a gas or vapor during the excursion have been successful.⁵

If the core disruption is accompanied by only very mild excursions, then a different approach must be taken. The calculated disruption develops initially in the hottest portions of the core and gradually spreads as initially cooler subassemblies begin to melt. The disruption progresses primarily because of melting since the pressures generated are relatively small. Once a local deterioration of the subassembly ducts has been calculated to develop, the accident enters what has been termed the "transition phase".⁷ In this phase, the usual accident-initiation models may not be valid for the growing molten region. On the other hand, the entire core cannot be treated with a hydrodynamic model, since the pressures are low and much of the core may still be intact. Models are currently under development to analyze the fuel motion in a growing molten region. One such model²⁶ characterizes the molten region as a boiling pool of fuel and steel. By coupling this model with an extension of an accident-initiation approach in those portions of the core that retain intact subassembly duct structure, the overall fuel motion can be assessed. This can then be used as a basis for calculating the net reactivity and power history during the disruption. Multidimensional neutronics techniques may be required for this because of the gross fuel-motion under consideration. Some recent studies²⁷ suggest that damage potential from such a situation might be small.

Another problem encountered with mild excursions is the need for an analysis of the extended fuel motion. To analyze accidents to the attainment of a stable configuration, it is necessary to describe the permanent dispersal of at least part of the fuel from the core region. When the pressures driving the dispersal are mild, the outward motion can be strongly inhibited by interaction with the structure surrounding the core region. For example, the molten fuel tends to freeze and plug in the relatively narrow (and cold)

passages typically available for fuel dispersal.²⁸ Molten fuel initially dispersed upward could reenter the core region via gravity if its motion is arrested by plugging effects. Molten fuel dispersed outward in any direction could potentially reenter if overpressurization were to develop ahead of it.

In general, the disassembly models used to analyze energetic prompt-critical excursions are quite well developed. Efforts to provide an analysis of the extended material motion following a disassembly excursion are currently underway.^{29,30} Techniques to analyze gradual core disruptions that progress into the transition-phase are also being developed, but are in a somewhat more formative stage.

Damage Evaluation

The rate at which the thermal energy generated in a core disruption sequence can be converted to work energy must be established. Work energy and, potentially, damage can either come from the expansion of the core materials themselves or the interaction of these materials with the sodium coolant and its consequent vaporization and expansion. Early studies of the damage potential of core disruptive accidents assumed that the work on the surrounding reactor structures would be done by fuel expansion only. As larger oxide-fueled reactors began to be considered it was recognized that the transfer of heat from high temperature fuel to the sodium could considerably increase the potential work available to do damage. Hicks and Menzies calculated the maximum work potential that might result if heat transfer to the sodium were sufficiently rapid. Since very high heat transfer rates had been deduced from experimental reactor excursions (SL-1, SPERT) attention turned to predicting these rates by assuming fragmentation of the fuel into sodium. Typically, this was done parametrically by varying the fuel particle size and mixing time and calculating the resulting heat transfer rates as a basis for assessing damage potential. Results of parametric calculations

have been reported.^{32,33} An excellent review of work-energy conversion has been given by Padilla.³⁴

Since the heat transfer rates in these parametric models depend directly upon the particle size and mixing times, efforts have been made to establish the mechanisms by which mixing and fragmentation of fuel materials could take place.^{35,36} It now appears that these mechanisms may be sufficiently slow that they are not the major source of damage potential, although they are still of interest in assessing the course of accident behavior.³⁶ Because of the high thermal conductivity of steel, the possibility of high temperature molten-steel interacting with the sodium must be carefully evaluated.

Once the pressure source term has been established, the response of the system can be analyzed. This is usually done with a hydrodynamic calculation of the pressure propagation coupled with an analysis of the structural response of the important system components. The REXCO series of codes developed at ANL³⁷ is widely used for this purpose.

Post-Accident Heat Removal

Another important phase in an accident analysis is an evaluation of post-accident heat-removal (PAHR). The objective is to analyze the long-term decay-heat-removal from the fuel following an accident. If early neutronic shutdown is achieved with only slight material motion, the PAHR problem is primarily one of establishing the degree of in-place cooling of the bulk of the core. For accidents that proceed into gross core disruption, one ultimately wants to know where the fuel comes to rest such that it can be permanently cooled. This has mainly taken the form of postulating the spectrum of post-disruptive dispositions of fuel and analyzing the subsequent coolability. The disposition of core materials and the required measures to assure cooling of the core debris are, of course, dependent on the reactor design under consideration.

In particular, whether one is dealing with cooling the core with an intact primary system or whether one must assume the primary system is no longer intact makes a significant difference in the relative importance of the key phenomena involved in postaccident cooling.

Considerable technology has been developed to analyze several of the heat transfer modes which could be important in PAHR. Predicting the heat removal from internally heated debris beds and molten pools are two of the most important technical problems.^{39,40} Both the development of analytical models and supporting experimentation are underway.

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B. Phenomenological Areas and Research Needs for Near-Term Oxide LMFBRs

1. Pin-to-Pin Failure Propagation

A large LMFBR will contain on the order of 100,000 pins. It is likely that such plants will experience pin failures towards the end of each fuel loading cycle and might be expected to continue operation with as many as 1% of the pins failed. Although it is expected that the probability of failure propagation is very low, the possibility of operating with hundreds of failed pins indicates a need for investigation. Failure propagation has economic implications since failure of a large number of pins would necessitate a reactor shutdown. However, the safety-related concern is whether propagation could lead to cooling interruption in a subassembly, substantial cladding or fuel melting in that subassembly, and potential for propagation of the disruption to other subassemblies. The analysis of pin-to-pin failure propagation and the related experimental evidence have been recently presented in detail by Crawford, et al.,¹ and reviewed by Van Erp, et al.²

Whether pin failure propagation can become a significant safety concern is tied closely to the means of detecting fuel failures or cooling interruptions in a subassembly and the relationship of that detection to protective system action. If individual subassembly thermal-hydraulic monitoring (thermocouples and/or flow meters) is provided then gross cooling interruption could be detected within at most a few seconds. Similarly if failed fuel monitoring (delayed neutrons and/or fission product detection) is provided in the cover gas or in the primary coolant system then pin failures could be detected in perhaps tens of seconds or a few minutes at most. If protective action is taken on signals indicating substantial pin failures or cooling interruption, then a concern

is the propagation of failures on a time scale too rapid for detection and protection. If such systems are not provided, then slow propagation could also be a concern.

In much of the recent work done in this area questions related to rapid pin failure propagation have received the greatest attention. Three potential modes of propagation have been considered, i.e., (1) fission-gas release, (2) local flow blockages, and (3) local molten steel/fuel-coolant interactions. The first of these is a probable direct result of an initial pin failure. The second could be a result of debris from the exterior system, of fuel debris from a failed pin, of fuel swelling following sodium ingress, or of deformed cladding extending into the coolant. The third is usually considered as a result of molten fuel release from a failed overenriched pin. Molten steel-coolant interactions, perhaps resulting from local undercooling and sodium boiling, have received less attention as a potential pressure source.

Failure propagation due to fission gas release has been studied analytically and through out-of-pile simulation experiments. These investigations have considered both transient thermal effects, e.g., hot gas impinging on cent pins or local flow reductions, and transient mechanical loadings, e.g., pressure pulses. It has been concluded¹ that neither of these effects is likely to cause further pin failures.

Local blockages due to debris is a more difficult problem if only because of the difficulty in establishing the nature of the debris. Generally, a reactor will be designed to eliminate the possibility of quantities of debris being carried in the coolant such that any problem could occur. It is also not expected that random pin failures would result in significant quantities of debris being ejected from the failed

pin. Nevertheless, calculations have been performed which indicated that even if boiling is assumed in the wake of a severe local blockage, then dryout would be unlikely. The conclusion has thus been made¹ that for local blockages to lead to further propagation of failures, sufficient debris must be in the coolant channel to produce molten fuel in the fuel pin. A failure of the pin and release of further debris to form more blockages would be required to result in a potential for further rapid failure propagation. This is thought to be unlikely.

Failure propagation from molten fuel ejection has also been investigated with regard to failures of an overenriched fuel pin. Attempts were made to investigate this question in the D1 and D1A experiments in TREAT.³ Ironically, and definitely encouragingly, fuel pins with overenriched sections did not fail in the TREAT transients. Analyses¹ of temperature distributions including the effects of molten fuel redistribution have also indicated that, for credible levels of overenrichment, little or no molten fuel would exist and failures would not be expected. Even if failures followed by molten fuel ejection into sodium are postulated, upper bound analyses have indicated that pressure pulse propagation would not result in further failures.

In summary, it is expected that rapid fuel-pin failure propagation is not a serious safety problem. Slow failure propagation is more difficult to rule out but is easier to detect. It is encouraging to note that in this area there is considerable analytical and experimental evidence supporting this position. Further, there has been little or no evidence of pin failure propagation for about 200 failed pins out of 20,000 pins in the world's LMFBR irradiation experience. It is also encouraging that in recent EBR-II experience, some fuel pins have

achieved very high burnups (~16%) before failure. Thus it may be feasible to design and operate commercial LMFBR's such that fuel pin economics is governed more by fission-product buildup and neutron poisoning than fuel-pin failures.

On the other hand, if reactors are designed and operated such that at the end of a fuel subassembly's life in the reactor, some 1% of the pins would be statistically expected to be failed, then failure propagation should remain a significant area of investigation. If 1% of the pins in a subassembly are failed for any reason, then the remainder might be near failure since the pins in most subassemblies are not too dissimilar. Slight perturbations such as due to a single pin failure might then challenge the integrity of unfailed pins. It is expected that as any new mechanisms for propagation under these conditions are identified they can continue to be successfully addressed through analysis and out-of-pile experiments. There may remain a need for proof-test experiments under relatively prototypic conditions involving end-of-life fuel pins. The character of the problem suggests the use of long-term operation with failed fuel in a steady-state reactor. Many of the needs can be addressed in the Sodium Loop Safety Facility (SLSF) in the ETR. The use of EBR-II, perhaps with modifications, for extensive run-beyond-failure tests might also be appropriate.

Because of the progress made to this point, the availability of out-of-pile facilities including the Failed Fuel Mockup (FFM) facility at ORNL, the availability of SLSF, and the possibility of using EBR-II, we do not see a need for pin-to-pin failure propagation experiments being a primary requirement for other new safety test facilities. It is likely that a new transient facility would be used for certain phenomenological experiments, such as for the effects of release of a small amount

of molten fuel, but requirements for such experiments would not be appreciably different from requirements for transient overpower and undercooling experiments to be discussed later.

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2. Transient Overpower

A sequence of events often associated with the initiating phase of a transient overpower core disruptive accident has been identified for a postulated 50¢/sec-reactivity-insertion-without-scrum accident in FFTF.¹ In summary, the sequence is (1) fuel-pin failure under overpower conditions, (2) ejection of a fission-gas/fuel mixture into flowing sodium, (3) interaction of this mixture with the sodium, (4) sweepout of the fuel from the core region leading to a neutronic shutdown, followed by (5) cooling of partially disrupted subassemblies. Considerable effort has been directed at modelling steps (2) through (4) of this assumed sequence of events, notably in the MELT-III code,² the SAS/FCI module³ of the SAS3A code,⁴ the HOPE code,⁵ and the PLUTØ code.⁶ A significant effort has also gone into modelling of fuel-pin behavior under transient conditions. Very little detailed modelling has been developed for analyzing disrupted subassembly cooling.

This sequence is one of three possible paths for this kind of accident. A second sequence involves a rapid addition of reactivity due to sodium voiding and fuel motion leading to a prompt or near-prompt critical transient and core disassembly. The initial fuel motion reactivity is determined strongly by the pin failure location. Studies for both FFTF² and the SNR-300⁷ indicate that if the failure location is well above the core axial midplane, then the fuel motion reactivity tends to lead to a neutronic shutdown, provided fuel is indeed swept out of the core region. On the other hand, if fuel failure is assumed to occur at or very near the core midplane then a transition to core disassembly conditions is calculated. These conclusions are based on analyses of postulated moderate reactivity insertion rate (\$.50/sec to \$3/sec)

accidents. For higher postulated reactivity insertion rates prompt criticality may be reached prior to any pin failures and a "classical" sodium-in reactor disassembly might occur. For lower and presumably more credible insertion rates, such as a few cents per second, the sequence might be expected to tend more toward neutronic shutdown with in-place cooling since fuel pin failures should occur more non-coherently.

The current analyses, for the most part, assume that fuel ejected into the coolant stream is swept from the core. If a neutronic shutdown occurs, then partially disrupted subassemblies are assumed to be coolable by the still-flowing sodium. A possible sequence, however, involves loss-of-coolability and continuance of the accident sequence into a transition phase as defined earlier. In this case, loss-of-coolability could occur due to a combination of flow blockages from fuel plugging and a neutronic power higher than decay power if only a small amount of sweepout occurs. Whether neutronic shutdown occurs is also dependent on the assumptions made on the total magnitude of the reactivity insertion. An unbounded insertion leads ultimately to complete core disruption; however, this is an unrealistic assumption.

As noted above, the fuel-pin failure conditions, and particularly location, are important in determining the subsequent sequence of events. A number of models have been developed and used recently in fuel-pin failure analyses of TREAT transient overpower experiments, notably the SAS/DEFORM-II,⁸ PECT-II⁹ and BEHAVE-II¹⁰ codes.

The primary emphasis in these codes is on predicting cladding strain due to thermal expansion of the fuel against the cladding, but thermal expansion is only one of the mechanisms which may be involved in pin failure. For fresh (unirradiated) pins, meltthrough of the cladding, probably in conjunction

with localized boiling and cladding dryout, is the likely mechanism for failure in mild overpower transients, such as in the TREAT H2 and E4 experiments.¹¹ In rapid overpower transients, internal pressurization by fuel vapor and fill-gas could cause a burst-type failure in fresh pins as in the S11 and S12 experiments in TREAT.¹² Internal fuel motion and pressure relief can be a significant factor in delaying failure particularly if the fuel porosity is not great enough to accommodate the fuel expansion on melting. This was evidenced by the D1 and D1A experiments¹³ in which a short (3.8 cm) section of a pin was filled with high-density overenriched fuel pellets in an attempt to fail the pin and release a small amount of molten fuel. In these experiments the pins did not fail even though the overenriched pellets in D1A were about 50% molten. Evidently axial motion and pressure relief were effective in reducing the loadings of the fuel on the cladding.

With irradiated pins, transient fuel swelling due to intragranular fission-gas-bubble precipitation and coalescence can add to the thermal-expansion loadings on the cladding. Perhaps more importantly, transient release of intergranular gas can lead to a loading on the cladding either directly or through cracked and essentially strengthless unrestructured fuel. This mechanism is simulated in a "burst criterion" pin failure option¹⁴ in SAS/FCI, but the transient gas release is treated parametrically. This criterion is similar to that proposed by Stuart and Thomas.¹⁵

Mechanistic modeling of transient fuel swelling has been underway for some time as included in the GRASS¹⁶ and FRAS¹⁷ codes. Modeling of gas release is not so far along, although FRAS has been successfully calibrated with gas release data from HEDL out-of-pile experiments¹⁸ and with the TREAT

H3 experiment.¹⁹ FRAS models the release of the gas to fuel grain boundaries but currently must assume complete release after the gas reaches grain boundaries. This is consistent with the HEDL out-of-pile release measurements since in these there was probably little holdup of gas in grain boundary porosity. The comparison of FRAS with gas release in the central irradiated pin in H3 was also consistent in that the release determination was made during post-mortem examinations in which gas reaching grain boundaries would have been released. However, in many transient situations much of the gas could be trapped in unconnected grain boundary porosity.

DEFORM-II was used to precalculate the H3 irradiated-pin failure threshold experiment in TREAT (as were the PECT-II and BEHAVE-II codes). DEFORM-II calculated about 0.3% plastic deformation in the test pin, indicating that the pin should not fail.²⁰ Post-test measurements of the unfailed pin gave no plastic deformation within the 0.2% measurement error. The 0.3% plastic deformation calculated as compared to none measured could be explained as due to either analysis, initial pin characterization, experimental energy input errors, or measurement errors. It is likely, however, that the one-dimensional plane strain approximation used in DEFORM-II overpredicted the effective radial thermal expansion of the fuel after cladding-fuel contact since axial plastic flow of the hotter fuel in the center of the pin would reduce the effective radial expansion.

A DEFORM-II analysis²¹ of the later H5 experiment did predict a plastic hoop strain at the experimentally determined failure time which was in good agreement with burst tests on irradiated cladding from the PNL-17 pins used in both H3 and H5. However, the "burst criterion" mentioned above also gave good agreement with the failure time.

Depending on the fuel-pin irradiation and transient history and on thermal conditions, any or a combination of the cladding loading mechanisms and also meltthrough can lead to cladding failure. Detailed models which treat all of the mechanisms involved should be and probably can be developed in the long-term, and then verified by experiments in either TREAT or a new facility. Currently, some emphasis is also being placed on experimental correlations derived from the failure threshold experiments in TREAT. This is discussed in a recent paper by Scott et al.,²² and, encouragingly, there is considerable evidence that, at least for FFTF TOP conditions, fuel pins will fail well above the midplane. There are good arguments for this whether failures are from meltthrough, cladding strain due to thermal expansion, or cladding bursting from internal pressurization. In each case, the cladding being hotter above the midplane could be a dominant factor. However, justifying these arguments by the current TREAT experiments is somewhat difficult. The major shortcomings are not with TREAT itself but are rather with the lack of prototypic fast-reactor irradiated pins. EBR-II irradiated pins are 33 cm (13 in) long as compared with the 91 cm (36 in) FFTF pins, and they suffer from the relatively flat EBR-II axial power profile. This latter effect results in nearly all of the pins having only one characteristic region (high-power or low-power) along the length rather than having a high-power (fully restructured) region in the middle with low-power (unrestructured and gassy) regions at the ends.

As indicated, there will be a continuing need for fuel failure threshold experiments. There is and will also continue to be a need for in-pile phenomenological experiments, particularly fission-gas related experiments, under a variety of conditions. Aside from very rapid transient overpower experiments, many of these can be performed in TREAT. The major difficulty,

and this could be significant, is the lack of long-term steady-state operational capability in TREAT and the resulting inability to "precondition" the fuel prior to the transient. Here, as in the remainder of the report, we use preconditioning to mean operation on the order of a week without a shutdown prior to the transient. This will be discussed in more detail later in the report.

The first analyses of a mild transient overpower accident in FFTF assumed that the pins failed at the midplane leading to a prompt-critical transient and core disassembly.²³ The FISFAX model²⁴ was then developed and used in an evaluation of a sodium-in prompt critical transient in FFTF and was the first to include the effect of sodium-induced (FCI) dispersive motion in terminating rapid transient. This concept was included in the next generation of models, such as now in MELT-III and SAS/FCI, which initially indicated that fuel sweepout could terminate mild reactivity transients (of the order of 50¢/sec) well before reaching prompt criticality. As of yet, there is little direct experimental evidence which confirms these models. The primary difficulty in the TREAT experiments, particularly those in which the transients are continued well beyond initial fuel failure, such as E6 and E7,¹¹ is that continued fuel melting and motion lead to fuel plugging as the fuel moves out its original position. Fuel plugging was not initially modeled in the TOP-related codes, but presumably will be included in more advanced models.

Although fuel plugging is a real effect and must be included in the models, the TREAT experiments tend to overemphasize plugging effects. There are three primary reasons for this. The first, not inherent in TREAT, is the low resistance to flow reversal in the MARK-II loops used in the experiments. A second

effect is that of scale. The relatively high surface to volume ratio in the seven pin tests leads to high heat removal rates from fuel which has moved up and begun to freeze against the test container. A third effect is that the decay heating level in the irradiated fuel is extremely low, although this is of lesser importance since in most cases plugging seems to occur even with significant fission heating.

The latest modeling of the processes expected to occur following fuel-pin failure in a transient overpower situation is extremely detailed as exemplified by the PLUTO code⁶ and the HØTPIM model²⁶ being included in MELT-III. It is likely that the level of modeling detail will continue to increase, whether in attempts to improve the whole-core accident analyses, to understand the experiments or to do both. There will be an obvious need to obtain more detailed experimental information both from phenomenological and integral experiments. It is usually a strong desire of the analyst and model-builder to "see" as accurately as possible such phenomena as fuel expulsion from the pins, fuel sweepout, and fuel plugging. While we would hope that much information, particularly that leading to improved intuition and models, could be obtained from out-of-pile experiments, in-pile experiments with high image resolution provided by fast neutron or gamma hodoscopes or x-radiography is a definite need.

The phenomena which might occur in a mild transient overpower accident also need to be investigated in larger pin bundles than have been done to date. Overcooling of edge pins in subassembly cans can delay the failure of these pins compared to the failure of the central pins of the same subassembly. To what extent the initial pin failure in a subassembly promotes failure propagation in an overpower situation is not known. This propagation might

be due to pressure pulses, molten fuel and fission gas impingement, or to flow blockages from ejected and plugged fuel. The non-coherence also has considerable effect on the hydraulics after the initial pin failure.

In very non-coherent failure situations, the relatively small amounts of initially ejected fuel might be quickly quenched by the liquid sodium, leaving solid fuel particles to plug or to be swept out of the subassembly.

At the same time open coolant channels might be left near unfailed pins to maintain coolability of the entire subassembly throughout the accident.

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3. Transient Undercooling

There are several transient undercooling accident initiating events which can be postulated, such as whole core loss-of-flow (including transients due to pump coastdown and pipe ruptures), high inlet temperature transients due to loss-of-heat sink, and single subassembly flow blockages. Without protective action these can potentially lead to a sequence involving sodium boiling, cladding melting and fuel motion. These phenomena are generic to unprotected undercooling accidents but may occur with different sequences and consequences depending on the initiating events and on the reactor design. Since the current status of analysis and experimentation is strongly tied to the FFTF unprotected LOF (pump coastdown) analyses¹ we will briefly review the models and supporting experiments relevant to those analyses.

Sodium boiling models have a long history of development in both the U.S. and in Europe. The most advanced U.S. model in production use is that contained in the SAS3A code² and is basically a multiple-bubble slug-annular flow model. This model considers the formation and continued presence of a number of sodium vapor bubbles in a channel and follows the bubbles either as they move up a channel and collapse (or reach the exit), or as they become sustained and void the channel. In sustained bubbles, the liquid sodium film known to remain on the pins after initiation of bubble formation is allowed to move under the influence of vapor streaming in the bubble.

This model has been compared with several experiments, particularly R-Series experiments in TREAT.³ In general, agreement between calculated and experimental results for the R-Series experiments is good.⁴

Flow reversal and initial test-section voiding are predicted well

by the SAS calculations, and experimental and calculated coolant temperatures at the top and bottom of the active fuel agree well into boiling, thus indicating that the overall void-growth behavior is predicted relatively well.

Although the agreement between averaged quantities is fairly good, there is no evidence that the early voiding actually exhibits a multi-bubble character in which the vapor bubbles fill the complete flow area. While the multibubble approach is perhaps phenomenologically correct for a single subchannel at uniform temperature, it may not be correct for situations involving local variations in temperature since initial bubbles may not fill the complete flow area. Nevertheless, the treatment in SAS3A may be a reasonable approximation for predicting the general voiding behavior in the seven-pin R-Series experiments.

For larger test sections or for full-subassembly reactor situations, a larger degree of radial temperature variation and noncoherence of boiling initiation would be expected. The effect of noncoherent initiation on gross boiling behavior is not yet known. Advanced models being developed will require further experimental investigation both to provide a basis for making new model assumptions and to verify the completed models. It is expected that much experimentation can be performed out-of-pile with both simulant materials and with electrically-heated sodium loops. Out-of-pile experiments should be adequate to study sodium boiling as an isolated phenomenon. However, investigations of the interactions of sodium boiling with cladding and fuel motion lend themselves to in-pile experiments.

In early analyses⁵ of the loss-of-flow accident for FFTF it was noted that cladding would melt and tend to move some 2-3 seconds prior to fuel melting (the reactor power is relatively constant throughout

this phase since void reactivities have little effect in FFTF). It was recognized that upward sodium vapor velocities could be high enough to cause flooding (wave formation) of melting cladding and could lead to upward cladding motion and initial plugging above the fuel column. Cladding melting after this would tend to drain down and plug below the fuel region. The CLAZAS module⁶ of SAS3A was then developed to treat the dynamics of cladding motion. CLAZAS uses an explicit calculation of molten cladding motion under the influence of gravity, the sodium vapor pressure gradient, the frictional drag due to streaming vapor, and the friction between moving cladding and the fuel pin. Out-of-pile experiments have indicated that the model is qualitatively correct in its assumptions of flow regimes, and a comparison of CLAZAS calculations with the TREAT L2 experiment⁷ showed good qualitative agreement of cladding blockage configurations. In general the model does tend to predict more massive blockages above the fuel column than seems to occur in experiments. The R-Series TREAT experiments indicate much larger blockages below the fuel column than above, but the position and extent of lower blockages seem to be predicted well by CLAZAS.

The CLAZAS model may be reasonably appropriate for the seven-pin experiments. The model does not allow for cladding entrainment due to higher sodium vapor velocities than are expected under LOF conditions but which might be expected under other voiding conditions. Also, since the SAS3A sodium boiling does not include noncoherence effects it does not allow for the sodium vapor to seek lower flow resistance paths near those pins on which the cladding has not melted, an effect which would decrease the vapor velocity in the molten cladding region and thus stop the flooding process and upward motion of cladding. As noted earlier,

as advanced models are developed including noncoherence effects, verification experiments will be necessary. In FFTF, neither sodium voiding nor cladding motion appear to carry an undercooling accident to a conclusion; i.e., either disassembly or neutronic shutdown.¹ Fuel motion is a significant effect, however, since it does have the potential to force either conclusion. A common assumption has been that fuel slumping during melting can add reactivity rapidly enough to drive the reactor to near or above prompt critical and a mild disassembly. Considerable attention has been given to dispersive mechanisms such as the release of retained fission gas, to sodium vapor drag on fuel particles, and steel vaporization following rapid entrainment as the fuel melts. All of these possibilities are modelled in the SLUMPY fuel motion module⁸ in SAS3A, but the experimental basis of this modelling is limited at present. Out-of-pile fuel melting experiments at HEDL,⁹ the L-Series and R-Series in TREAT, and direct electrical heating experiments¹⁰ at ANL have been and are continuing to be used to develop and/or verify the SLUMPY models. The recent F1 and F2 experiments¹¹ in TREAT were performed as phenomenological experiments for the specific purpose of SLUMPY verification. These single-pin dry capsule experiments included the unique use of a uranium fission-heated pin container in an attempt to construct the proper thermal environment for the single test pin.

The discussions above have centered on the analysis of a postulated unprotected loss-of-flow (pump coastdown) accident in FFTF. Other postulated accident initiating events could lead to different orderings and relationships of phenomena. For the most part, however, if the detailed understanding attempted for the FFTF LOF case could be achieved, then it is probable that this understanding could be extended to most

other situations. Some particular situations will probably require analysis development and corresponding experimental verification beyond that. For example, sodium boiling dynamics under high heat flux conditions will be important in the analysis of larger reactor unprotected LOF accidents. Boiling dynamics under low flow and low heat flux conditions as might occur in shutdown loss-of-flow accidents should receive attention. Similarly, boiling dynamics under high flow but high inlet temperature conditions would be important for postulated unprotected loss-of-heat-sink accidents. As noted earlier, for most cases boiling dynamics can be studied through out-of-pile experiments, but there will be a continuing need for in-pile experiments involving interactions of boiling dynamics with cladding and fuel dynamics. In some cases, non-coherence effects may be important and may require fairly large-scale and perhaps full-subassembly experiments.

As noted, there will be a need for in-pile phenomenological experiments to investigate fuel dynamics following sodium voiding and cladding melting. In modelling fuel motion under these conditions, such questions as how and when fuel motion is initiated, what form the fuel is in, what are the effects due to sodium vapor, steel entrainment and fission gas release, etc., will be considered. Some basic understanding can be achieved through out-of-pile experiments, but in many cases fission heating will be necessary. It may also be necessary to use pre-conditioned fuel, since fission-gas release and fuel fragmentation could be strongly dependent on the fuel structural nature. In these experiments there will be a strong incentive to obtain as detailed test information as is feasible and, particularly, precise visual resolution. Here as in other areas in-pile transparent loops using gas cooling might be useful to provide visual images as a supplement to fast-neutron hodoscopes or flash x-radiography.

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4. Transition Phase

Postulated accidents that proceed into gross core-disruption without an accompanying energetic prompt-critical excursion enter what was defined earlier as the "transition phase". This phase is characterized by regions of molten core materials that develop initially in the most overheated subassemblies and spread to successively colder subassemblies as the duct walls melt and molten materials from adjacent subassemblies coalesce.

One of the main phenomena that must be treated in the transition phase is the behavior of growing molten regions. The time-dependent geometrical distribution of the fuel is needed to determine the ensuing reactivity and power history. The thermal and mechanical attack on adjacent subassemblies and structure is also required to determine the rate of growth of the molten region including meltout to the above and below core regions. The molten regions consist of a mixture of fuel and steel with the possibility of some entrained sodium. The calculated heating rates can vary widely, depending on the reactivity and power history, including the possibility of one or more mild excursions. The only restriction in the modeling is that the excursions not be so severe as to produce high disassembly pressures. The treatment of severe bursts involves somewhat different phenomena as discussed in the next section.

The analysis of the transition phase is complicated by the fact that no single idealized model is applicable throughout the entire core. In addition to molten regions, a continued description of those regions that retain intact duct structure must also be

provided. The phenomenology involved is similar to that described for the initiation-phase, although special techniques may be required to carry the analysis over longer time spans.

The analysis of the molten regions and the remaining intact sub-assemblies are coupled neutronically through the net reactivity and power level of the system. Multi-dimensional neutronics calculations could be necessary because of the gross material motions that are involved.

As mentioned earlier, an unprotected flow coastdown accident in a small reactor like the FTR appears to lead directly into a transition phase situation.¹ This is largely because the reactivity effects associated with sodium voiding, molten cladding relocation, and initial fuel motion are sufficiently small that neither vigorous bursts nor neutronic shutdown is assured prior to gross sub-assembly melting. An unprotected TOP accident in a small reactor could progress into the transition phase unless early termination and permanent in-place cooling is established.

A progression toward gross disruption without energetic excursions may be less probable for larger reactors where early sodium voiding could insert enough reactivity to lead directly to a disassembly excursion. It still would probably not be possible to exclude less severe accident sequences that could proceed into disruption without going directly into disassembly.

An additional mechanism for producing transition-phase conditions could be subassembly-to-subassembly failure propagation. This might arise if a subassembly became undercooled, causing

sodium voiding and fuel melting. If corrective actions were not taken, the ducts of the subassembly could melt and the molten materials could attack neighboring subassemblies. This could produce voiding and subsequent melting in the neighboring subassemblies, leading to a growing molten region.

The analysis of the dynamic behavior of fuel-steel mixtures under various heating rates is necessary for any sequence including a transition-phase. There are two aspects of this phenomenon for which experimental information would be useful. The first of these is a knowledge of the flow regime that the molten materials would assume under the range of conditions that can be encountered. This is essential in predicting the geometrical distribution of the materials, which in turn is required to calculate the net reactivity of the system.

The second critical aspect is the heat transfer rate in all directions. This is needed for calculating an energy balance and hence the temperature of the mixture. The temperature of the mixture is undoubtedly one of the key parameters that influences the flow regime of the materials (e.g., do the materials remain dispersed or collapse into a dense puddle under a given set of conditions?). The heat transfer rates are also needed to predict the melting attack on the surrounding structure. This is required in predicting the ultimate dispersal of the core materials following a transition-phase accident sequence.

Another phenomena of potential importance is pressurization of the mixture, perhaps by entrained sodium, which could accelerate

dispersal. Finally, the migration and possible separation of the fission products could have a significant effects on the long term heating rates within a mixture of core materials.

Another phenomena of potential importance is pressurization of the mixture, perhaps by entrained sodium, which could accelerate dispersal. Finally, the migration and possible separation of the fission products would have a significant effect on the long term heating rates within a mixture of core materials.

The above phenomena must be understood for a range of conditions. Some of the key parameters are:

- (1) The heating rate (power level) of the mixture. Heating rates ranging from the decay heat level up to levels produced in moderate excursions should be considered.
- (2) The boundary conditions on the molten region. Regions from the size of a single subassembly up to the entire core region should be considered. The conditions at the boundary surfaces, especially the number and disposition of leakage paths, can vary from very porous to completely sealed.
- (3) The composition of the fuel-steel mixture (especially the fuel-steel ratio).

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5. Burst Phase (Core Disassembly)

Another area of phenomenology that is important in accident analysis involves the response of a reactor to a rapid prompt-critical reactivity insertion. Under these conditions the fuel temperatures rapidly increase. Depending on the magnitudes of temperatures, high fuel pressures, and possibly fission product pressures, create an internal pressurization that can rapidly disperse the core, leading to a termination of the excursion. When the pressure gradients created are large compared to the structural resistance of the reactor, then the motion of the core materials can be treated by assuming that the core materials behave as a homogeneous fluid. This is the basic assumption in the standard disassembly techniques.¹

The primary objective of analyzing the burst phase of a postulated accident is to estimate the fission energy produced during the burst. This is influenced by the system conditions at the beginning of the excursion (power level, temperatures, degree of sodium voiding, etc.) the rate of insertion, and the feedback effects that develop during the transient. The main feedback effects are due to Doppler broadening of cross-section resonances and to material motion. The negative Doppler coefficient will generally reverse the reactivity increase and cause the power to drop, and this is followed by permanent termination of the excursion resulting from movement of the fuel.

As excursions become milder, additional phenomena must be considered. Heat transfer from the fuel can become significant as the

time scale for disassembly increases. Although the fuel pressures can decrease as a result of this cooling, pressures could also develop as a result of the sodium and/or steel in the core. However, significant heating of these nonfuel materials is unlikely on the time scale of even a mild disassembly unless a very rapid heat transfer mechanism is in effect.

Structural effects could also become important during mild excursions. For example, if pressures do not exceed the duct strength for a period of time, then the presence of the intact ducts could constrain the radial motion of the molten material. Further if the pressures generated in the core region are lower than the strength of the surrounding structure, then treating this structural material as an unrestrained fluid is again invalid. As milder excursions are considered, there comes a point when it is no longer suitable to treat them with a disassembly model. The main phenomena of interest in core disassembly analysis are those associated with relatively severe excursions with some consideration of those effects that allow extension of the methods to milder transients. Excursions initiated by reactivity insertions on the order of \$10/sec or more are normally appropriate for disassembly analysis, with those above about \$50/sec requiring no special modification to the standard hydrodynamic treatment.

(1) Equation-of-State

The equation-of-state of the reactor fuel, including any contribution of fission products, is needed to assess the internally generated pressures that may disassemble the core

under prompt-burst conditions. In those situations where significant heat transfer from the fuel to the other core constituents (such as sodium and steel) can occur, their equations-of-state would also be required. The contribution of the fission products to the overall fuel pressure may be of great importance because fission product pressures are typically much higher than fuel vapor pressures under conditions of thermodynamic equilibrium.² Factors other than the equilibrium pressures are also important, however, and are highlighted below as separate phenomena.

(2) Fuel Motion Under Burst-Heating Conditions

In its classical form, the equation-of-state relates the pressure of a material to its temperature and density on internal energy and density. The functions used in disassembly analysis are derived by consideration of physical quantities such as compressibilities, coefficients of expansion, latent heats, heats of compound formation, estimates of critical values and when available, direct experimental data. The pressure gradient, as calculated during the power transient establishes the accelerations and consequent velocities and displacements of the core material. Experiments and calculations of several relatively uncomplicated systems (Godiva-uranium metal fuel, Snaptrap-ZrH₂, UH₃ fuel, Kiwi-graphite core, Kewb-water solutions)^{3,4} give encouraging evidence that the scheme can be made successful with a given amount of adjustment to the equations of state. Of the four reactors mentioned three

had homogeneous (or nearly so) cores and the complicating factor of cladding containing the fuel until pin internal pressures were high existed only for the Snaptran reactor. It may be the case, for example, that with $\text{UO}_2\text{-PuO}_2$ fuel within intact pins only for extreme temperatures will the classically derived equation-of-state be appropriate and an "effective equation of state" would have to be used for lower temperatures.

An "effective equation-of-state" could be inferred from experimental measurements of the motion during power transients, either in-pile or self-driven. It may be that the effective equation-of-state and the classical equilibrium equation-of-state do not appreciably differ, but this needs to be established. An "effective equation-of-state" derived from experiments may be especially necessary if effects due to fission products are included in disassembly analysis.

(3) Fission Products

To include the influence of the fission products, we need to know their concentration and distribution at the beginning of the burst. This requires an understanding of the basic phenomena of fission product retention and migration. Considerable heating of the fuel generally occurs during the initiating phase of a postulated accident. The influence of these early accident stages on the retention and disposition of fission products needs to be understood.

For example, a certain fraction of the fission gas may be in the form of small bubbles migrating through the fuel

grains. Upon very rapid melting, fission gas in this form may not be able to pressurize until these small bubbles grow and/or coalesce into larger bubbles thus overcoming the large surface tension effects. Rate effects such as this could significantly delay the attainment of the equilibrium pressures under prompt-burst conditions.

(4) Fuel-Coolant Interaction

When an excursion is induced in a core with sodium present, the possibility of rapid heat transfer from the fuel to the sodium could influence the excursion as discussed earlier. This phenomenon is probably important only for a rather mild excursion where the time scale for disassembly is relatively large. If rapid enough heat transfer mechanisms could be established, however, the fission energies generated during certain excursions could be significantly reduced,² although work energy might be increased³. The main phenomena of interest here are the rate and nature of the fuel-coolant interaction process under prompt-burst conditions and additional experiments in addition to those completed⁴ may be necessary.

(5) Fuel-Steel Interaction

As with sodium, rapid heat transfer from fuel to entrained steel could also provide a pressure source during mild excursions. Effective equation-of-state and heat transfer relationships for fuel-steel mixtures are needed for use in disassembly analysis of these situations.

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6. Extended Material Motion

In analyzing a disruptive accident to the attainment of a permanently subcritical configuration, it may be necessary to describe the motion of molten core materials ejected from the immediate core region. The analysis of this extended motion requires the understanding of several phenomena, including the following:

The flow regime (or regimes) of the ejected material under the conditions of interest.

The interaction of this material with the surrounding structure; including such effects as freezing and plugging, condensation, and the response of the structure.

The interaction (if any) with sodium either in the upper subassembly structure or the above-core sodium pool.

The possible modes of material reentry (if any) back into the core region following an initial ejection.

The heat transfer rates, and hence the melting attack, as molten core materials encounter and possibly penetrate the surrounding structure.

The rate and mode of material ejection through various types of leakage paths.

The form of the material dispersal following ejection into the above-core sodium.

The form the extended motion takes can be a strong function of the accident conditions. For example, an energetic prompt-critical excursion could produce a massive ejection of high density

material driven by high pressures resulting in permanent shutdown while a more gradual accident sequence could result in a low density, two-phase material trying to penetrate the surrounding structure.

Some of the variable conditions to consider include:

The heating rate in the fuel (both fission and decay).

The amount of molten steel entrained in the dispersing material.

The degree of core region pressurization (some pressure can assure shutdown).

The possibility of a plugging of flow passages due to the freezing of molten materials ejected earlier.

One of the key phenomena in the extended motion area is the possible reentry of materials. The mode and degree of coherence of any reentry strongly affects the possible reactivity insertion rates associated with this type of motion. Parametric studies have been performed for severe reentry situations.^{1,2}

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7. Energy Partition and Damage Evaluation

As discussed earlier, the analysis of energy partition and damage evaluation involves determining what fraction of the thermal energy generated during the accident will ultimately be converted to work and what the result of this conversion will be in terms of damage done to the system. There are two basic aspects to this problem. The first is the determination of the pressure source. The pressure can arise from fuel and/or steel vapor generation, or from the heating and resultant vaporization of sodium. The second aspect is the propagation of these pressure loadings to important components in the primary system and their subsequent response.

There are two important domains in which interactions can occur. The first is the result of pin failure in nonvoided sub-assemblies, as usually indicated by analysis of transient over-power excursions. In this case the interaction would occur within the confines of the subassembly structure. The second would occur when molten materials ejected from voided core regions encounter sodium in the upper subassembly structure or the above core sodium pool.

Since sodium is a very effective working fluid, a rapid interaction and transfer of heat from the high temperature core materials could lead to a more efficient thermal-energy-to-work conversion than would occur from the expansion of the core material vapor. The interaction phenomena must be understood for a variety of conditions. Some of the important variations include:

The location of the interaction (e.g., within the subassembly structure vs the above-core sodium pool).

The flow regime and geometry of the core materials when they encounter the sodium, i.e., the flow regime of ejected material could vary from a low density two-phase flow to a high velocity single-phase slug.

The amount of molten steel. Because of its high thermal conductivity relative to fuel, large amounts of molten steel could increase the heat transfer rate to the sodium.

The temperature of the core materials. This could vary strongly depending on the energetics of the accident sequence.

The REXCO¹ system of codes has been shown to compute certain vessel damage reasonably well and is a valuable tool for the calculation of energy partition and damage evaluation. Tests performed at the Stanford Research Institute have been analyzed and the results have been applied to the analysis of the FTR containment system^{2,3,4,5,6}

The Pajarito Dynamics Code (PAD) has been used to calculate spherical sodium vapor explosions⁷ and the kinetic energies of a sodium column above a slab core⁸ which is postulated to experience a power excursion with heat transfer to sodium allowed. The results are thought to be reasonable insofar as the heat transfer coefficient to sodium is valid. Other experiments and analytical methods have been applied to European reactors.⁹

The basis for analytical prediction of damage therefore appears to be relatively well founded. The primary need for further experimental work is for a more precise characterization of the pressure-volume-time behavior of the pressure source for a variety of accident situations. This sort of information is expected to become more

precise as a result of phenomenological and integral tests discussed in this report. An increased understanding of the dissipative mechanisms which are expected to occur would help reduce some of the excessively conservative design margins which are required when these effects are ignored. A defensible characterization of bulk fuel/steel-coolant interactions, for example, in which pressure generation is milder than that which is thermodynamically possible, would ease the containment burden. Considerable work is going into this area and the results are promising.¹⁰

As the basis for predicting CDA behavior becomes more firm, further experimentation in the area of primary system structural dynamics may be useful. Also, as analytical techniques for predicting structural response are improved to consider responses to non-axisymmetric loadings, further experimental verification should be performed.

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8. Post-Accident-Heat Removal

Post accident-heat removal (PAHR), is the removal of decay heat following termination of a fission heating phase of a CDA. It is apparent that the ability to remove this decay heat depends upon the nature of the debris,¹ e.g., particle size distribution and the geometry of remaining core. The nature of the heat removal problem is determined by the preceding course of events.

Various considerations, related to understanding post-accident heat removal, are discussed below. These considerations are used to assess the needs for in-pile and out-of-pile experiments.

Decay Product Disposition and Behavior

Post-accident-heat removal considerations require knowledge of the magnitude of the fission-product decay heat as well as its distribution among the molten fuel and reactor phases as a function of time. Equilibrium calculations of decay product distribution between molten uranium oxide have been made and compared with experiment by Fischer, Schilb and Chasanov.² Some questions regarding the results were raised by Catton and Erdmann.³ Of more importance, however, are questions of chemical kinetics and lack of homogeneity in the decay product redistribution process. The process may be of such complexity that analysis is impractical. Lack of information regarding decay product distributions could require the PAHR systems to be designed overly conservatively.

Fission product gas is a consideration in the initiating phase of a postulated accident. Some of the fission gas will probably

escape into other reactor materials (sodium, molten clad) and decrease the decay heating level of the fuel itself. This could change the subsequent core disruption calculations and result in either more or less of the fuel inventory requiring containment (catching). Noble gases would probably be separated from that fuel as the fuel pins lose their integrity. Other volatile fission products may not be released instantaneously. Thermodynamic interactions might govern the final disposition, with kinetics determining intermediate conditions.

It is of interest to know the release of decay products from the fuel during the initiating and transition or burst phases. Also of importance is deposition on structural materials due to condensation of vaporized heat-generating fission products.

Fuel Debris Beds

Fuel debris beds are postulated to occur whenever molten fuel and sodium come in contact with one another. The debris could take on many forms from large chunks of fuel and steel immersed in sodium to fine powder. If the debris is coolable by being immersed in sodium, a bound on the temperature exists (the boiling point of sodium). If, however, the debris dries out, the fuel can melt and a more severe cooling problem presents itself.

In experimental investigations, characteristics of the debris, such as shape, size and percentage of steel, should be representative of what is expected in an accident situation. Out-of-pile experiments using induction heating or joule heating can be carried

out, but the situation is not the same as in a reactor. For example, inductive heating of the particles is particle-surface area dependent, and requires that the particles be electrically conducting and the coolant to be a nonconductor. This is the opposite of UO_2 -sodium mixtures. If one electrically heats the fluid, a UO_2 -sodium mixture can be used but the heating mode is very different. Preliminary experiments⁴ indicate that the heating mode is very important. It is believed that out-of-pile experiments can lead to an understanding of the phenomena and possibly even heat transfer correlations that are adequate. Confirmation, however, should still be obtained by some in-pile experimentation to be sure that particle shape, particle volume heating and other possibly unanticipated effects are adequately accounted for. Assuring coolability of debris in sodium requires knowledge of how the debris is formed, the effects of steel particulate or chunks on the cooling process, and dry-out of debris beds of various mixtures.

Preliminary experiments at UCLA indicate that debris bed experiments do not have to be too large. It is possible that experiments planned at Sandia in the ACPR might be sufficiently large. A good indication of whether or not a debris bed experiment is large enough is the number of bubble columns emanating from its surface. For a water-lead or water-steel system, four inches in diameter is sufficient. Bed leveling experiments will have to be larger in size. At this time, it is not clear how large. The 75KW induction heater at ANL is presently being used to run experiments on leveling and a 12-inch bed has been heated. This should allow some conclusions to be made in the near future.

The question always arises as to whether the debris being investigated is truly representative of a reactor situation. The ANL experiments⁵ used UO_2 particles resulting from a grinding operation. It has been observed at UCLA that many different types of debris result from an FCI. For example, under some circumstances a moss-like structure was found. In any event, some of the experiments should be conducted using debris which resulted from an FCI. Some effort should be expended to plan an experiment where the debris is formed in-pile and heated to dry-out in situ.

Molten Fuel Melt-Out

Downward debris relocation after neutronic shutdown is a complex process involving possible streaming or freezing with plugging. The rate of downward penetration is controlled by the rate of heat transfer from above. This leads one first to consider the adequacy of models such as the Jackson, Ostensen and Henninger boiling pool model (ANL).⁶ The Jackson, Ostensen and Henninger model establishes the initial disposition of the core, and knowledge of streaming or plugging leads one to estimates of how much of the core can be tied up in the core support structure.

Both in-pile and out-of-pile experiments should be carried out. Out-of-pile experiments, to our knowledge, have not been conducted to establish freezing phenomena in heat generating fluids. The use of micro-wave heating suggests itself for out-of-pile experiments with water as a working fluid. Other fluids may be also useful. Flow through channels as well as drainage films should be investigated. Confirmatory in-pile experiments might be necessary to be sure unknown factors are not overlooked.

As noted earlier the interaction of a molten region or "pool" with its surroundings requires knowledge of the heat transfer from it under a variety of conditions and the behavior of the bounding surface. Sufficient knowledge of heat transfer from a volumetrically heated pool is not available even for infinite horizontal layer geometry.

There are many possible conditions of the pool that should be understood. Lateral boundaries can cause convection patterns that will result in very high heat transfer at the pool upper edges. The possibility of either a mixture or separate layers of UO_2 and steel could cause differences in the heat transfer to the boundaries. The type of velocity boundary condition (liquid-solid, liquid-liquid or liquid-vapor) and temperature boundary condition (isothermal, non-uniform temperature or imposed stabilizing temperature gradient) can cause the heat transfer to be significantly modified. Another important factor is the possibility of thermal radiation within the pool. Most of these effects or conditions can be adequately studied out-of-pile. Confirmation of the studies could be obtained from in-pile experiments.

In that accident analysis requires a prediction of melting rates, one must also investigate melt-through of steel and other possible sacrificial materials. The actual process by which the bounding material is melted and transported away from the interface needs understanding. It is necessary to establish the rate at which the boundaries of the molten pool would melt and the effect of possible plumes of steel on the pool, as well as geometrical irregularities caused by the melting process.

If the molten pool is large enough, 8 to 16 cm deep, boiling will occur due to the volumetric heating (at 1% FFTF full power). The boiling process is very complicated and, even though one might understand the phenomena, an experiment requires boiling fuel to establish characteristics such as required superheat, effect of molten steel, etc. Some out-of-pile experiments have been conducted (I^2R heating of molten salt at ANL). Micro-wave heating may again be helpful. In any event, the out-of-pile experiments will require in-pile data to allow adequate modeling to be accomplished.

If a molten pool is oscillated, the downward heat transfer can be significantly augmented by disruption of the protective stratified layer. Further, the vibration induced shear can disrupt the solid fuel protective layer and expose the steel to very high temperature fuel. The question of how to characterize the disturbance is not answered at this time. This aspect needs analytical effort and it appears as if an as yet undefined in-pile experiment will be needed.

Ex-Vessel

At this time melt-through cannot be precluded to a desirable level of uncertainty if a large fraction of the core reaches the reactor vessel, and ex-vessel containment should be a strong consideration. Many of the uncertainties involved will be removed as the planned programs are conducted.

Material compatibility is a key part of the basic information that is needed. It is necessary that the fuel-steel-sodium-sacrificial material interaction be understood. Considerations

involved are bed material solubility (and the rate it goes into solution), bed melt rate, interface behavior and effect of bed cracks as well as crust formation and consequences. Thermodynamic considerations such as sodium reacting with basalt or concrete can be examined out-of-pile. Mixtures of fuel-steel-sodium and sacrificial material, however, will most likely have to be examined in-pile. The interaction and resulting pool growth may be slow, and long duration tests will be needed. The possibility of an interaction with concrete as a sacrificial material should not be ignored. Some refractory metals (tantalum and tungsten) are being considered as poisons in the core debris receptacle and need examination since they are soluble in stainless steel. Consideration of graphite would yield information for the GCFR as well as information necessary to design the heat exchangers surrounding the bed.

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9. Radioactive Material Transport

The ultimate consequence of concern for postulated core disruptive accidents involves the exposure of the public to fission product and heavy metal aerosols and vapors. In order to predict these exposures in an accident analysis, three possible stages of the post-shutdown accident sequence must be considered:

- (1) Generation of fuel vapor and fission products and their rate of escape from the primary vessel: This involves the release of fission products from the core and their interaction with sodium and structures, condensation of sodium, fuel and condensible fission product vapors, and rates of any leakage from the vessel. Fractionation of fission products may be important. Tacit in this study is an understanding of the mode and extent of vessel failure, if any.
- (2) Dispersal of this material within the secondary containment and concomitant rates of escape from the building: This involves characterization of radioactive particles in terms of size distributions and activity distributions. Some data exists indicating that a concentration of different isotopes exists on various sized particles. Hence, isotope and activity distributions with respect to particle size may need to be determined. The chemical form, or "in-vivo" solubility of the isotopes must be known to determine biological half life and the critical organ for dose determinations. An understanding

of sodium fires is also needed. Pressure buildup due to a sodium spray in either an inerted or air-filled reactor head cavity area must be considered.

- (3) Motion of the material under meteorological influence: Dispersal, settling, and resuspension under varying terrain and meteorological conditions must be considered.

Stage (1) has received little attention and only initial modeling has been performed. Stage (2), particularly with respect to sodium fires, has been studied extensively. Stage (3) has also received considerable experimental and analytical attention, particularly in the water reactor program, and will not be considered further in this report.

Computer codes such as VENUS-II might be used to characterize the state of the core immediately following a power excursion. Advanced disassembly and extended motion codes can eventually be used for analytical predictions of the amount and type of vapors produced in a given accident sequence. We know of few experiments,^{1,2} however, which have been performed to attempt to characterize the vapors (fuel and fission products) which might result from a fast reactor CDA. The experiments which have been performed are far from prototypical and suffer from scale effects. Modeling and experimentation^{3,4,5,6,7,8,9,10} relating to the behavior of CDA effluents within the primary containment are also preliminary in nature.

The behavior of CDA effluents within the secondary containment obviously depends strongly upon the manner and form in which they escape the primary containment. Nonetheless, some experimental work^{11,12,13,14,15,16} has been done in this area. Computer and analytical modeling has also been performed.^{17,18,19,20}

Uncertainties exist in the manner in which particle coagulation reduces particle concentrations within the containment. Sedimentation rates depend upon the surface to mass ratio which results from coagulation.

The adequacy of air cleaning systems and engineering controls is also a consideration, and a large impact on aerosol release rates may result. Accidents may present unique challenges to these systems, including high flowrates and temperatures and rapid loadings.

Although not specifically addressed in terms of in-pile facility needs in this report, consequence analysis involving meteorological dispersion, population dose, and radioactivity-induced population injury is important. The uncertainties in these areas may lead to orders of magnitude of conservatism in reactor accident risk assessment. These areas should be coupled to the safety program since they can have a tremendous impact on the priorities.

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C. Possible Trends for Large LMFBRs, Advanced Fuels, and GCFRs

1. Large LMFBRs

It is likely that large power-producing LMFBRs will be 1000 MWe or greater. It is also likely that the first of these large reactors will employ mixed oxide rather than advanced fuels, taking advantage of the more advanced oxide fuel technology available. The accident analysis methods developed for FFTF are only beginning to be applied to large LMFBRs. A study performed in 1972 did make an attempt at a quantitative assessment of accident trends and a comparison of accidents in large (1000 MWe and 4000 MWe) LMFBRs.¹ The basis for this study was a very early version of the SAS2A² computer code. A study is now in progress at ANL to determine both the applicability of the newer computer models as well as to attempt to understand the transient behavior of a large LMFBR.³

This section presents some speculations concerning large core accident behavior. The 1972 study is summarized and some of the results of that work are used to pose problems which require further study and modeling. The current large core accident studies are also briefly described and some experimental needs for large cores are identified.

Large Core Behavior Under Accident Conditions

For some core designs, the behavior of large cores could be similar to the behavior of an FTR-size LMFBR. On the other hand, cores with large, positive sodium-void reactivity worths may behave significantly different than smaller LMFBRs during a transient. For example, it is believed to be virtually impossible for an FTR-size core to undergo an energetic disassembly due to sodium voiding.⁴ Some large LMFBRs, depending on design, may have maximum sodium-void worths on the order of eight to ten dollars.⁵ If a situation were to arise in which rapid

voiding occurred before fairly high fuel temperatures were reached, the possibility exists that an energetic disassembly could occur before any appreciable fuel motion. Thus, sodium voiding might be the dominant mechanism for large core postulated accidents. On the other hand, the effects of primary loop coupling, axial expansion, scatter refueling, etc., could create a situation in which sodium voiding is insufficient to cause an appreciable burst. This is similar to FTR behavior in which a transition phase is more likely than an energetic disassembly.

Large cores, with a large amount of fuel and many critical masses, may have a higher probability of undergoing a secondary criticality after extended fuel motion. On the other hand, noncoherence arguments may be more effective, since a given mass of fuel would have a smaller reactivity effect.

Of course the speculations presented above are very dependent on core design. There may be significant differences in large core accident behavior resulting from future design changes. For example, such changes as placing the fission-gas plenum on the bottom of the core, removing the upper axial blanket, using thinner cladding, changing the outlet temperatures, and increasing the core height will affect the accident sequences and consequences. Changes to both the neutronic and thermal-hydraulic models may have to be made in the current computer codes such as the SAS system to model such large cores.

Previous and Current Work

As mentioned, a 1972 study was performed primarily to compare the accident behavior of a typical 1000 MWe LMFBR to that of two 4000 MWe cores designed explicitly for the study. Comparison calculations were performed for large and small values of sodium superheat in an effort to determine whether any anomalies can occur because of the choice of

superheat, as well as to look at the sensitivity of similar accidents to variations in superheat. Also, switches from predisassembly to disassembly calculations were made at three or four points in time (corresponding to different fuel temperatures) so that a determination of the uncertainties in the initiation of a disassembly calculation could be assessed.

The predisassembly calculations presented in the study were performed with an early version of the SAS2A computer code. The output of this code was then used as input for a disassembly calculation performed with the VENUS-II computer code.

All the calculations were performed assuming beginning-of-life fuel and thus neglected the effects which result from the presence of fission gas. A fuel-coolant interaction module was not included in the versions of the SAS-VENUS codes, nor was a fuel-slumping module. Also, the sodium voiding module did not treat the thickening, thinning, or removal of liquid film from cladding and structure under influence of gravity and shear forces due to streaming vapor after boiling occurs. The degree of sodium superheat was chosen to be either 10°C or 100°C for all of the transients studied.

Table I indicates some of the important parameters calculated and used as input for the three cores employed for the study. The four enrichment zone, large core permitted the effects of a high degree of radial power flattening to be assessed.

Most of the calculations were for unprotected LOF accidents and only those will be discussed here. In these LOF cases, the larger sodium-void effect in the 4000 MWe cores tended to shorten the time from boiling initiation to disassembly switch, as compared to the 1000 MWe

Table I. Reference Core Parameters

	1000 MWe Core	Two Zone, 4000 MWe Core	Four Zone, 4000 MWe Core
Core Geometry			
Active Core Height (cm)	109.7	127.0	127.0
Zone 1 radius (cm)	73.5	155.6	118.3
Zone 2 radius (cm)	103.9	230.3	168.1
Zone 3 radius (cm)	---	---	205.4
Zone 4 radius (cm)	---	---	230.3
Height-to-Diameter (active core)	0.53	0.28	0.28
Axial Blanket Thickness (cm)	38.1	30.5	30.5
Radial Blanket Thickness (cm)	34.8	24.9	24.9
Fission Gas Plenum Length (cm)	130.1	163.8	163.8
Thermal Hydraulics			
Thermal Power (Mwt)	~2500	~10000	~10000
Sodium Outlet Temperature (°C)	625.0	567.0	565.0
Core ΔT (°C)	168.0	163.0	161.0
Radial Peak-to-Average Power			
Zone 1	1.12	1.06	1.01
Zone 2	1.31	1.43	1.03
Zone 3	---	---	1.11
Zone 4	---	---	1.26
Core Average	1.22	1.25	1.10
Axial Peak-to-Average Power (Core Average)	1.28	1.28	1.28
Nuclear Characteristics			
Effective Delayed Neutron Fraction, β_{eff}	0.003297	0.003175	0.003167
Prompt Neutron Generation Time, Λ (sec)	0.328×10^{-6}	0.674×10^{-6}	0.673×10^{-6}
Doppler Coefficient (Tdk/dT)			
Na-in	-0.0067	-0.0088	-0.0088
Na-out	-0.0040	-0.0041	-0.0041
Whole Core Sodium Void Worth (\$)	3.84	5.67	5.94
Sodium Void Worth (no blankets) (\$)	4.59	6.38	6.68

cores tended to shorten the time from boiling initiation to disassembly switch, as compared to the 1000 MWe core. Higher reactivity insertion rates due to sodium voiding reactivity occurred in the 4000 MWe cores, and the disassemblies, which are sensitive to reactivity insertion rates, appeared to be relatively more severe for the 4000 MWe cores than for the 1000 MWe core. In the 1000 MWe core, reactivity insertion rates as high as \$177/sec were calculated while the rates for the 4000 MWe core were approximately \$80/sec higher.

Another important effect of core size is illustrated in Tables II and III by the amount of energy deposited in the molten fuel at the end of each disassembly calculation. This represents the amount of energy available if all the molten fuel were suddenly solidified. The difference in energy in molten fuel between the 1000 MWe and the 4000 MWe cores appeared to be directly proportional to the amount of fuel (volume) in the core.

An effect which is important to potential LMFBR design is the difference in the accident consequences caused by flattening the radial power profile of a core. The SAS-VENUS calculations indicated that a more highly flattened core exhibits a more coherent sodium voiding pattern, larger generated ramp rates, and a somewhat more energetic disassembly for the loss-of-flow accidents studied.

The effects of superheat variations on the loss-of-flow transients studied were consistent with the expected results. Boiling initiation in the hottest channel occurred approximately 0.60 second earlier in the 1000 MWe core, for the 10°C superheat case, than for the 100°C superheat case. Boiling initiation occurred approximately 0.50 second earlier in the four zone, 4000 MWe, 10°C superheat case, than for the 100°C case. The resulting sodium voiding progressed more slowly for the

Table II

1000 MWe Core Disassembly Results
(100°C Superheat, Loss-of-Flow and Overpower Transients)

SAS2A-VENUS Switch * Temperature (°C)	Case	Ramp Rate at Switch Point (\$/sec)	Final Energy in Molten Fuel (MW-sec)	Mass of Molten Fuel at Termination* (Kg)	Final Average Core Temperature (°K)
3500	Loss-of-Flow (SAS2A ρ /sec)	61.00	7.25×10^3	12,800	3500
	Loss-of-Flow (Max ρ /sec)	177.00	11.5×10^3	13,500	4250
5000	Loss-of-Flow (Max ρ /sec)	177.00	11.9×10^3	14,000	4900
6000	Loss-of-Flow (Max ρ /sec)	177.00	12.8×10^3	14,200	5600

* Total Fuel Mass (excluding radial blanket) $\approx$ 24,000 Kg

Table III

Four Zone, 4000 MWe Core Disassembly Results
(100°C Superheat, Loss-of-Flow and Overpower Transients)

SAS2A-VENUS Switch Temperature (°C)	Case	Ramp Rate at Switch Point (\$/sec)	Final Energy in Molten Fuel (MW-sec)	Mass of Molten Fuel at Termination* (KG)	Final Average Core Temperature (°K)
3500	Loss-of-Flow (Max ρ /sec)	256.00	4.7×10^4	54,000	4400
5000	Loss-of-Flow (SAS2A ρ /sec)	139.00	5.36×10^4	54,000	5150
	Loss-of-Flow (Max ρ /sec)	256.00	5.61×10^4	55,400	5200
6000	Loss-of-Flow (Max ρ /sec)	256.00	7.08×10^4	55,700	6150

* Total Fuel Mass (excluding radial blanket) $\approx$ 82,600 Kg

10°C superheat cases since less excess energy was deposited in the sodium prior to formulation of the initial bubbles. The 10° superheat accidents took longer to reach the high temperatures for disassembly switch than the 100°C superheat case, since the time rates of change of the sodium-void reactivity additions were smaller for the low superheat cases than the high superheat cases. However, maximum ramp rates for the disassembly calculations were only a few dollars per second different for the 10°C superheat cases than for the 100°C superheat cases. It appears that once sodium voiding has established itself in these cores, the course of the accidents is similar, regardless of the degree of superheat.

In summary, the work performed for this study indicates a number of important considerations for large core postulated accidents even though the study used an early version of SAS. Sodium voiding appeared to be the controlling mechanism for the accidents calculated. The use of the newer models now employed by SAS may, of course, affect this conclusion.

Work is now underway at ANL to study large core accidents and to compare the behavior of 1000 MWe oxide and carbide-fueled LMFBRs. This is discussed in the next section. In addition, work is being performed⁶ to study very large LMFBRs (4000 MWe) using newer versions of SAS than employed in the previous study. This study is very preliminary in nature, but initial indications are that initial disassembly energy generations are much less than with the earlier studies.

Experimental Needs as Related to Large LMFBRs

Much of the planning for new facilities can be performed relatively independently of core size. Studies of fuel motion, cladding motion, freezing and plugging, etc., are generic and results can be applied to

all reactor sizes. However, thought should be given to experiments, particularly integral experiments, which would simulate large core behavior. Secondary criticality effects should be considered carefully. In addition, the effects of sodium voiding on the accidents should be better understood since voiding plays a more important role in large reactors.

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2. Advanced Fuels

Future LMFBRs may employ mixed carbide or nitride fuels rather than mixed-oxide fuel. The higher average fissile atom densities of these "advanced" fuels allow higher breeding ratios than for oxide-fueled LMFBRs. In addition, the higher thermal conductivity of the advanced fuels offers the possibility of higher specific power than for oxide fuels. Such factors favor the eventual use of carbides and nitrides over oxides in large, power-producing LMFBRs. Relatively little is known about the physical properties of advanced fuels or their in-reactor transient behavior. Most of the accident studies that have been performed in the past have concentrated solely on the oxide-fueled LMFBR, and thus, analytical modeling and experimental verification for advanced-fueled LMFBRs lags behind the oxide LMFBR. Work is now underway at ANL to assess postulated accidents in large, carbide-fueled LMFBRs and to determine the applicability of the current analysis tools (the SAS computer code system) to the calculation of postulated accidents in carbide-fueled LMFBRs.^{1,2} (Carbides are now being analyzed since more data exists than for nitrides.) In addition, work on reactor design is also in progress at Combustion Engineering³ and on advanced fuel property studies at LASL,⁴ ANL,^{5,6} and Battelle Columbus Laboratory. It is certainly clear that basic developmental work is needed in both analytical and experimental areas to bring the knowledge of advanced fuels to parity with that of oxide fuels.

This section discusses the current status of accident analysis for advanced fuels (specifically carbides). Some of the typical differences between carbides and oxides are discussed in terms of their effects on possible accident sequences and consequences. In addition, the current work in progress at ANL concerning advanced-fueled LMFBR accident

analysis will be discussed and some judgments will be made as to the experimental needs related to advanced-fueled LMFBRs.

Since a major objective of the present study is to define facility needs, the emphasis in the advanced fuels discussion is on determining if there are areas with potentially more stringent requirements than with oxides. It should not be inferred that there is a prior judgment on the relative safety of oxides and other fuels; this judgment could only be reached after further work.

Advanced Fuel Properties

In order to understand advanced-fueled LMFBR behavior under postulated accident conditions, it is necessary to understand some of the physical properties of advanced fuels. Much of the following discussion will center on a comparison of advanced fuel behavior (specifically carbide fuel since more is known than for nitride fuel) and oxide fuel behavior. Much of this discussion will be speculative in nature.

The thermal conductivity of carbide fuel is approximately five to ten times greater than that of mixed oxide fuel.⁵ A number of design and safety consequences may result from this difference in the conductivity. The carbides are able to attain higher power densities (linear heat ratings) than the oxides while running at relatively colder center-line temperatures. Carbide fuels will probably be run considerably below melting and contain somewhat less stored heat than oxide fuel in an LMFBR. The higher thermal conductivity of carbide fuel may be responsible for departures from accident sequences now proposed for oxide fuels. For carbide fuel, the tendency may be for the fuel to freeze and plug more easily than oxides. Also, the higher thermal conductivity may enhance postaccident cooling under low heat flux conditions.

Many of the recent advanced fuel designs employ a sodium rather than a helium bond as found in oxide fuels. Sodium is used in order to take advantage of its high thermal conductance. The sodium bond may present certain new safety questions. If the fuel pin is vented, the bond may boil during an unprotected LOF before channel boiling occurs. This would significantly reduce heat transfer from the pins and change the sequence of events in an accident. The voiding of the bond may also act as an extra pressure source on the cladding, creating earlier cladding failure than found in oxide-fueled cores. A current reactor design employs approximately 6%, by volume, of sodium in the gap.⁷ The loss of this sodium bond may be a significant additional reactivity source.

Carbide fuel is known to release smaller amounts of fission gas under steady-state irradiation than oxide fuel.⁵ During a transient, it might be expected that the larger amount of retained fission gas in the carbide fuel may be responsible for earlier fuel dispersal than in the oxide fuel. Thus, the gassier carbides may behave differently than oxides during an unprotected TOP.

Although the high temperature equation-of-state for carbide fuel is not well known, it is thought that carbide fuel vapor pressures are somewhat lower than oxide vapor pressures.^{4,6} This is of importance in both reactor disassembly calculations and calculations of potential work energy available from FCIs. If carbides do, indeed, have lower vapor pressures than oxides then one would expect later fuel dispersal in reactor disassembly situations. The lower vapor pressures could also influence fuel failure thresholds under rapid transient conditions.

Possible Accident Sequences

Although few analyses have been made of advanced fuel LMFBR behavior

during postulated accidents, we can still provide qualitative judgments on the differences in sequences between these and oxide LMFBRs. Such speculations can be made, in general, for large cores and then specific differences in oxide and advanced fuel (specifically carbides) behavior can then be discussed.

Unprotected Loss-of-Flow Accident

As discussed in the previous section, an unprotected LOF in a large LMFBR, for any fuel type, could be considerably different than that currently predicted for an FTR-size reactor due primarily to the effects of sodium voiding. Early sodium voiding might tend to lead to rapid power excursions, and such a sequence would be more likely to terminate in a classical disassembly created by voiding, although as noted earlier, recent studies indicate this may not be the case.⁸ Fuel pin failure and fuel-coolant mixing in the nonvoided portion of the core would be more likely to occur during the sodium voiding phase.

In general, a large advanced-fueled LMFBR might tend to follow this kind of accident sequence. However, with sodium-bonded pins in the core, the bond might boil prior to coolant channel boiling (depending on whether the pins were pressurized or vented). If the bond sodium were vented to the coolant channels then the loss-of-heat transfer could lead to rapidly rising fuel temperatures. The initial fuel operating temperatures would determine whether fuel melting occurred prior to bulk channel boiling. If the fuel linear powers were comparable to those found in oxide cores then it is likely that initial channel boiling would occur prior to fuel melting. If the fuel pins were not vented then the pins would tend to pressurize. This pressurization, in combination with plenum fission gas pressure, could lead to early pin failure prior to

fuel melting. The early pin failures and possible channel voiding could potentially create a more energetic initial disassembly (due to sodium voiding reactivity insertions) than that resulting from sodium boiling alone.

Accident termination could occur in at least three different ways. A significant amount of fuel could be swept out of the core leading to early and benign termination. A second possibility is a relatively mild initiating accident phase, possibly involving a prompt critical burst resulting from sodium voiding, cladding relocation, or fuel motion. This could lead to neutronic shutdown due to inherent dispersal mechanisms such as release of retained fission gas. Plugging of the fuel channels could occur in this case and recriticality might be a question. Third, a classical energetic disassembly might occur if the reactivity insertion rate resulting from sodium voiding, fuel slumping, etc., were large.

Unprotected Transient Overpower Accident

Positive reactivity resulting from sodium voiding following initial fuel pin failures could lead to a more rapid power transient in a larger LMFBR than in an FTR-size LMFBR during an unprotected TOP. Fuel motion effects could compound the reactivity insertion leading to an early and energetic reactor disassembly with sodium still present in most of the core. On the other hand, dispersive fuel motion could also cause a strong negative reactivity insertion.

The differences between advanced fuels and oxide fuel could be significant in determining the energetics of the accident but might not lead to a significantly different accident sequence. Fuel failure thresholds could be somewhat higher for advanced fuels than oxides due to colder operating temperatures and more rapid heat transfer to the

coolant. On the other hand, sodium bond behavior could determine the failure threshold and lead to earlier failures than for oxides. If failures occurred prior to significant fuel melting, then the initial reactor disassembly might be dominated by sodium voiding reactivity rather than fuel motion reactivity. If fuel failures did not occur until appreciable fuel melting, with possible high fuel internal pressures from fission gas release, then the sequence of fuel expulsion into flowing sodium and fuel motion in the channel and in the pin might be the most important reactivity effect. The available work energy from a fuel-coolant interaction during and following disassembly potentially could be larger with the high conductivity advanced fuels than with oxide fuel. Again, accident termination could be through either sweepout, plugging and possible recriticality, or an energetic disassembly.

Current Work

As mentioned previously, work is now in progress¹ at ANL to evaluate the applicability of the SAS computer code system in analyzing hypothetical accidents in large LMFBRs fueled with oxides and carbides. This study involves two 1000 MWe LMFBR core designs modified slightly from the original Combustion Engineering design.³ Table I shows a comparison of some of the selected core parameters of the oxide and carbide cores used in this study. The models presently assume that the core consists of beginning-of-life fuel (ten full-power day burnup). The materials data is the latest available. The reactivity data, including reactivity tables, Doppler coefficient, etc., were calculated at ANL using the EXPERT⁹ computer code.

At present, only unprotected LOF calculations have been performed for both the oxide and carbide cores. These calculations have been performed without employing any models to calculate the effects of

Table 1

Summary of Principal Reactor Parameters

Oxide and Carbide 1000 MW(e) Cores⁽³⁾

	Initial Core	Carbide Core
Fuel in driver region subassemblies	(U,Pu) O _{1.96}	(U,Pu) C _{1.05}
Fuel in radial blanket subassemblies	UO ₂	UC
Number of driver region subassemblies (includes 35 control subassemblies)	547	505
Number of radial blanket subassemblies	270	246
Lattice pitch, inches	4.76	4.76
Number of fuel pins per drive region S/A	169	91
Peak linear pin power, kw/ft	15	30
Driver region S/A - Fuel rod clad OD, in.	0.27	0.37
Number of fuel pins per radial blanket S/A	61	61
Radial blanket S/A - fuel rod clad OD, in.	0.52	0.52
Driver region fuel rod spacing	wire wrap	wire wrap
Cladding material/thickness, mils	CW 316SS/15	CW 316SS/15
Bond	Gas	Na
Core pressure drop, psi	75	75
Maximum coolant velocity, ft/sec	23	26
Length of driver region, ft	3.0	3.0
Smeared fuel density, gm/cm ³ , Driver region	9.3	13.3
Isothermal Doppler coefficient, T dk/dT	-0.007	-0.008
Reactivity coefficient for voiding Na % $\Delta k/k$ (Complete core and axial blankets)	1.33	1.40
Reactor nozzle to nozzle coolant temperature rise, F	300	300

sodium film motion, clad motion, or fuel-coolant interaction. This was done to assess the importance of sodium voiding reactivity additions in these cores, as well as to establish a base-line for comparison purposes when the other effects are included. A simplified model to account for the thermal effects of sodium bond boiling was incorporated into the SAS code for the carbide core studies. A more sophisticated model is now being developed to account for both pressure and reactivity effects neglected by the first model.

Although the results of these studies are very preliminary they do appear to support some of the judgments presented earlier. The oxide core tends to exhibit channel boiling prior to the carbide core, while the carbide core has a more extended power burst sequence than the oxide core. These differences probably result from the lower steady-state temperatures and higher thermal conductivity of the carbide fuel, coupled with the drastic decrease in bond conductance when the sodium bond boils. The same carbide calculation without sodium bond boiling resulted in higher channel voiding rates and a faster transient in terms of the time it takes to increase the fuel temperatures to a specified level. Thus the loss of bond, for this case, acted as a delaying effect by decreasing heat transfer to the channel sodium. In addition, reactivity ramp rates were also lower for the bond boiling case than the non-bond boiling case.

Experimental Needs Related to Advanced Fuels

Substantial developmental work is needed in advanced fuels behavior. There is a great deal of uncertainty in the limited physical properties data which exists. Accident analysis modeling in the area of advanced fuels is in an embryonic state and must be improved.

With the uncertainties associated with advanced fuels, their impact

on safety test facility needs is difficult to assess. If emphasis should be placed on addressing the safety questions which relate to a near-term commercial breeder using oxide fuels, then advanced fuels testing could be delayed. On the other hand, since serious consideration is being given to advanced fuels for advanced commercial designs, new safety test facilities should be able to contend with the appropriate tests. The primary concern of using advanced fuels in a safety test facility is that higher test power densities would be needed in the test fuel to simulate advanced-fuel high-power operation. Also, fuel pins will probably be somewhat larger than in oxide reactors and a test vehicle using the same number of pins would need to be slightly larger than for the oxide tests.

In summary, we do not expect great differences in accident phenomenology or sequences for a large carbide-fueled LMFBR as compared to a large oxide-fueled LMFBR. We have not considered nitride or other types of fuel. Since, as has been noted and is reviewed in Reference 10, nitride fuel decomposes at high temperatures (depending on nitrogen overpressure), then significantly different accident sequences might result with nitrides. This could be very significant and should receive early attention. In this, as in most of the questions concerning advanced fuel behavior in accident conditions, much of the initial efforts should be directed at out-of-pile and in-pile phenomenological experiments to develop a basic understanding.

Advanced oxide fuel pin designs should also be considered in facility requirements and designs. Design options which might be used in advanced oxide fuel pins include higher-strength (and thinner) cladding, higher smeared-density-fuel, larger-diameter pins, higher linear-heat rating, longer active-fuel-length, and fission-gas-plena below the core.

We would also not expect large differences in accident phenomenology or sequences with these kinds of design changes; however, these should be investigated further.

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3. Gas-Cooled Fast Reactors (GCFRs)

The designers of GCFRs have argued that any event leading to large scale core melting is sufficiently improbable that it should be neglected.¹ They have similarly argued that rapid reactivity insertion transients leading to disassembly type events need not be considered.² However, the Regulatory Staff and the ACRS have each identified both events as requiring attention in further review of the concept.³ The Reactor Safety Study, WASH-1400 (the Rasmussen Report) confirms the justification of this concern. In order to assess public risk all potential accident paths, including those leading to core meltdown, must be evaluated.

At this stage, it appears appropriate to predicate at least part of the safety research for the GCFR on the basis that both meltdown and core disassembly will be investigated. Problems related to post-accident-heat-removal are addressed, in part, in a separate portion of this report. Differing modes of core meltdown (including a postulated overpower or superprompt-critical burst) may lead to differing situations for any post-accident-heat-removal system. The presence of the contents of the secondary system (water) in the PCRV must be considered for some sequences.

With regard to reactivity accidents, a rather extensive examination of all possible modes of rapid reactivity insertion (within the general region of the original core location) seems to be warranted. Some preliminary analysis has been performed,² but the designer has generally identified only mechanisms (gravity) which will lose reactivity if melting occurs. Others³ have identified reactivity insertion mechanisms and have performed disassembly analyses.

There are preliminary indications³ that an originally intact PCRV can withstand a substantial energy release (with regard to mechanical effects). If this is proven, it is conceivable that emphasis need not

be placed on safety research experiments (or analysis) whose intent is to narrow the arbitrariness in analysis or provide a mechanistic basis for setting burst magnitude. If it can be decided that given a conservatively large disassembly, PCRV integrity can be assured as a design basis, it may be that the subsequent events will be most important in how they impact on post-accident-heat-removal capability (and perhaps the interaction of the steam generator contents).

In the absence of (1) a better understanding of the detailed behavior of the GCFR during severe accidents and (2) an "accepted" safety philosophy, it is not clear that a major experimental program to understand mechanisms important to core melting and disassembly is required that is significantly different from that provided for LMFBR, even assuming that the safety approach includes the need to design for disassembly.

The emphasis in this report will be on experimental programs for near-term commercial LMFBRs. We believe that, for the most part, experimental facilities designed for LMFBR conditions will be useful for GCFR-related experiments. The primary modification which would be required for GCFR safety tests would probably be provisions for high pressure gas-cooling of the test fuel as opposed to sodium cooling.

An experimental program currently is underway for GCFRs. One program, the Core Flow Test Loop (CFTL)⁴ will test the transient behavior of core elements, local and general flow blockages within an assembly, and pin-to-pin failure propagation. A proposed facility, the Gas Reactor In-Pile Safety Test Loop (GRISTL)⁵, could be used to test fuel, control and blanket assemblies to reduce overly stringent design margins. The GRISTL test program is expected to begin in 1980 at a cost of around \$40 million. These experimental programs are probably adequate until safety test needs are demonstrated which are outside the scope of these programs or which are not treated in the LMFBR program.

References

IV. C. 3

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V. Experiment and Facility Definitions

Summary and Discussion

This chapter describes experimental and facility characteristics for a class of studies which span the identified experimental needs and which could potentially resolve the important safety uncertainties that are currently projected as generic. The following areas are generally discussed in each case, with quantitative estimates made to the extent currently possible.

Key Phenomena of Interest

Experiment Size Requirements

Data Acquisition Requirements

Test Frequency

Prototypicality

Total Energy Requirements

Steady State Operation Requirements

Period Requirements

A brief discussion accompanies each area in each class of experiments. The discussion includes the reasoning used to arrive at the specific figures quoted.

A digestion of the above mentioned discussion follows in which an attempt is made to specify characteristics for four facility types on the following basis:

1. desirable - If the facility had these characteristics, the analyst would require no further information upon which to base his judgments.
2. acceptable - This represents a compromise and allows for a more realistic design target.
3. necessary - If these specifications cannot be met, the facility is not worthwhile and should not be built.

The four facility types are:

A: This facility is required for determining failure thresholds and characteristics for fuel and cladding in more or less prototypical fuel pins under a variety of transient conditions. The ability to precondition fuel pins, preirradiated elsewhere, and to then impose a burst sufficient to fail pins of prototypic enrichment is desired. The number of pins in the test region need not be large (≤ 37) but instrumentation to resolve fuel motion at the single pin level is needed. The axial location of pin failure and early stages of fuel and clad motion should be observable. Such a facility would be used for examining the behavior of clad and fuel during both protected and unprotected transients. It would furnish a data base for the analysis of the initial stages of any postulated CDA and would be used for examining possible autocatalytic modes of fuel failure.

B: This facility would be designed for the study of gross fuel/steel motion within a subassembly, including possible failure of subassembly walls, and for a limited class of observations on radioactive material transport. As compared to facility A, the ability to accommodate more test fuel is needed, but prototypicality of pins and accuracy of observation is less important. The ability to build in a source of decay heat is desirable but could perhaps be replaced by fission heating. A burst capability is required. Enriched test fuel is acceptable but would be difficult to preirradiate elsewhere.

C: Experiments involving substantially more test fuel than a single FFTF subassembly may be important but difficult or impossible to perform in a type B facility. It is therefore recommended that a few self driven integral tests be considered for simulation of whole core involvement. In particular, such tests could be valuable for checking detailed neutronic-hydrodynamic calculations or for verifying the existence of a dispersive transition phase. As few as seven FFTF subassemblies with enriched fuel could be self driving

(i.e., critical) but it may not be possible to suitably mock up feedback effects including the Doppler and sodium void coefficients in such small systems. These possibilities deserve further study.

D: Experiments to test post accident heat removal are viewed as requiring no burst capability and low power densities in extended geometries. For these reasons, such experiments may be best performed in a separate and relatively economical facility.

The extent to which these capabilities are embodied in existing facilities is examined in Chapter VI. However, to anticipate results presented there, the existing facilities permit a range of valuable experiments but do not fulfill individually or collectively the requirements for any of the four facilities. In particular, the type A facility needs a preconditioning capability which TREAT and PBF lack as well as a harder spectrum in the test region. Both the type A and B facilities require burst and viewing capabilities which the SLSF and FFTF could not provide. There are no existing candidates for type C or D facilities.

It is possible that many of the experiments suggested for type A, B, and D facilities could be performed in a single, although perhaps cumbersome and costly facility. We suggest, however, that the characteristics of these facility types are somewhat distinct and incompatible so that it is reasonable to consider a mix of simpler facilities each designed for the efficient execution of a more limited class of experiments. At the very least we believe that it will be fruitful to carry out preliminary conceptual designs for each type of facility. Their characteristics can then be compared in more detail. It might, for example, then appear attractive to first construct a type A facility for testing of prototypic pins and to later replace the core by one having a larger test region for experiments using larger quantities of enriched fuel as in a type B facility.

A. Experiment Specifications

1. Fuel-pin Failure and Initial[†] Cladding and Fuel Motion

This category involves a wide range of experiments focusing on the isolated behavior of fuel pins from steady-state conditions, through cladding failure, and into an initial cladding and fuel motion phase. This would include most of the TREAT experiments performed in support of FFTF. It is mandatory for most of the experiments in this class that detailed diagnostic test data be available. It is also necessary that in-pile testing be accompanied or, better, preceded by extensive analysis development and out-of-pile experimental programs. An understanding of the factors involved in fuel-pin failure and initial fuel motion should be developed at a high priority level.* If detailed models can be developed based on available information, then in-pile testing can be aimed more towards proof-testing or calibration of the models rather than towards providing extensive data for direct experimental correlations.

a. Phenomena of Interest

(1) Fuel pin failure threshold, location and mode for all relevant conditions and fuel types.** Mechanisms and phenomena which should be addressed include the following:

Fission-gas induced transient fuel swelling.

[†] The terms "initial cladding and fuel motion" are used to mean any time from the very beginning of cladding and/or fuel motion to the complete loss of fuel-pin geometry.

* Analytical and experimental methods developed to date have failed to provide an adequate understanding of pin failure threshold or location and the uncertainties in potential ultimate accident consequences.

** By "fuel type" we mean the character of the fuel and cladding result in sults from its steady-state power density history, its irradiation period, its chemical composition and its associated cladding alloy.

Transient fission-gas release.

Fuel-cladding relative thermal expansion.

Fuel-cladding gap/bond behavior.

Fuel-cladding mechanical interaction (effective loading and stress concentrations).

Cladding-coolant heat transfer (boiling and local dryout).

Fission gas release through cladding failures.

Fuel vaporization and equation-of-state.

(2) Initial fuel motion under all relevant conditions, e.g., low power, high power, voided, non-voided, etc. Mechanisms and phenomena which may be involved include:

Fuel pellets or pin mechanical behavior after cladding failure (bowing, cracking, fragmentation, collapse).

Solid/molten fuel and fission-gas ejection through cladding failures.

Fuel slumping.

Fuel vapor induced motion.

Steel entrainment in fuel (boilup, dispersal).

Volatile or gaseous fission product effects on fuel motion (foaming, frothing, and dispersal).

Fuel-steel thermal-hydraulic interactions with coolant (cladding motion and entrainment, fuel-cladding sweepout, rapid sodium vaporization, vapor explosions).

Spacer (wrapper wire, grid) effects on motion.

Effective equation-of-state of fuel, steel, fission products and coolant mixtures.

Effect of incoherent failures.

b. Relevant Accident Situations

The following is a set of accident situations or physical conditions for which fuel failure and/or initial fuel motion are of interest. A single test facility should not necessarily be used for all these situations. Further, the situations covered by in-pile experiments should depend strongly on the adequacy of analysis methods and out-of-pile experiments.

(1) Protected transients (whole core).

(a) Undercooling (for example, pump coastdown, loss-of-heat-sink).

(b) Overpower (including effects of local hot spots).

NOTE: a facility for these experiments might also be useful for studies of fuel behavior under normal operational transients (power cycling, startup, shutdown).

(2) Mild unprotected transient overpower - a few c/sec to a few $\text{\$/sec}$.

This should include both reactivity-insertion-initiated accidents and other accidents which include reactivity insertions (such as due to voiding during a loss-of-flow accident). The possible experiment conditions should include both low and high flow rates and various sodium voiding configurations.

(3) Rapid unprotected transient overpower (a few $\text{\$/sec}$ through dis-

assembly conditions). The reactivity insertion rates in this case would be presumed to be initiated by events occurring during an accident. Again, this implies a variety of thermal-hydraulic conditions.

The upper limit of insertion rates is not known, but may be as high as $\text{\$100/sec}$. In these extreme situations, a primary interest would be in the effective equation-of-state, and particularly, fission product effects and thermal interactions.

(4) Undercooling conditions (near or below normal operating power density). We include in this category both local and whole-core undercooling conditions in which, again, the emphasis in the experiments is on detailed information of fuel-pin failure and initial cladding and fuel motion. Many of these situations involve sodium boiling (or voiding for other reasons). Since the voiding dynamics will have an important effect, thermal-hydraulics should be reasonably prototypic. Some of the specific undercooling situations are given below. There is considerable overlap in the phenomena involved in these.

(a) Flow blockages. Much of the work in this area should be done out-of-pile as in the Failed-Fuel Mockup Loop (FFM) at ORNL.⁽¹⁾ However, questions concerning cladding or fuel motion could require some in-pile testing.

(b) Rapid flow reduction. This includes severe pipe rupture accidents in which flow rates decrease rapidly and the sodium saturation temperature is neared or reached in about one second. In a pipe-rupture-with-scrum accident involving limited sodium boiling, fuel-pin failures and thresholds for loss of core cooling are of primary interest.

(c) Gradual flow reduction. This includes an undercooling transient initiated by primary pump coastdown. Without protective action, this may lead to a sequence of sodium boiling, cladding melting, and fuel motion such as has been analyzed extensively for FFTF.² For larger reactors, in which sodium voiding reactivities might lead to a significant power transient, this category would overlap the transient

overpower categories. Initial cladding and fuel motion is of primary interest.

(d) High inlet temperature. This includes loss-of-heat-sink transients initiated in the intermediate or steam systems. Without scram, some of these transients could lead to a sequence resembling a loss-of-flow without scram but with a high core flow rate. This could influence initial cladding and fuel motion.

(e) Gas bubble passage. This is of interest as both an undercooling and reactivity insertion mechanism, particularly in designs with fission gas plena below the core. These designs should have fewer pin failures in the plenum region; however, verification that slow gas release would not lead to other cladding failures might be necessary.

c. Data Acquisition Requirements

- (1) Excellent visual resolution is required, i.e., some means of reconstructing detailed material configurations as a function of time.* A resolution of about 1 mm radially (the characteristic size of a coolant channel) and about 5 mm axially (the characteristic size of a fuel pellet) is desirable for a 19-37 pin array. Time resolution would depend on the particular transient of interest, and would be as small as a few tenths of msec for EOS type experiments.
- (2) Detailed thermal information is required. Fuel temperatures should be known within 50-100 K. The temperatures could be pro-

* We use "visual" throughout this report to indicate use of devices such as a fast neutron hodoscope or flash x-radiography to provide images of material motion.

vided by analyses calibrated with direct measurements in a few experiments; however, as many thermal-hydraulic measurements as possible should still be used in each experiment.

(3) Gas release rate measurements will be desirable for some experiments. Pressure transducers will be necessary in all experiments. These should have accuracies near 1% of the expected peak values in a given experiment.

d. Experiment Frequency

It is expected that a large number of experiments in this category would be performed. Many different conditions are relevant and for each a number of experiments would be necessary. A facility might be expected to be used for 20-40 experiments per year. This would obviously depend on the need for lengthy preconditioning at steady-state power in the test reactor.

e. Prototypicality Requirements

(1) Preconditioning.

For the experiments in this category, in which fuel pin behavior is of primary interest, preconditioning in the test reactor is an important consideration. Here, preconditioning is defined as operation at normal operating power density a few hours to a few weeks in the test reactor without a shutdown prior to a test transient. We use preirradiation to mean irradiation on the order of months to achieve a significant burnup and buildup of fission products. This could be in the transient test reactor but more likely would be in a steady-state irradiation facility such as EBR-II or FFTF. If preirradiated in a separate facility and tested without preconditioning, the test

fuel would have (1) unhealed shutdown and startup cracks, (2) a non-equilibrium fission gas distribution, and (3) a lower fission product decay heat generation density. The last of these is not significant for this class of experiments; however, the first two may be quite important in determining fuel-pin-failure thresholds in transient overpower situations and could also influence initial fuel motion.

Ceramic fuels such as mixed oxides tend to crack easily in tension at lower temperatures. Near the melting point the fuel tends to be ductile and exhibit a viscous flow behavior.³ During startup, cracks develop in radial-axial (r,z) planes in the fuel pellets as the hotter and expanding center forces the outer part into radial tension. These cracks may initially extend to the pellet center-line, but as operating temperatures are reached the cracks tend to heal.⁴ Healing is rapid at higher temperatures (perhaps minutes at 2500 K, hours at 2000 K, days at 1700 K) and does not occur below perhaps 1600 K. During an initial startup with fresh fuel the crack healing process occurs simultaneously with porosity migration and grain growth (restructuring) in fuel pin regions operating at linear powers above about 300 W/cm (9 kW/ft). This leaves the pellets in high-power density regions with the characteristic structure of a central void, columnar grain region (above about 2000 K), equiaxed grain region (from about 1600 K to 2000 K), and unrestructured fuel (below 1600 K). Roughly speaking, startup cracks heal to the outer edge of the equiaxed region. The crack-healing and re-

structuring processes also tend to "weld" the fuel pellet interfaces together forming a somewhat rigid fuel-pin column.

During cooldown from steady state, the fuel pin develops cracks in circumferential-axial (θ, z) planes as the center fuel contracts. Cracks also tend to develop at the original pellet interfaces.

Whether unhealed startup and shutdown cracks have a determining effect on fuel pin failure characteristics is not known at this time. Nonprototypic effects can be identified, but only speculations can be given as to their importance. For example, the pellet mechanical behavior and interaction with cladding is affected by unhealed cracks. In a transient overpower situation, the outer edge of the equiaxed region may form a strong "bottle" and transmit gas pressures from the interior of the pin to the cladding through cracked unrestructured fuel. The presence of unhealed radial cracks allows the possibility that fission gas released in the interior of the pin may escape prematurely. Also, cracking during shutdown and startup may release some of the gas retained in the central void or fuel so that the equilibrium gas distribution is disturbed.

Unhealed circumferential cracks may affect temperature distributions. If the gap is appreciable then a low gap conductance would cause higher central fuel temperatures than prototypic. That this could be a noticeable effect was indicated by post-mortems of the H-5 experiment in TREAT. In the central pre-irradiated (but not preconditioned) pin, it was noted that fuel melting extended neatly up to a well-developed circumferential crack. 5

Again, whether these effects are ultimately important in determining fuel failure is not known. It might be possible to resolve this through a combination of analysis, out-of-pile experiments, and experiments in existing in-pile facilities, but the ability to precondition fuel for the order of one week must be a requirement for a new facility to be used for experiments in this category until further information is available. We note that out-of-pile direct electrical heating experiments⁶ have the potential to examine cracking and crack healing and perhaps the effects of cracking on fission gas release. Also, fuel can be preconditioned in the SLSF loop in the ETR, and slow overpower transients might examine some of the effects. These kinds of experiments could be useful in the development of a mechanistic fuel behavior model including all the effects of preconditioning.

(2) Cladding-fluence-to-fuel-burnup ratios

An important factor in the design and use of a transient test facility is the enrichment of the test fuel. Here we use enrichment to indicate the total fissile atom ($^{235}\text{U} + \text{Pu}$) to total heavy metal atom ($^{235}\text{U} + ^{238}\text{U} + \text{Pu}$) ratio. The mixed oxide driver fuel used in FFTF is unenriched in ^{235}U and has a $\text{Pu}/(\text{U}+\text{Pu})$ ratio of about 22% in the inner zone and 27% in the outer zone³. Commercial LMFBR's could have enrichments as low as perhaps 12%.

Use of these relatively low enrichment fuels is difficult in a safety test facility since the energy generated in the test fuel is proportional to its enrichment. On the other hand, use of higher enrichment irradiated fuel means that the test fuel will have a nonprototypic cladding-fluence-to-fuel-burnup ratio and possibly a mismatch between cladding failure characteristics and transient

loadings as compared to a prototypic pin. Steady-state cladding and fuel swelling, gap closure, and gap reopening late in life will also be different than for a prototypic pin.

Whether these are determining factors for fuel failure thresholds is not known at this time. This uncertainty coupled with the difficulty of irradiating high enrichment pins in FFTF requires a new test facility to permit failure threshold tests to be performed with irradiated pins of 25% enrichment (this assumes about 10% burnup or a 2% enrichment loss). This will allow the direct use of high-burnup high-power irradiated FFTF outer-zone driver fuel. It would also be useful, and perhaps necessary, to perform experiments with 20% enriched (initially 22% with 2% loss due to burnup) irradiated pins from the inner zone. It should also be recognized that testing of even lower pins will be desirable.

(3) Thermal-hydraulic characteristics and test size

It is necessary that fuel and cladding temperature distributions be reasonably prototypic. In single-pin or small bundle failure-threshold tests this might be achieved even with a significant thermal neutron flux. A most important consideration is that temperature distributions be predictable and reproducible from one experiment to another.

For experiments extending beyond fuel-pin failure into initial cladding and fuel motion, it is necessary that the hydraulic characteristics be prototypic. This impacts more strongly on experiment assembly and vehicle design, but the test facility must be capable of accepting full-length pin experiments including pins with active fuel lengths as long as perhaps five feet.

In these experiments it is desirable to use as few pins as possible in order to maximize data acquisition resolution and test throughput. For failure threshold tests it may be adequate to use single pins. For tests extending into initial cladding and fuel motion it may be necessary to use bundles of 19-37 pins.

The central pin in a 19-37 pin array should be reasonably prototypic for initial cladding and fuel motion insofar as heat losses are concerned. This is based simply on the idea that this central pin would be protected by at least two outer rows of pins. On the other hand, if noncoherent voiding in a full subassembly is a dominant effect in determining initial cladding and fuel motion, then it may be difficult to simulate these effects in a small array.

In this case, a flat power profile is necessary in at least some tests in order to examine coherence of failure. Graded enrichments could be used to achieve this provided the enrichment variation was not large. It would be useful in some tests to use a higher power central pin or central seven pins in a 19 or 37 pin array so that data acquisition could be focused on fuel motion in a single or few pins. In general, improving data acquisition at the expense of a slight loss in prototypicality is beneficial.

f. Energy Requirements

(1) Transient energy requirements (above steady state) for this class of experiments can be given for two cases. First, consider an effective equation-of-state test in which it is desired to reach a peak fuel temperature of 6000 K in a rapid transient after first boiling sodium and melting cladding at steady power. Assuming the peak fuel temperature is at 2000 K prior to the burst,

then the energy needed during the burst is approximately given by

$$E_{\text{burst}} = C_{p1} (3000-2000) + H_f + C_{p2} (3700-3000)$$

$$+ H_v + C_{p2} (6000-3700)$$

$$C_{p1} \cong .35 \text{ J/gm-}^\circ\text{K} \quad C_{p2} \cong .548 \text{ J/gm-K}$$

$$H_f \cong 280 \text{ J/gm} \quad H_v \cong 1420 \text{ J/gm}$$

$$E_{\text{burst}} \cong 350 + 280 + 380 + 1420 + 1260$$

$$= 3690 \text{ J/gm}$$

This assumes complete vaporization of the sample and is therefore a conservative upper limit.

Another extreme can be given by slow reactivity insertion (transient overpower) or loss-of-heat-sink situations. These can be somewhat arbitrarily represented by assuming a linear power increase from steady-state to a factor of three times steady-state in 30 seconds. This would give a total energy requirement of 30 full-power seconds above steady-state operation. At a steady-state power density of 200 W/gm this represents an energy addition above steady-state of 6000 J/gm. The factor of three would lead to bulk boiling in peak subassemblies in core designs like the FTR or the CRBR. The 30 second figure is arbitrary but represents roughly the time to fuel failure in an unprotected reactivity insertion transient of a few cents per second. This also is a rough upper limit for what might be expected in transient overpower capability for a steady-state reactor such as the ETR.

(2) Steady-state operation requirements

We have stated that preconditioning is a requirement for most of the tests in this category. If preconditioning and the

implied indefinite steady-state operation are not provided, then tests in this category would be of questionable utility. Equivalent steady-state operation for around one minute should be adequate to perform the transient experiments following preconditioning.

g. Period Requirements

The burst width capability desired for the most rapid transient overpower experiments is very short. For example, the burst width (full width at half maximum) given by a \$50/sec VENUS-II sodium-out disassembly calculation (Fig. A) is about 3 msec (sodium-in disassembly burst widths would be even shorter). This insertion rate, although severe, is typical of those calculated with early models for sodium boiling during a loss-of-flow without scram accident in 1000 MWe class reactors.^{8,9} More recent analyses using newer models have given reactivity insertion rates in the \$10-20/sec range.¹⁰

Although a burst width of this magnitude is desirable, we do not know if it is necessary in terms of the phenomena involved. The development of pressures from fission products on a short time scale may be an important mitigating factor in a super prompt critical excursion. If further work indicates this to be important, then short burst width experiments may be necessary as proof tests. If it can be shown through analyses and/or out-of-pile experiments that time scale is not too significant at the millisecond level, then requirements on in-pile experiments could then be relaxed. Also consideration could be given to a small special-purpose facility to investigate effective Equation-of-State questions as the phenomena become better understood.

In summary, a burst width of 1-5 msec is desirable. If this cannot be achieved in a facility capable of meeting the energy and preconditioning requirements already noted, then out-of-pile experiments or a separate in-pile facility should be considered. The Sandia ACPR and SPR III reactors are candidates.

POWER

POWER (1000 MW)

FIGURE A

References

V. A. 1

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2. Gross Fuel/Steel Motion Within a Subassembly

The broadest class of extended fuel motion experiments include all of the events which allow the material within an assembly to either come to rest in a coolable configuration within the assembly or to become permanently dispelled. The particular class of experiments described in this section only relates to reactor designs which involve subassembly cans. Some commercial concepts might not utilize cans, and some concepts involve perforated cans. In these cases, experiments for extended material motion may involve larger test regions than those of subassembly size. A thorough understanding of extended material motion within a subassembly may preclude the need for as many experiments in ductless geometries.

Key Phenomena of Interest:

- a. Boilup - Do energy, mass and momentum transfer laws dictate that core materials always disperse immediately upon melting and for all time thereafter? Are energy loss mechanisms important, or do fast reactor materials tend to always disperse, if drastic power/flow mismatches are initiated. (If not, under what conditions can they collapse into a dense configuration?)
- b. Freezing/plugging blockages - Does material tend to freeze and plug flow passages on its route outward? How strong are the plugs, and what is the response of the plug-structure? How extensive are the plugs? When is in-place cooling not possible? Is there an impact of these events upon boilup? What is the behavior of blockages over "long" periods of time.
- c. Melting attack - What are the heat transfer rates under a variety of dispersion regimes, power density level, and boundary configurations?
- d. Reentry dynamics - If material is prone to reenter, how does it do so? (e.g., What is the flow regime, etc.) What are the controlling forces in reentry mechanics, such as vapor pressures and gravity?

How much material is involved in reentry?

e. Subassembly can integrity - Under what range of thermophysical conditions does the assembly wall remain intact, so as to direct material motion axially?

f. Threshold for mechanical removal of subassembly internals in the axial direction - What thermophysical conditions in the core region allow for rapid and permanent dispersal of fuel out of the assembly? Attaining these conditions may be impractical in some cases if the can integrity is lost before dispersal is achieved.

Data Acquisition Requirements:

a. Reasonably good visual resolution is required during the test, but primary emphasis is on observation of the entire assembly, particularly in the axial direction. Radial resolution of around 5 mm would appear to be adequate. The axial scale of interest includes an entire subassembly (around 12 to 14 feet). It may be possible to perform useful tests in which only portions of a subassembly are under "visual" observation. In any case, axial resolution of around 25 mm is probably adequate. Resolution of the relative motion of fuel, steel and sodium is most valuable here.

b. A broad knowledge of the thermophysical environment for corroboration of calculational models - Knowledge of material temperatures to within 100 °K is desirable. If these must be inferred through analysis, then energy input must be known to within about 1%. Pressure measurements to within 10^4 Pa (0.1 atm) or so, are needed, since pressures within the assembly fluctuate only slightly during some tests.

c. Temporal resolution - Because of the hostile environment within

the assembly during these tests, the survivability of in situ instrumentation is questionable. Nonetheless, the tests will be of limited value unless thermophysical and visual information can be gathered somewhat continuously during the tests. Hence, capability for gathering data in a few msec intervals would be desirable. Some tests may last up to a minute, or so, and variable data intervals which are keyed to on-line events would reduce the total amount of data to be gathered. This is not an essential feature, however,

d. Heat flux out the assembly - Knowledge of the heat flux as a function of axial position and time would provide additional information regarding the rate of disintegration of the wall from within. Coupled with thermophysical data, heat fluxes may allow estimation of the rate of vapor condensation on the wall.

e. Fission product release - Fractionation of fission products during the test can affect decay heat power levels. Hence, some measurement of the transport of these gases and vapors would be useful, but is not essential in all tests.

Test Frequency:

Fewer tests in this area are needed than in the pin failure area if data acquisition on each test is adequate. Stochastic effects may be of less important than in fuel failure tests. Nonetheless, a wide range of conditions are of interest and several tests are needed to cover this range. A very large number of out-of-pile phenomenological tests should be performed in order to increase our confidence in the calculational tools to be applied in the more complicated in-pile test. Perhaps 10 or so in-pile tests per year are sufficient here.

Prototypicality Requirements:

Preconditioning is needed in some tests, in order to build in short lived fission products. Simulation of decay heat with low level fissioning is possible for some tests, if these are performed in conjunction with true decay heat tests. Prototypic enrichment is not needed since the fluence/burnup ratio is not expected to strongly affect the phenomena important to gross motion of core material. For example, in fuel sweep-out type tests (overpower transient) the cladding failure location could be controlled by weakening the cladding or by other means.

A reasonably hard neutron spectrum is desirable to avoid non-prototypic heating in fuel near the assembly wall. Some flux peaking is permissible since the turbulence in most tests will tend to flatten the radial temperature profile anyway. Excessive graded enrichment within the test region is unacceptable since eventual mixing of materials introduces uncertainty in the test power profile.

Less than full radius assemblies may be acceptable as long as some region within the assembly is away from edge effects. Around 1/2 of a full assembly may be adequate. Initial thermal conditions within the assembly must be correct, since heat transfer phenomena are of primary interest.

Total Energy Requirements:

For ramp rates ranging from \$1/sec to \$50/sec, temporary survivability of assembly walls may be possible. Higher ramp rates may induce pressures which lead to assembly destruction. Hence, peak temperatures of interest are around 6000 °K. If we assume a steady state capability for the driver, the partition of energy in the class of tests becomes

$$E = E_{\text{burst}} + E_{\text{decay}}$$

$$E_{\text{burst}} = C_{p1}(3000-1700^*) + H_f + C_{p2} (3700-3000) + H_v + C_{p2} (6000-3700)$$

$$C_{p1} \approx .35 \text{ J/gm}^\circ\text{C} \quad C_{p2} \approx .548 \text{ J/gm}^\circ\text{C}$$

$$H_f \approx 280 \text{ J/gm} \quad H_v \approx 1420 \text{ J/gm}^{**}$$

$$E_{\text{burst}} \approx 450 + 280 + 380 + 1420 + 1260$$

$$= 3790 \text{ J/gm}$$

This energy density corresponds to the peak power at the axial midplane.

Note that for a 5000 °K peak temp, $E_{\text{burst}} = 3150 \text{ J/gm}$; a small increase in E_{burst} allows for substantial changes in peak temperature.

$$P_{\text{decay}} = .06 P_o = 12 \text{ w/gm.}, \text{ where } P_o \text{ is the nominal power density, } 200 \text{ w/gm.}$$

Another class of energy requirements are posed by an LOF sequence, followed by a burst which raises local fuel temperatures to 6000 °K and a one minute period of decay heat power level.

Currently, it has been argued core materials tend to disperse and remain so even for decay heat power levels (Boil-up). Energy requirements for such a test are modest, and can be easily accommodated by a steady state capability.

Period Requirements:

The limitation on energy deposition rates depends upon getting sufficient energy into the fuel before appreciable motion occurs due to

* This corresponds to the average fuel temperature at the axial midplane of the assembly.

** This number assumes total fuel vaporization, and hence is conservative.

gravity. The total burst width* of a \$50/sec excursion, starting at delayed critical, is around 27 msec for a Demo. sized core. Appreciable motion (several cm) does not begin until around 28 msec and becomes large by 30 msec. A burst which establishes the same longtime motion of fuel is all that is required. If material is allowed to fall from rest due to gravity no more than 1 cm, during the burst, this means a burst width of 50 msec is adequate. This burst width will be adequate for many cases of interest. A 5-10 msec capability might be desirable in some cases.

Steady-state Operation Requirements:

All tests in this class can be performed within one minute or so. Thirteen seconds at full power (200 w/gm) are needed to bring adiabatic fuel ($T_0 = 1700^\circ\text{K}$) to full vaporization. Some boilup experiments might be performed within one minute by rapidly bringing fuel to its melting or boiling point, followed by power levels at or below nominal for 30 seconds or so. If preconditioning is utilized to build in decay heat, long time, low power level heating is not required. A prototypical LOF sequence takes somewhat longer^{1,2} but on the order of a minute is sufficient³ to gather information needed to predict the longtime behavior of the material. Melting of steel and fuel blockages may take up to ten seconds and duct melting can occur in a time scale of 1-20 seconds depending on power level. These trends are expected to continue in larger, commercial reactors.

Comments:

Energetics, period, power profile (or neutron spectrum) and data acquisition requirements are quite severe and tend to be limiting in terms of defining a facility suitable for a broad range of safety tests.

A facility for performing gross motion experiments is not necessarily suitable

* Total burst width is defined as the time at which energy integration essentially ceases.

for fuel failure threshold and initial motion tests since we have not insisted upon perfectly prototypical flux profiles and fuel enrichments. This is the primary reason behind the separation of failure threshold and extended motion tests. A single facility to perform both classes of tests very likely could not be built in a timely and economic manner. Separate facilities may be more practical.

The large amounts of fuel required for extended motion tests impose relatively stringent reactivity control requirements upon a driver reactor. Serious consideration of a system driven by a subcritical driver is therefore appropriate.* The reactivity trace in a small, self-driven system is neither prototypic of the calculated accident sequence in a larger reactor nor controllable. Self-driven tests are therefore of limited utility. A converted thermal neutron spectrum driver may be a reasonable choice for the single assembly extended motion facility. In this case, burst width limitations cause the most serious concern, but burst width is one of the less important requirements of an extended motion facility.

* For tests not involving energetic bursts.

3. Gross Material Motion Involving More Than One Subassembly

The class of experiments in this area involves two generic aspects of CDA's. The first aspect entails communication of core materials (fuel, steel and sodium) between assemblies. The primary interest is to determine the impact that assembly wall failure has on overall dispersal of materials from the core and/or the growth of disruption within the core. The second aspect involves the manner in which the core material reaches a coolable geometry, along with the overall thermal energy to mechanical damage conversion process. Included in this class of tests are those which relate to ductless, or perforated duct, commercial plant designs.

Key Phenomena of Interest:

- a. Subassembly failure propagation - Under what conditions do assembly walls fail during accident situations such as flow blockage, flow coastdown, overpower transient, etc? Do these failures tend to propagate or are they self-limiting?
- b. Growing disrupted region - How much of the core eventually becomes disrupted (molten) in various power/flow mismatch situations? How does this disruption develop and what are the time scales for relevant events?
- c. Large scale fuel/steel - Sodium interactions - In what manner do bulk core materials and sodium mix? How does sodium eventually re-enter a hot voided core? What are the energetics and dynamics of these interactions? Are there scale effects?

Experiment Size Requirements:

The more subassemblies which are involved, the more difficult it becomes to make clean experimental observations of the phenomena under study. Even though disruption may eventually lead to whole core involve-

ment, we must compromise between our interest in observing the growth and behavior of such a disruption and our interest in understanding the mechanisms at play in the disruption. For this reason, we suggest limiting the size of this set of tests to seven full length subassemblies, in conjunction with some method for establishing the correct thermal environment outside the outer assemblies.

Less than full radius assemblies might be acceptable provided the reliability of calculational techniques is sufficient to extrapolate from experiment to the full size situation. It might be possible to perform these experiments with small assemblies in the same facility as used for the previous category of experiments. In the case of ductless experiments, it is thought that an experimental region of the size mentioned above would be sufficient to observe the dynamics of a growing molten region.

Data Acquisition Requirements.

a. Visual resolution can be on a rather coarse scale in these experiments since our interest is in the very gross behavior of the material. Hence, 10 mm radial resolution and 25 mm axial resolution is adequate.

Since many tests will involve azimuthal variations in assembly character, multidirectional viewing is needed. For example, some tests might involve off-axis control rod assemblies and assemblies of varying burnup. Ideal operation would involve two viewing directions in the horizontal plane. Observation along the entire length of the assembly is, again, mandatory. Observation of relative motion of fuel, steel and sodium is very desirable.

b. Thermophysical data should be known as well as in the single sub-assembly tests,

Energy input - within 1%

Temperatures - within 100 °K

Pressures - within 10^4 Pa (0.1 atm)

c. Temporal resolution - Only very mild prompt bursts are of interest (see below), and hence 10-20 msec resolution over a period of a minute or so is adequate, in most cases. For FSCI's, time scales may be shorter (msec) but the emphasis in these tests is on the macroscopic, global behavior of bulk core material. Current experiments and analysis tend to indicate that severe FSCI's do not occur, but if pressures of a few hundred atmospheres were to occur from mixing, material could accelerate rapidly to speeds of the order $5 \times 10^3 \frac{\text{cm}}{\text{sec}}$ or so.⁽⁴⁾ Hence, to resolve axial motion of a few cm's, msec time resolution is needed. A pressure trigger might be used to change framing rates dynamically during the test.

d. Heatflux out of the outer ring of assemblies - Information of this sort would help to predict incipient outward failure of the outer ring.

Test Frequency:

The same comments are applicable here as in the corresponding subsection on single subassembly tests. A large amount of what is needed to understand multiassembly tests can be learned from single assembly tests and calculations. Hence somewhat fewer tests might be envisioned to assess multiple assembly behavior. Certainly, at least 10 or so tests are in order.

Prototypicality Requirements:

Preconditioning is needed in order to build in high power (5% full power) fission products. Prototypic enrichment is not needed as long as

the power density history is programmable to match calculated whole core behavior. Some radial flux variation is desirable since rather strong power gradients across the seven assemblies may be of interest. The neutron energy spectrum which achieves the (nearly) correct thermal conditions within the test is needed. A fast spectrum may be required to achieve these conditions.

Total Energy Requirements:

The growth of a molten region takes place, by definition, rather slowly compared to prompt burst time scales. Assembly meltthrough is expected to occur in from 1-20 seconds after contact with molten fuel. Nonetheless, mild supercritical bursts cannot be precluded during the sequence of gradual core melting. Hence, we wish to provide capability for inducing prompt bursts of several dollars per second. Peak temperatures of 5000 °K are adequate, since larger temperatures are expected to develop thermal and pressure loads which dispel the core materials rapidly into the region above the assemblies.

The energy requirement analysis is quite similar to that for single assembly tests, with a peak temperature of 5000 °K. The most stringent test would be an unprotected partial flow blockage in a single assembly. Full power for a few minutes may be needed to allow voiding of the blocked assembly. This might be followed by a period of 20 seconds or so to lead to can wall failure. One might imagine* another minute or so for disruption to spread to surrounding assemblies. Hence, full power operation for around five minutes may be needed. Based on 200 w/gm full power operation and a peak temperature of 5000 °K

$$E = 200 (300) + 3150 = 9150 \text{ J/gm}$$

* Few, if any, detailed calculations of this progression exist.

Period Requirements:

While ramp rate simulations in these tests correspond to lower ramp rates than in the single subassembly tests (\$10/sec vs \$50/sec), the same considerations generally hold regarding facility period requirements. Material should not be allowed to move more than a specified amount under external forces during the burst.

Steady State Operation Requirements:

As mentioned under "energy requirements", tests at full power for five minutes can be envisioned. Considerable uncertainty exists whether even longer (10 min) tests would be desirable. For this reason we recommend design of a facility with essentially full steady state cooling capability. This capability would be compatible with the desire to precondition fuel. Since bursts need only be mild in these tests, driver core design becomes more feasible.

Comments:

These experiments comprise a subset of the class of tests alluded to as "Facility C" in the next chapter. Due to the limitation imposed by choosing a seven subassembly test size, a few important effects related to gross core behavior are not covered by these tests. If fluid-fluid explosive self mixing is relevant to dispersion energetics, then the geometry, quantity, and thermophysical condition of material mixed should be varied. This sort of experiment is best performed out-of-pile after one learns about the conditions of mixing from the in-pile tests. Integral tests, could serve as the best proof test for a whole core treatment based on modeling derived from the 7 assembly experiments. If instrumentation of the single subassembly tests provides thorough and reliable data, less precise data may be acceptable in the seven assembly tests. Few seven bundle tests would be needed if we had a high quality single assembly test program.

The requirement to precondition fuel is admittedly a stringent constraint on facility design. Nonetheless, simulation of decay heat in grossly disrupted geometries is extremely difficult to achieve with fission heating. There may be a possibility to dope high power fission products into the test fuel. The feasibility of this technique is doubtful, however, and a cursory study might rule it out.

Reactivity control requirements are extremely severe in seven assembly tests since so much reactivity is associated with the mobile fuel within the test. Performing the experiment with the power trace desired and without damaging a driver core is difficult. Since seven enriched and reflected assemblies can be made critical by themselves, one is tempted to study a driverless system controlled by fast acting rods in a reflector. An alternate approach might consider a disposable driver in which one would not rule out damage induced by the reactivity fluctuations of the test assemblies. Replacement cost and reactor down time would be of primary interest in this approach.

References

V. A. 2 and V. A. 3

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4. Coupled Hydrodynamics/Neutronics

The primary goal of coupled hydrodynamic/neutronic experiments is to examine the interplay between the various phenomena studied in the experimental areas outlined previously and the space-time behavior of the neutron flux in a full-scale reactor. Experiments of this sort may be complicated, difficult to instrument adequately, and the cost per experiment probably would be high but the reward per experiment may be large. Hence, only a few such experiments are appropriate, and heavy reliance must be placed on proven calculational methods along with a rather complete set of supporting experiments which lend confidence in our understanding of the relevant phenomena. These experiments are of the proof test type and their purpose is to provide a test for the calculational methods; one hopes that generalized conclusions about reactor behavior for related reactor types, or accident conditions, can then be deduced from the calculational technique.

Key Phenomena of Interest:

- a. Disassembly energetics - For a single, prompt critical neutronic burst, what are the reactivity quenching mechanisms and their magnitudes, how much fission energy is deposited in the core and what fraction of the energy is converted to system damage?
- b. Recriticality from dispersed geometries and autocatalysis - Is there evidence of recriticality or autocatalytic effects? What is the nature of these effects? What is the impact of noncoherence? How much fission energy is deposited in the burst(s)? How much of the energy is converted to damage?
- c. Final disposition of core material - How much of the core remains coolable in-place? What is the location and form of core debris following the excursion?

d. Post-shutdown material transport - How do fission products and other radioactive core materials move in the primary vessel? What fraction of the material produced escapes, and on what time scale? What is the physical form of escaping material?

Data Acquisition Requirements:

a. Visual resolution will be most difficult in these tests since the reactor core is optically thick. Techniques such as those used in KIWI-TNT^{1,2} might be valuable in the assessment of early stages of fuel motion. An excessive amount of in-core instrumentation may interfere with the disassembly process and produce nonprototypical results. Since the phenomena of interest might include recriticality and autocatalysis, gross observations on the motion of core material are needed. Resolution can therefore be quite coarse, but the entire core barrel region should be included. Techniques which key on the fuel alone (not necessarily sodium and steel) might be adequate, since fuel motion dominates the reactivity feedback and determines the flux shape. Observation of CDA bubble size and motion would also be valuable.

b. As much thermophysical information as practical is needed. Even if instrumentation survives only through the early stages of disassembly, this would be valuable information. Precise power levels, temperatures, and pressures are not needed, since the disassembly process is relatively⁴ insensitive to these parameters;*

Reactor Power - within $\pm 1\%$

Fuel Temperatures - within $\pm 100^\circ\text{K}$

* Nonetheless, a reasonable estimate of these variables is needed to relate to the analysis.³

Core Pressures - with $\pm 10\%$ of peak pressure. For a \$50/sec excursion which leads to $p_{\max} = 100$ atm an error of 10 atm around 100 atm is acceptable.

Core Material Velocities - For use in kinetic energy characterization of the damage potential. Knowledge of total kinetic energy to within 20% is desirable and to within 100% is acceptable, depending on the experiment.

c. Temporal resolution - A simulation of an initiation sequence may be required to establish the correct reactor core state prior to the burst, with sufficient instrumentation to define the conditions at the beginning of the burst. Data acquisition during a \$100/sec test might require around a second, while \$10/sec tests might require slightly longer due to the time necessary to reach prompt critical as well as the potential for more complicated and time consuming extended motion phenomena. The low ramp rate case might require instrumentation for up to 10 seconds. Since the total burst width of a \$100/sec excursion is around 3 msec, 0.1 msec time resolution is needed for burst observation.

d. Gas sampling - Continuous measurements on the amount and type of radioactive material released through head seals, etc., would be extremely valuable. Characterization of any materials released would allow relatively definitive experiments to be performed on the leakage rate out the secondary containment.

Test Frequency:

One thoroughly instrumented test would be better than none, as long as the test was not construed to be a demonstration of fast reactor safety in general. Several tests would, of course, allow us to span a broader range

of conditions of interest. The practicality of performing such tests depends on cost, and a trade-off between full prototypicality and cost is called for (see below). Tests of this sort probably should not be performed before a large class of phenomenological tests are understood and the reliance of calculational tools is high. A good example of the potential utility of a single test is seen in the KIWI-TNT and SNAPTRAN experience. Considerable credence was given to calculational tools, after a normalization point was established.

Prototypicality

The more prototypicality in reactor and vessel internals geometry and size the less uncertainty in overall test conclusions will result. The following conditions might be achievable without great expense.

1. Tests should be run in a sodium environment.
2. The core region should resemble the most likely commercial plant design.
3. An inexpensive simulation of the core support and radial blanket/reflector/shield structures should be included to provide the proper inertial constraints.
4. Simulation of any above core structure (such as instrument support structure, flow chimneys, etc.) that would influence the long term fuel motion should also be included.
5. The possibility (and effect) of scaling down the above-core sodium pool, vessel and head structure needs to be examined.
6. The head region could be designed to measure any sodium slug impact energy, as well as act as an energy absorption member. It could also include typical penetration designs to allow testing of durability under impact loadings, degree of sodium leakage and aerosol formation, etc.

7. Since a rigid and reuseable outer vessel might be desirable, the effect of vessel deformation could be simulated with a replaceable liner.
8. The system would need a simple control mechanism capable of reactivity insertions up to the 100 \$/sec range.
9. Programmable sodium flow conditions would be desirable, but may be prohibitively expensive for a destructive test machine. Initial voiding in the core region could be achieved mechanically, using rupture disks to expel sodium.
10. Preconditioning of the core is desirable but may be impractical since this implies full steady state capability. Pre-irradiation of the core in another facility might be possible, and would add some information regarding the impact of fission gas on accident dynamics and radioactivity release.

Energy and Period Requirements:

It is difficult to imagine a driver core which could survive the rigors of even scaled down versions of this class of tests. Hence, we conclude that a facility of this type would be self driven, and would exhibit the time constants and thermal behavior of a fast reactor. A control system might be needed to induce excursions up to the \$100/sec range.

Comments:

The value of integral tests cannot be overemphasized, provided the bulk of the STF program has given a firm basis for relying on overall accident analysis techniques. A few data points of this type would add considerable confidence that analysts had not "forgotten" any crucial phenomena. If instrumentation were adequate, integral tests could serve

as proof tests for the effectiveness of key phenomenological areas such as boilup, incoherence, etc.

Serious consideration should be given to the advantages of the Nevada Test site for integral tests. An existing experimental work force, experienced in hazardous experiments involving fissionable material greatly facilitates the ease of doing integral tests. Expenses related to environmental impact studies, etc., would be minimal at Nevada since most of the preliminary work has been performed as part of the weapons program.

References

V. A. 4

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5. Post Accident Heat Removal

The goal of PAHR experiments is to understand the long term cooling capability and disposition of accident generated core debris. This entails studies of heat transfer in internally heated mixtures of fuel and steel in an appropriate post accident environment. A good deal of information can be obtained for simulant materials in out-of-pile experiments involving electrical heating techniques. These techniques are more difficult to use when employing real reactor materials. An in-pile facility to provide the proper environment and heating mechanism for prototypical material combinations is therefore suggested. A reactor design primarily for the study of PAHR appears to be justified based on the need for reliable engineering data for use in the design of internal and external containment concepts.

Key Phenomena of Interest:

- a. Decay product disposition and behavior - How do fission products fractionate from fuel during and after a CDA? Does this depend on fuel type and nominal power rating?
- b. Fuel debris beds - How do these beds form and how are they characterized (fine powder or spongy debris)? Does this depend on fuel type? What percentage of steel is present? How and at what power levels and in what geometry does dry-out occur? Under what circumstances does self leveling occur?
- c. Molten fuel melt-out - How do freezing, plugging, and melting occur in below-core structures? At what rate will molten fuel above such blockages melt through?
- d. Molten pools - What are the heat transfer phenomena and rates? How does steel behave as it is melted at the periphery of a pool? Are solidified, insulating layers stable? What impact does boiling have on melt rates? What are radiation heat transfer rates?

e. Ex-vessel retention systems - What is the interaction of fuel-steel-sodium and sacrificial materials? What is the rate of pool growth? Are there interactions of the sacrificial material with concrete?

Data Acquisition Requirements:

a. Visual observations of the manner in which fuel melts downward through normal and sacrificial material would be useful. Observation of a melt front presumably requires a resolution of a few mm. This front may be difficult to resolve if the melting material has an optical density close to that of the fuel of interest. Rough observations of average density in a boiling pool might lead to information about vapor fractions and perhaps some qualitative feel for the boiling regime. If debris beds were allowed to form naturally, information on the debris character (e.g. moss, dust, particulate) would be helpful. Also, observations on the behavior (leveling?) of the bed would be useful. Since average fragment size appears to be around 200 μ in some tests, sufficient resolution may be impractical, and reliance on out-of-pile tests may be necessary.

b. It is reasonable to request reliable data on the thermal environment in PAHR tests. Temperature measurements in debris beds are straightforward. Measurements of molten pool temperature distributions are more difficult, due to the hostile environment within the pool. The temperature variation in the pool may be as high as 600 °C.⁽¹⁾ In GCFR tests, optical interferometry might be used to get pool surface temperatures at least. Measurements of directional heat fluxes will allow partial inference of what pool temperatures must have been, and of what the total fission product decay power level was.

c. Gas sampling - Quantitative, continuous data on the fractionation

of fission products is needed. Identification of the amount of various fission product species which are released would allow inference of the decay power which remains in the fuel.

Test Frequency:

Again, a large number of phenomenological out-of-pile tests will reduce the need for exhaustive in-pile tests. Nonetheless, a substantial number of in-pile tests are needed to establish prototypical conditions in the cases of interest. A few tens of tests are probably needed.

Prototypicality Requirements:

Preirradiation of fuel in other facilities or doping would be adequate to build in long-lived fission products. Preconditioning may be needed to build in short lived fission product species. If fission heating is being used to simulate decay heat the reactor should have a fast spectrum so that uniform, internal heat generation can be achieved. The test section should be able to accommodate a sample size up to around two feet on a side and around three liters of UO_2 . For the safety of the facility itself, low enrichment samples could be used and would be compatible with the fast spectrum.

Energy, Period and Steady State Operation Requirements:

Experiments requiring large rapid heating rates could probably be performed in another test reactor that may be used in studies of transient overpower accidents, etc. A steady state device capable of operation for up to a month at power is desired. If long-lived fission product isotopes are built into the test fuel in another facility, then operation of a test reactor is only required to restore the short-lived isotopes which have decayed during the transfer of fuel from the irradiation to the test facility. The test material might be rapidly removed from the core region

and allowed to generate its own internal heating. On the other hand, if fresh fuel were utilized in the tests, then steady facility operation would be used to simulate decay heat. Hence, the requirements for operation of the facility for up to one month periods involves experimental tradeoffs which should be more carefully studied. The power level which is acceptable depends on details of test/driver coupling and on the test enrichment utilized. Up to 10% nominal test fuel power is desirable. Advanced fuels may operate^{*} at power densities as high as around 400 watt/gm.² Hence, decay power levels of up to 40 watt/gm are desired in the test fuel.

The major design differences in GCFR's as compared to LMFBR's which could effect post-accident heat removal are (1) helium coolant, with down flow, (2) high pressure system, (3) fuel assemblies hung from top, (4) large thermal shields made of graphite surrounding the core, (5) core and primary system contained in pre-stressed concrete reactor vessel (PCRv) and (6) fuel elements loaded from the bottom.

This last difference may have one of the most important effects on the design of a PAHR system because there will be several entries penetrating the lower shield through which the fuel handling machine works.

Of the design differences above, the use of helium coolant changes the heat transfer problem considerably. There will be no natural convective heat transfer from core debris and molten layers, and dry-out problems are not considered. The molten fuel/coolant interaction does not occur so that there will be no particulate debris. Instead, thermal radiation becomes a significant heat transfer consideration, as well as possible interactions between molten steel and molten fuel.

From the discussion above it appears that three types of experiments are necessary. The first are basic in-pile heat transfer experiments in

*. While advanced fuels are theoretically capable of this performance level, some question remains as to the needed coolant volume fraction and concomitant doubling time implications.

which thermal radiation is an essential consideration. These should include experiments with radiative heat transfer between the molten debris and possible materials used for thermal shields such as graphite. The second are geometrical or design oriented, which are required because of the bottom entry for fuel handling equipment. The third are material compatibility problems. This includes molten fuel/graphite, molten fuel/concrete (for the PCRV), molten fuel/ SiO_2 (which may be used to line the PCRV) and other possible sacrificial material.

Many of the needed experiments will be similar to those discussed for LMFBR's with sodium absent. For example, molten pool experiments with molten steel above will be no different. Material compatibility experiments need only be expanded to include materials particular to GCFR's. It is not clear that the radiation heat transfer questions should be investigated in-pile.

Comments:

A fast flux facility with a large negative Doppler coefficient, for protection against possible transients induced by the experiment should be considered for PAHR experiments. This feature, along with low enrichment samples, would alleviate some safety consideration for the plant itself. Perhaps a high leakage (pancaked with an annular center of about 15 inch radius) core capable of driving test fuel to around 40 watts/gm would suffice.

A number of the data acquisition requirements discussed are desirable but not vital provided sufficient out-of-pile data are available. For example, careful measurements of temperature profiles in the sacrificial material might obviate the need for temperature measurements within the molten pool. A substantial portion of the fractionation experiments might be performed out-of-pile and non-radioactive isotopes of the relevant fission products could be studied in simulant fluids.

References

V. A. 5

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6. Radioactive Material Transport

The emphasis in this section is on an attempt to understand the nature and amount of radioactive material which might be released in a given CDA. Hence, there are two prime concerns. First, we wish to integrate results from structural dynamics tests and calculations with post-shutdown material transport within the primary containment. Second, we wish to understand the motion and nature of any radioactive material which has escaped the primary containment. These phenomena must be understood for a wide variety of accident initiators and in a broad class of commercial fast reactor designs. Information in this area, once again, will arise from a mixture of in-pile and out-of-pile tests.

Key Phenomena of Interest:

- a. Formation and nature of vapors and gases produced during a CDA - When are gaseous or volatile fission products released during the accident? How much condenses? How much sodium vapor is produced at the end of the CDA and what fraction of the core materials and fission products are entrained?
- b. Motion of accident effluents within the primary containment and out of the containment. What fraction and species of radioactive material produced in the CDA reach a leakage area within the containment? What time dependent leakage ensues, on both short and long time scales? How much sodium is ejected and in what form (bulk, spray, etc.)? Note that very slow leakages are included in which containment is not breached, but leakage through head penetrations and seals may ensue.
- c. Integrity of intermediate containment systems - Do sealed head cavity areas remain intact? What degree of retention ensues? Do guard vessel structures remain intact? How effective are PAHR systems? Can further aerosols be produced?

d. Motion of accident effluents within and out of the secondary containment - What role do sodium fires and/or sodium-concrete reactions play? How do aerosols coagulate and fallout or condense within the secondary building? How effective is the secondary containment; how well do filters and airlocks function?

The emphasis, in the following discussion, is on vapor formation, behavior and escape of released materials from the primary containment. Experiments on the behavior of vapors within the secondary containment have been performed in the past, and facility requirements are better defined than in most cases in this report. Indeed, adequate facilities may currently exist for this purpose and the experimental need may reduce to a more careful application of these facilities along with improved initial conditions and analytical models.

The tests performed in this area should stress a phenomenological approach. This phenomenological data should then be used in conjunction with data obtained in integral tests to serve as verification of a given analytical model.

Data Acquisition Requirements:

a. Visual observation of the rate of formation and size of a "CDA bubble"* is of interest. The subsequent growth and motion of this bubble within the primary containment is also an intrinsic part of the analysis and should be investigated experimentally. Since the amount of vapor that actually escapes the core barrel is not well known for a given accident sequence, experimental information is needed. Visual observation of early bubble growth might allow some

* By this, we mean the bubble(s) of radioactive vapors and gases which result from the given CDA.

inference of the amount of vapor produced. Later measurements will establish the amount of vapor that actually reaches the reactor head region. These two measurements will serve to help tie down calculational methods.

A natural vehicle to observe CDA bubble behavior might be an ancillary part of single or seven subassembly experiments. Scale effects are expected to play a role since condensation effects may depend on the surface/volume ratio of the bubble or bubbles. Hence, experiments smaller than single subassembly size are of limited utility. In fairly energetic tests, bubbles of the scale 10 cm to 50 cm in diameter may be created. In less energetic tests, lower vapor production rates might cause smaller bubbles to form, of the order 1 cm in diameter. Hence, it appears that a resolution of the order 5 mm is adequate. Since the density variation between the liquid and vapor of sodium is around 10^3 , x-radiographic techniques would appear to be suitable.

b. Thermophysical data - Temperature and pressure probes within the CDA bubble may upset bubble dynamics. More reliable information might be obtained by making pressure, temperature and mass flowrate measurements at the top end, inside the test assembly(s). Desirable tolerances for this type of experiment are:

Temperature : within ± 50 °K

Pressure : within $\pm 10^5$ pa (± 1.0 atm)

Flowrate : within $\pm 10\%$ maximum rate

If and when radioactive material escapes the simulated primary containment, measurement of radioactive species size and activity distributions is needed.

c. Temporal resolution - The time scale of interest in CDA bubble motion is the order of a few seconds. Bubble formation, however, may occur more rapidly (order 0.1 sec). Time resolution of 10 msec is therefore adequate. A fuel-steel coolant interaction may occur on a more rapid time scale, but the details of that interaction are of secondary interest here. The primary goal is to observe and understand the longtime behavior of material transport over large distances.

d. Gas sampling techniques which measure the amount and type of radioactive material which leaks or is emitted through simulated reactor head structure is needed. A wide class of situations might be tested, including cases in which seals remain largely intact and cases in which the seals are breached to varying extents.

Test Frequency:

Presuming that radioactive transport experiments are run in conjunction with single subassembly tests, a few tens of tests could be performed and would be adequate.

Prototypicality Requirements:

Establishing the proper two phase mixture of reactor materials at the assembly outlet region is important in CDA bubble experiments. Out-of-pile controlled simulations of various vapor mixtures should be performed to gain confidence in analytical modeling. In-pile tests, however, should simulate the natural formation and transport of accident vapors. Preconditioning of test fuel probably is not needed, but the presence of prototypical amounts of fission product species is needed. These need not necessarily be the radioactive isotopes, however, and hence preirradiation is optional. Establishing adequate fission product isotopic distributions in an artificial manner may be impractical, however.

The neutron spectrum used to establish vapor production is not expected to be important. The initial thermal conditions within the assembly should be nearly correct so that vapor production, motion, condensation, etc. are prototypical. Full size assemblies may not be necessary, provided condensation mechanisms are well understood.

Total Energy and Period Requirements:

As far as CDA bubble behavior is concerned, the energy and period requirements specified for single subassembly tests are sufficient. However, they may not be necessary in most cases. For example, 4100 J/gm is sufficient to bring fuel from 720 °K to 6000 °K,* provided a steady state power level is not needed to establish voiding profiles in a LOF sequence. Furthermore, a burst width of 50 msec would certainly accomplish this. Since CDA bubble studies seem to have requirements which are a subset of the single subassembly tests, it is tempting to consider performing these tests in the same experimental vehicle.

Comments:

A good understanding of the mechanisms by means of which radioactive material could escape a primary system subsequent to a DBA does not exist. Therefore, definition of a viable experimental program in this area is currently difficult. Nonetheless, some indication of the needs in this area are alluded to above, as to STF requirements. Many studies which do not determine the STF requirements are called for and a more thorough experimental program definition is needed. The potential payoff in terms of reducing accident risk uncertainty is so large (many orders of magnitude) in this area that much more experimental and analytical emphasis is called for.

* Assuming total vaporization, which is conservative.

B. Facility Characteristics.

One can identify four separate types of facilities to satisfy the broad range of experimental requirements specified in this section. It may be reasonable to combine some of these four types of facilities into two or three facilities. Specification of the characteristics of the four systems will allow us to make better judgements regarding the proper mix of facilities. Recommendation of facility mixes to satisfy identified experimental needs will eventually be based on the facility characteristics discussed below. The facility types are defined in Table I in terms of the kinds of experiments to be performed. The utility of each of the four facility types discussed depends on cost/benefit evaluations which remain to be performed.

TABLE I

CANDIDATE FACILITY TYPES

Facility A - Few pin experiments related to pin failure and initial cladding and fuel motion.

Facility B - Experiments related to gross fuel/steel motion within a subassembly and to a limited class of radioactive material transport phenomena.

Facility C - Experiments involving material motion in more than one subassembly as well as those involving coupled hydrodynamic/neutronic phenomena, and radioactive material transport.

Facility D - PAHR-related experiments involving large masses of fuel and materials with relatively low internal heat generation.

Designation of three categories of characteristics for each facility type will be discussed. These are defined as:

Desirable - The ideal conditions which the analyst seeks. These conditions reflect (to some degree) the situation which is believed to exist in a real LMFBR under a variety of CDA situations. If the conditions denoted "desirable" were achieved in the STF, then the analyst would express a high degree of confidence in analytical techniques which match the experimental results. Furthermore, a high degree of reliance could then be placed on these techniques in computations involving a broader range of accident situations than those actually studied experimentally.

Acceptable - These conditions might reflect more realistic design targets. Analytical techniques that were found to match the experimental results could be extrapolated to reactor conditions with an acceptable level of confidence.

Necessary - The facility must have these characteristics as a minimum to be a worthwhile candidate. Experiments or facilities which are at this minimum level would not necessarily allow direct extrapolation to reactor conditions but are nevertheless judged to be worthwhile. Additional sources of information or data would be required to allow extrapolation to reactor conditions.

The following comments are primarily based on subjective evaluation. Considerable analytical work and engineering judgement are needed before these facility characteristics should be used as design requirements. We set forth these comments more for guidance and as a departure point for further discussion than as rigid guidelines.

1. Facility A

(1) Experiment Size:

Desirable: Up to 37 full-length pins. This should include allowance for up to 1.5 m (5 ft) active core length for advanced designs.

Acceptable: Up to 19 full-length pins. Active core length should be keyed to near-term commercial LMFBRs.

Necessary: Up to 19 full-length pins, although many experiment could be performed with a single pin.

Active core height of .9 m (3 ft) would be acceptable if this were a severe cost or design constraint.

(2) Steady-State Operation:

(a) Pre-burst: (Assuming preirradiation in separate facility).

Desirable: Two weeks of full power operation. (sufficient time to restructure fuel, heal cracks, etc.).

Acceptable: One week for fuel conditioning. (Around one minute is needed for accident sequences such as extended undercooling or very slow TOP's).

Necessary: One hour (or whatever time would allow significant crack healing in columnar-grain fuel).

(b) Post-burst: None is necessary. (Extended fuel motion is not an objective provided other facilities are available to investigate extended motion).

(3) Burst Characteristics:

(a) Energy Deposition (peak axial location)

Desirable: We wish to reach peak fuel temperatures of 6000 K. An energy deposition (peak axial location) of 3700 J/gm^{*} is desirable (this allows complete vaporization of fuel) during a burst. For slow transients, an energy deposition of 6000 J/gm above steady-state is desirable, but with a cooled driver core this could be the lesser requirement.

Acceptable: Peak fuel temperatures of about 5500 K would be acceptable. An energy deposition during a burst of 2500 J/gm would allow for reaching this temperature including some fuel vaporization. For slow transients, an energy deposition of 5000 J/gm would probably be adequate.

Necessary: If slow transient overpower (50¢/sec-\$3/sec) and EOS-type experiments are not included then energy depositions of about 1200 J/gm would be acceptable for many fuel failure and initial motion experiments.

* Assuming preconditioning establishes an initial peak temperature of 2000 °K.

(b) Burst-Width:

Desirable: A burst width (FWHM) of one msec is desirable.

Acceptable: Five msec is acceptable.

Necessary: If rapid transient overpower and EOS-type are not included, or if time scale is shown to be relatively unimportant then burst widths of about 100 msec would be adequate.

(4) Power Distributions:

(a) Axial Distributions:

Desirable: Capability of achieving prototypic axial distributions within about 2% variation is desired. This should be achievable with from .9-1.5 m (3-5 ft) active test fuel lengths.

Acceptable: Capability of achieving prototypic axial distribution within about 5% variation is acceptable for active test fuel lengths of .9-1.2 m (3-4 ft).

Necessary: Capability of achieving prototypic axial distribution within at worst 10% variation is necessary for active test fuel lengths of .9 m (3 ft).

(b) Radial Distributions:

Desirable: Variations in fuel temperature distributions from pin to pin without enrichment grading should be less than 20°K for experiments in which flat distributions across the test assembly are desired. Distributions should

also be within 20°K of that calculated for a pin in the reactor case. There may be some experiments in which higher temperatures in a central pin(s) is desired. It is unlikely that experiments considered here will attempt to simulate edge or power gradient effects. Average temperature of coolant associated with each pin should not vary radially more than 10°K across the test assembly. Cladding temperatures should not vary more than 10°K from pin-to-pin across the test assembly.

Acceptable: Variations of 50°K in fuel temperatures and 20°K in coolant and cladding temperatures in the same sense as above and without enrichment grading would be acceptable.

Necessary: Variations of no worse than 100°K in fuel and 30°K in coolant and cladding temperatures are necessary. Enrichment grading is permissible if necessary.

(5) Test Fuel Enrichment:

Desirable: Use of enrichments prototypic of commercial LMFBRs (as low as 10-12%) would be desirable.

Acceptable: Use of high-burnup FFTF inner-zone irradiated driver fuel ($\sim 20\%$ enrichment for 10% burnup fuel) would be acceptable for failure threshold tests. Use of outer zone irradiated

fuel ($\sim 25\%$ enrichment for $\sim 10\%$ burnup fuel) would be acceptable for other tests.

Necessary: Use of high-burnup FFTF outer zone fuel ($\sim 25\%$ enrichment) would be necessary for failure threshold tests. Higher enrichments could be used for other tests as necessary.

(6) Frequency and Number of Tests:

Desirable: Twenty-Five tests/year would be a desirable level of capability for this facility (depending on pre-conditioning needs). We would expect that a total of several hundred experiments would be performed in the facility. An individual test facility core life of two years, depending on cost, is desirable with 25 tests per year at an average of one week/experiment steady-state operation.

Acceptable: A level of about 15 tests per year would be an acceptable capability for the facility. Of the order of one-two hundred experiments might be an acceptable total number of tests throughout the facility life. A core life of two years (15 tests assuming average steady-state operation of one week/experiment) is acceptable.

Necessary: Capability to perform at least 10 tests/year is necessary. A core life of at least one year (pre-conditioning could be negligible) is necessary.

(7) Visual Data Acquisition:

Desirable: A visual resolution (η, γ - hodoscope, x-radiography techniques) of .5 mm radially, 2 mm axially in a 19-37 pin array is desirable. A time resolution (during rapid transients) of .5 msec is desirable. At least two viewing directions are desired. An axial extent of observation of 1.5 m is desirable. A feature of allowing focusing down to a single-pin level with perhaps .1 mm resolution would be desirable. Observation of relative motion of fuel, steel, and sodium is desirable.

Acceptable: A resolution of 1 mm radially and 5 mm axially over an axial extent of 1.2 m (4 ft) would be acceptable. A time resolution of 5 msec (during rapid transients) would be acceptable. Two viewing planes are required. Observation of relative motion of fuel and steel, but not sodium, is acceptable.

Necessary: A resolution of at least 5 mm radially and 25 mm axially over an axial extent of at least .5 m is necessary. One viewing plane is the minimum acceptable, i.e., some form of observation of at least fuel motion is mandatory.

The characteristics of interest in Facility A summarized in the following table.

	<u>FACILITY A</u>		
	DESIRABLE	ACCEPTABLE	NECESSARY
No. pins	37	19	19
Pre-conditioning	2 wks	1 wk	1 hour
Steady-state (experiment)	1 min	1 min	1 min
Post-burst steady-state	--	--	--
Burst energy deposition	3700 J/gm	2500 J/gm	1200 J/gm
Slow transient deposition (above steady-state)	6000 J/gm	5000 J/gm	--
Burst width (minimum)	1 msec	5 msec	100 msec (no EOS exp)
Radial power distribution (temperature) variation	20° K	50° K	100° K
Fuel enrichment (test fuel)	≥10-12%	≥ 20% (failure threshold) ≥ 25% (others)	≥25%
Visual data resolution			
radial	.5 mm	1 mm	5 mm
axial	2. mm	5 mm	25 mm
Axial extent	1.5 m	1.2 m	.5 m

We wish to stress that conditions specified under the "acceptable" category are subject to revisions reflecting design constraints and costs. Tradeoffs in prototypicality can be made since we must realize that not all situations of interest to fast reactor safety can be mocked-up in the test facility. For example, if instantaneous thermal conditions can be shown to play the major role in failure threshold and position, then excessive concern

over preconditioning may be unwarranted. Nonetheless, we wish to remove as many uncertainties as possible from the analytical experimental symbiosis. It is for this reason that we stress the importance of phenomenological tests which will put analytical techniques on a firm footing.

2. Facility B

(1) Experiment Size

Desirable: A facility capable of accepting a test fuel array of .23 m (about 9 in) across would be desirable. This would allow experiments with one full subassembly of 217 pins of 9 mm diameter* plus two rows of pins outside the subassembly or a full subassembly of 331 pins of 7 mm diameter** plus three rows of pins outside the subassembly. This would also allow experiments with seven subassemblies of 91 FFTF pins each.

A facility capable of accepting full-length subassemblies (up to 4.5 m) would be desirable. An active core length of 1.5 m is desirable.

Acceptable: A facility capable of accepting experiments with one full FFTF subassembly plus surrounding two pin rows would be acceptable. This would give a fuel test array size of about .15 m (6 in) across outer flats. Full-length subassemblies (3.6 m or 12 ft) of FFTF dimensions with an active core length of 1.2 m should be allowed.

Necessary: A facility allowing experiments with 37 FFTF

* This corresponds to the large diameter advanced fuel pin.

** This corresponds to expected typical diameter, large LMFBR oxide fuel pins.

dimension pins would be useful for some experiments in the gross motion category. For experiments with low power density (boilup, reentry) at least 127 pins would be necessary because of heat loss effects.

(2) Steady-State Operation

(a) Pre-burst:

Desirable: Preconditioning for a few hours prior to an experiment to build in fission products would be desirable. Large quantities of gaseous fission products could be built-in through pre-irradiation in another facility. Preconditioning is desirable in order to produce short lived fission products for experiments involving decay heating.

Acceptable: Preconditioning to build in fission products for ten minutes would be acceptable. One minute of equivalent full power steady-state operation would be sufficient for performing the experiments.

Necessary: Equivalent full power steady-state operation for at least one minute would be necessary. This would not allow sufficient fission product buildup but would allow some useful experiments to be performed. Simulation of decay heat through fission heating could be used, but this would require at least a 1.5 m neutron deposition length.

(b) Post-burst:

Desirable: If fission heating is used to simulate decay heating, then several minutes of low-power operation would be desirable.

Acceptable: With fission heating used for decay heat simulation, one minute of low-power operation would be acceptable.

Necessary: Same as above.

(3) Burst Characteristics:

(a) Energy Deposition (peak axial location)

Desirable: Assuming that steady-state operation for at least several minutes is available prior to a burst then an energy deposition above steady-state of 6000 J/gm would be desirable. It is assumed that experiments in the gross motion category would not necessarily have an extensive pre-burst transient above steady-state. A level of 6000 J/gm would allow a transient of five full-power (200 w/gm) seconds above steady-state prior to a burst of 4000 J/gm followed by a transient or high steady power level for an equivalent five full-power seconds.

Acceptable: An energy deposition above steady-state of 5000 J/gm would be acceptable.

Necessary: An energy deposition of at least 4000 J/gm is necessary. Again, this figure is conservative, since it allows for complete vaporization of the fuel.

(b) Burst-width:

Desirable: A burst-width of 5-10 msec would be desirable for some experiments in this facility.

Acceptable: A burst width of 50 msec would be adequate for most cases of interest.

Necessary: A burst-width of no more than 100 msec is necessary.

(4) Power Distributions:

(a) Axial Distributions:

Desirable: Axial flux shapes within 5% of prototypic for each experiment are desirable. Matching reactor power distributions is not as important as an accurate knowledge of the distributions actually achieved in the experiment. In distorted core situations, the power shapes tend to follow fuel motion and can be strongly different from nominal shapes. It is desired in some experiments to simulate changing power distributions. An on-line, fast response (order of tens of msec), feedback-controlled, axial power shaping system for the test region would be useful and desirable if at all possible.

Acceptable: Axial flux shapes within 10% of prototypic for each experiment is acceptable.

Necessary: Axial flux shapes within no worse than 20% of prototypic for each experiment is necessary.

(b) Radial Distributions:

Desirable: For experiments in which a flat radial profile is attempted, a variation of less than 2% in pin power across the test array is desirable. For some experiments, the capability to provide up to 50% linear variation in pin power across the array is desirable. Enrichment grading is not desirable to achieve these goals.

Acceptable: A variation of 5% across the test array is acceptable. Some flux peaking is permissible since the turbulence in most experiments will tend to flatten temperature distributions. Excessive graded enrichment is unacceptable.

A capability to provide up to 30% linear variation in pin power across the test array is acceptable.

Necessary: A variation of no more than 15% across the test array is necessary.

(5) Test Fuel Enrichment:

Whatever is available. Fuel enrichment variations from typical are not expected to play an uncertain or strong role in extended fuel motion experiments. However, limited availability of high-enrichment preirradiated pins may require use of FFTF driver fuel for some large scale experiments

(6) Frequency and Number of Tests:

Desirable: Ten tests/year and total of 100 is desirable.

Acceptable: Ten tests/year and total of 50 is acceptable.

Necessary: At least four tests/year and total of 20 is necessary

(7) Visual Data Acquisition:*

Desirable: A resolution of 5 mm radially and 25 mm axially over a 4 - 4.5 m axial span is desired. A time resolution of 5 msec during bursts is desired.

Acceptable: A resolution of 10 mm radial and 50 mm axially over a 4 - 4.5 m axial span is acceptable. A time resolution of 10 msec during bursts is acceptable.

Necessary: A resolution of at least 25 mm (1 in) radially and 100 mm axially over a 2 m axial span is necessary. A time resolution of at least 20 msec is acceptable.

The characteristics of interest in Facility B are summarized in the following Table.

FACILITY B

	DESIRABLE	ACCEPTABLE	NECESSARY
No. pins	331 + 2 rows	217 + 2 rows	37 (some exp) 127 (low-power exp)
Pre-conditioning	3-4 hours	10 min	--
Steady-state (experiment)	1 min	1 min	1 min
Post-burst steady-state (low power)	5 min	1 min	1 min
Burst energy deposition	6000 J/gm	5000 J/gm	4000 J/gm
Burst width	5-10 msec	50 msec	100 msec
Radial power distribution (variation from prototypic)	2%	5%	15%
Fuel enrichment (test fuel)	--	--	--
Visual data resolution			
radial	5 mm	10 mm	25 mm
axial	25 mm	50 mm	100 mm
Axial extent of view	4.5 m	4.5 m	2 m

* Measurement of reactivity variations due to test fuel motion could complement visual observations through the use of neutron kinetics codes.

3. Facility C

A wide range of important phenomena is associated with the large scale motion of core material. However, experiments in this area may be more difficult than small scale phenomenological tests for two reasons. First, experiments involving large quantities of mobile fuel are potentially uncontrollable (self driven) and may lead to considerable disruption of a re-usable test vehicle. This disruption could lead to costly clean-up procedures and hence to costly experiments. Second, the gathering of useful data is more difficult in large scale experiments. Obtaining and interpreting visual data is substantially more difficult than in small scale experiments. Deduction of material motion from thermophysical data becomes increasingly difficult as the experiment geometry increases in complexity. Nonetheless, the cost/benefit ratio for a variety of large scale tests remains uncertain and it would be premature to exclude these experiments from consideration at this time. The following types of experiments are assumed to fall within this category:

Subassembly failure and propagation

A growing disrupted region in a CDA

Large scale FSCI's

Fuel reentry mechanics; incoherence

Neutronic feedback and coupling - recriticality and auto catalysis

Fission energy to damage conversion efficiency due to a specified prompt critical burst

Radioactive material transport

Clearly, a large number of experiments would be needed in each category to really gain a thorough understanding of each of the phenomena

* These were discussed in more detail in Section A of this chapter under:

°Gross material motion involving more than one subassembly

°Coupled hydrodynamics/neutronics

°Radioactive Material Transport.

over the span of accident situations of interest. However, a modest experimental program, with modest goals might prove to be relatively inexpensive and to provide valuable, albeit limited, information. The cost/benefit ratio might be enhanced by a reasonable compromise of the objectives of the experimental program. Reliable conclusions regarding the utility of experimentally addressing large scale core motion experiments must await further study of the iterative tradeoffs involved.

The following facility characteristics are set forth as a departure point for that study. We currently include facilities ranging from those involving small scale 7 subassembly arrays to large scale, whole core integral arrays. The facility characteristics are therefore less specific than in the other cases.

(1) Experiment Size

Desirable: Ranging from 7 full size subassemblies to a complete core (~ 100 subassemblies).

Acceptable: 7 subassemblies of 127 pins each, with 2 rows of pins surrounding the array.

Necessary: 7 small subassemblies of 37 pins each with 2 rows of pins surrounding the array. This arrangement could be made to fit into facility B.

(2) Steady-state Operation

Characteristics are the same as for facility B except in the case of certain integral tests for the purpose of studying the energetics of a given prompt burst. In that case, no steady state operation is required, since a single, controlled prompt critical burst is the primary phenomena under study.

(3) Burst Characteristics:

(a) Energy Deposition (peak axial location)

Desirable: The magnitude depends on the amount of fuel involved in the test. Excessive energy deposition might give rise to unacceptable containment costs. If the facility is not reusable, then some damage to the containment is acceptable. It is desirable to design for the largest energy deposition possible within these constraints.

Acceptable: 4000 J/gm is adequate and could provide a substantial burst.

Necessary: 2000 J/gm is the minimum interesting deposition since this corresponds to fuel melting and would allow failure propagation and some FSCI experiments.

(b) Burst Width:

Desirable: For integral tests, simulation of a \$100/sec burst would be desirable. This corresponds to a burst width of around 1 msec (FWHM). Multiple subassembly tests, if self driven, might achieve a similar burst width.

Acceptable: An interesting class of experiments could be performed employing a driver core capable of 50 msec bursts.

Necessary: A very broad burst (.5 sec) is adequate if tests are limited to failure propagation

studies. In that case, we desire some means of fairly rapid and extensive fuel melting. Failure propagation tests involving basically steady power will be performed in SLSF. The need for a failure propagation facility which utilizes a large test region, performs rapid melting and which induces a mild burst is questionable, but is the basis for the "necessary" characteristics mentioned here.

(4) Power Distributions:

The flux will follow the fuel motion in tests involving a large quantity of fuel. If a driver core is used, knowledge of the flux shape to within 5% axially and radially is desirable. In 7 subassembly tests one may wish to establish a large radial power gradient across the array. In all of these cases, prototypicality is not as essential as an accurate knowledge of the power distributions.

(5) Test Fuel Enrichment

One may wish to use highly enriched fuel to minimize the critical mass of self driven systems and to thereby reduce costs and enhance the safety of the experiment. For 7 subassembly tests in a driver core, lower enrichment pins would be desirable in order to reduce the neutronic feedback to the driver. Since the energy balance and gross material behavior are the key interests in these tests, prototypical enrichment is not necessary.

(6) Frequency and Number of Tests

Desirable: As many as 5 tests/year and total of 25 tests is desirable

Acceptable: As few as 1 test per year and a total of 5 test is acceptable.

Necessary: A single well chosen test would be better than none.

(7) Visual Data Acquisition

For whole core integral tests, the optical thickness of the experiment almost precludes observation. One would probably have to rely on thermophysical measurements. Even 7 full subassembly tests probably fall into this category. For 7 small subassemblies, resolution equal to that for Facility B is appropriate.

Comment:

No table of facility characteristics for Facility C can be prepared at this time. Further restriction and definition of the experimental objectives is needed before the relative merits of various facility characteristics can be assessed. Further definition will be found in a later report, where particular attention will be paid to assessing the utility of integral excursion tests. The complexities introduced by scale effects, uncertain neutronic feedbacks, and difficulties discourages us from making strong recommendations for Facility C type experiments unless they can be designed specifically to examine specific phenomena.

4. Facility D

(1) Experiment Size *

Desirable: A facility capable of accepting an experimental apparatus about .6 m (24 in) across would be desirable. This would allow proof-in-principle testing of PAHR systems such as core catchers. All remaining experiments anticipated have a lesser space requirement.

* Note: size is experiment size, not cavity size which must be larger for instrumentation.

Acceptable: A facility which allows an experiment of .4 m (16 in) across would be able to handle all suggested experiments without modification except proof testing. Some proof testing, however, will be possible in a limited way.

Necessary: A smaller experiment of .23 m (9 in) in size will yield useful information on all aspects except design proof testing.

(2) Steady-State Operation

The following table defines the power requirements and test duration characteristics.

Experiment	Power Requirements (percentage of full power density)			Test Duration (hours)		
	Desirable	Acceptable	Necessary	Desirable	Acceptable	Necessary
Decay Product Disposition	0→100	15	5	10	6	2
Debris Beds	10	7	5	2	1	0.5
Molten Fuel Melt-out	10	7	5	10	7	5
Molten Pools	5	3	2	3*	2*	1*
Ex-vessel Devices	5			7 days	2 days	1 day
Summary	0→100	15	5	7 days	2 days	1 day

* Long time required to reach steady state; these numbers depend on test fuel power level capability.

(3) Instrumentation:

Desirable: Multiple measurements of temperatures, (4) pressures and (5) heat fluxes as well as (6) regression rates of bounding surfaces during melting. Gas analyzer for decay product disposition experiments.

Acceptable: The number of transducers can be reduced by half but a gas analyzer is needed. A hodoscope or other system to monitor surface motion only would be acceptable.

Necessary: For decay product disposition: Gas analyzer, temperature, pressure

For fuel debris beds: Temperature, pressure, visual observation of debris surface

For molten fuel meltout: Temperature, pressure

For molten pools: With repeated tests, could be done with almost no instrumentation. Perhaps only 2 temperatures, 1 pressure, gas analyzer, hodoscope or other device for monitoring surface motion.

Facility D may well be the lowest cost/benefit facility of the four mentioned types. Its routine operation could be made to be relatively safe and it is fairly certain that adequate experimental measurements could be made to allow decisions to be made regarding PAHR related safety analyses. A further utility can be envisioned as a design proof-test facility. Fission heating is adequate (desirable) in most cases since generic phenomena can be observed for known heating rates. Fission product fractionation can be assessed in other facilities.

VI. EXISTING FACILITIES

In this section the capabilities and limitations of existing in-pile facilities will be briefly reviewed and related to the experimental needs discussed earlier. In addition, several large electrically heated out-of-pile facilities may be noted. These include the Fuel Failure Mockup facility (FFM)⁽¹⁾ an electrically heated mockup of 19 FFTF pins in which the effects of partial blockage of sodium channels have been studied, and the OPERA facility at ANL, in which sodium expulsion has been studied during flow coastdown simulation in a seven pin test.⁽²⁾ Such facilities are of great value in understanding sodium boiling and expulsion, clad overheating, and perhaps the early stages of clad relocation. In the GGFR program a similar large Core Flow Test Loop facility (CFTL) is planned, at ORNL, to be capable of heating up to 91 pins.

A. Facilities

1. TREAT

The Transient Reactor Test facility (TREAT) is located at the Idaho National Engineering Laboratory (INEL) and has a thermal, air-cooled, driver which can be used in a pulsed adiabatic mode for imposing destructive fission-heated transients on full length (3') fast reactor fuel pins or other fissile material located in a central axial test region. The driver fuel is a graphite-UO₂ mixture having a C/U²³⁵ ratio of 10⁴ so that the neutron spectrum is predominantly thermal. A burst can have a minimum period of around 20 msec and a minimum width of around 100 msec. Prior to a burst, TREAT is frequently operated at more or less constant power for a few seconds so as to produce prototypic power densities in the fuel and to establish reasonable thermal profiles in the fuel. TREAT is equipped with a fast neutron hodoscope which is capable of detecting fuel motion (see Section VIII). In addition, the test capsules are usually instrumented with thermocouples, pressure transducers, and sodium flow meters.

TREAT has been used for many valuable experiments on sodium expulsion, clad relocation, fuel-coolant interaction, and fuel motion during various overheating and/or undercooling transients.⁽³⁾ It is expected that TREAT will continue to be an essential facility for conducting transient tests of fuel behavior for many years.

However TREAT has a number of limitations; the most serious are connected with the thermal neutron spectrum incident on the test region. Of course, the thermal neutron spectrum can be eliminated by use of neutron filters between the test and driver regions but only at the expense of a marked reduction in the energy deposited in the fuel. Therefore TREAT is usually run with an important thermal flux component in the test region. Because of the short mean free path for thermal neutrons in the fuel there will be, for a single fuel pin, a substantially higher power density at the edge of the fuel pin than at the center and, for a cluster of fuel pins a higher power density in the outer pins than in the inner ones. For a single fuel pin, this surface power peaking leads to radial distributions of temperature, thermal stress, melting, and fission gas release which are different from those which would be obtained in a fast reactor. Therefore, elaborate and perhaps uncertain calculations may be required in order to extrapolate from the pin failure thresholds and mechanisms observed in TREAT to those in a fast reactor.

The radial power distribution in a cluster of pins may be improved by using pins of lower enrichment in the outer ring but this still leaves a power gradient across the outer pins which in turn will lead to non-prototypic temperature distributions in such pins. With such a complicated temperature distribution, it is difficult if not impossible to study phenomena associated with coherence and propagation of pin failure during a transient.

A number of seven-pin tests have been run in TREAT. Tests with larger numbers of pins would not be expected to be useful because of the complicated power and temperature distributions in most of the pins. However in tests with

few pins, say seven or less, the test environment of the pins may be a dominant feature, at least during later stages of a transient. For example, all or nearly all of the pins are usually adjacent to steel which defines the sodium flow channel in the test region. Interactions of disrupted fuel, clad, and/or sodium vapor with such cool surfaces may be a dominant effect during a transient test and an effect which would not be found near most of the fuel pins in a fast reactor.

In addition, there are classes of experiments which cannot be performed in TREAT because of its relatively long pulse width or because of its inability to operate at significant steady power. Further limitations have been imposed on many of the past TREAT tests by lack of available test fuel, for example full length irradiated fuel pins, or by test loop features and do not represent limitations of TREAT itself.

2. SLSF (ETR)

The Sodium Loop Safety Facility (SLSF) is being installed in the Engineering Test Reactor (ETR) at INEL. ETR is a light-water moderated, heterogeneous, thermal test reactor which began operation in 1957 with a power of 175 MW. The active core height is three feet. The reactor is currently shut down during installation of the SLSF and it is planned that starting in Sept. 1975, the reactor will be devoted primarily to fast reactor safety tests using the SLSF, at the rate of around three tests per year.

For each test, the ETR will be used to drive fuel pins in a throw-away test module, which will include, in addition to the test fuel, a self-contained sodium loop with pump, a sodium-helium heat exchanger, and appropriate containment and neutron filters. Heat will be removed from the module by helium coolant and the helium system is limited to around 1.5 MW. Initial tests will use 19 pins and will be limited to 37 pins in the present system.

The ETR has no burst capability, but it can be used to irradiate enriched fuel at prototypic power densities or higher, for periods of days to weeks. With fully enriched fuel, power densities around twice prototypic are expected to be achievable. It is thus possible to precondition enriched fuel, to simulate LOF conditions, and perhaps to obtain melted fuel with flowing sodium still adjacent as in a TOP condition.

Since the ETR is a thermal reactor, there are problems in attaining a reasonably flat power distribution in a cluster of fuel pins. It is proposed to largely eliminate the thermal neutrons with a neutron filter and to grade the enrichment in successive rings of fuel pins so as to obtain nearly equal average power in all pins. There will remain radially non-uniform power densities within individual pins which may be around 5 - 15%. In addition, the ETR axial power variation is marked, and neutronic shimming may be desirable.

It does not appear possible to install a neutron hodoscope or similar detector of fuel motion in the ETR. Therefore once a test has progressed beyond a stage where much of the initial instrumentation (especially thermocouples) has been lost, it will be difficult to deduce what is going on. In-pile instrumentation for observing fuel motion would clearly be a big help and possibilities are under study at ANL. At best, however, it appears that such detectors could give only very coarse spatial information on fuel position.

It is worth noting that the SLFF tests are fairly expensive (a current estimate is \$1.5M per module) and that only three tests per year are planned. This illustrates the problems of testing more than a few fuel pins, particularly when preconditioning is required.

For GCFR safety tests, consideration is being given to a Gas Reactor In-Pile Safety Test (GRIST) loop in the ETR. If constructed, such a loop would presumably be similar to SLSF except that the test fuel would be in a high pressure He environment, rather than in liquid sodium.

3. PBF

The Power Burst Facility is a water moderated heterogeneous test reactor which has been constructed and will hopefully soon be operating at INEL. The fuel is a mixture of oxides of uranium, zirconium, and calcium with a U^{235}/U^{238} ratio of 0.233, which in a rod, is surrounded by an insulating zirconia sleeve and stainless steel clad. PBF was designed to operate in a burst mode with a period as short as 1.3 msec and a half width less than 10 msec for a burst from low power. The reactor was also designed to operate at a steady power level of 40 MW.

During startup, PBF has shown power oscillations at powers ≥ 10 MW, which are currently under investigation. Moreover the reactor is heavily committed to light water reactor safety tests for several years. These tests do not require operation in a burst mode and it is projected that the present core will in 1978, be replaced by a core having no burst capability. Thus the abilities of PBF to meet design performance, and the availability of PBF for fast reactor safety tests are uncertain. Nevertheless it is important to examine PBF design capabilities since these are not duplicated in any other facility.

The reactor core is 36" high and contains an 8.75" diameter central test cylinder, into which a sodium loop with test fuel could be inserted. Moreover a burst could deposit several thousand joules/gm in enriched test fuel. An ANC-ANL study is underway to better define the steady power levels and burst energy that PBF could induce in enriched oxide fuel pins. Since PBF is a thermal reactor it is necessary to use a neutron filter and graded enrichment to obtain reasonably flat power distributions in a cluster of fuel pins, and a series of optimization calculations are required to determine the filter-enrichment combination which is best suited for a particular experiment. At present it appears unlikely that PBF, operating at 40 MW, can produce sufficiently flat and prototypic power densities in fuel to adequately restructure fuel. Moreover, when operating at such power, PBF would have no residual burst capability. However it is important to determine

PBF capabilities definitively before decisions are taken on the construction of major new test facilities.

It appears possible to modify PBF for fuel motion observations, for example by installing a hodoscope. In particular, a portion of the concrete reactor shield which was only bricked-in could be removed for a hodoscope slot.

4. ACPR

The Annular Core Pulse Reactor (ACPR) is a Triga type (uranium and zirconium hydride fuel, water cooled) test reactor located at the Sandia Laboratories in Albuquerque. The reactor can operate at 600 kw steady power or in the pulsed mode in which, from low power, a burst of 1.3 msec period and about 5 msec FWHM has been routinely obtained. ACPR has a 9" diameter central experiment tube and the active height of the core is 12".

The ACPR is a thermal reactor so that there are again trade-offs to be considered involving, on the one hand, neutron filters for eliminating thermal neutrons and obtaining uniform power distributions and, on the other hand, fuel enrichment to enhance energy deposition. With appropriate choices it appears possible to obtain energy depositions of ≈ 4000 J/gm with surface/center ratios 1.2 and in fuel rods ≤ 0.5 cm in diameter, or in small fuel samples. A sodium loop can be inserted in the test region so that single (short and therefore non-prototypic) pins can be subject to very rapid transients in a sodium environment. Such tests would appear to be of value in elucidating rapid transient phenomena. A radial neutron hodoscope could not be readily accommodated in ACPR but some observations of fuel motion could be attempted through the relatively unobstructed test channel.

At steady state, power levels of 2 w/gm appear attainable in test fuel beds of 4" diameter with power ratios of 1.15. Such conditions could be of interest for PAHR studies.

An upgrading of the ACPR is planned, with some of the inner hydride fuel to be replaced by fuel having higher temperature capability, probably BeO-UO_2 . With this modification, energy deposition during a burst would be increased by at least a factor two.

5. SPR III

The Sandia Pulsed Reactor III (SPR III) is a fast metal (93% U^{235} alloyed with 10 wt % Mo) pulsed reactor with a large central irradiation cavity, 7" in diameter and 16" in length. During burst, a fluence of fast (near-fission) neutrons in excess of 6×10^{14} is obtained in the core, with a FWHM $\approx 45 \mu\text{sec}$. This fast fluence will not produce interesting energy generation in reactor fuel but a small amount of moderator (CH_2) may be placed around small fuel samples to obtain increased heating, $\sim 3000 \text{ J/gm}$, with surface/center power ≈ 1.2 , for 1/16" fuel diameters.

While such very rapid transients are not of direct interest in reactor safety analysis, they may be valuable for exploring phenomena such as fuel-coolant interactions or equations of state of clean or irradiated fuel.

Other fast pulsed reactors could be mentioned with somewhat different characteristics. For example Godiva at LASL and SPR II at Sandia can give even shorter pulses, but have much smaller test regions. Super Kukla at NTS has a large test region.

6. EBR II

This small (62 MWt) fast reactor at INEL has been used for many years in steady state irradiations of fast reactor fuels and, until the FTR begins operation, it will be the only United States facility for such irradiation. At present, failed fuel is rather promptly removed from EBR II to minimize contamination. However a cover gas cleanup system will be installed and over the longer term closed loops could be installed in EBR II. Thus as FTR assumes the irradiation responsibilities of EBR II, consideration might be given to use of the latter for testing fuel failure thresholds, pin failure propagation, and

problems of operating with failed pins. In addition the behavior of fuel during anticipated extreme transients or during thermal (load-following) cycling could be investigated. A disadvantage would be the short length (12") of EBR II fuel.

7. FFTF

The Fast Flux Test Facility (FFTF), now under construction at Hanford, is a 400 MWt fast reactor, designed to serve as the United States' major test reactor for prototypic irradiation of fast reactor fuel. As such it will presumably not be available for transient tests which would be at all hazardous to operation of the facility. The reactor will have experiments involving fuel failure might safely be conducted and in principle, LOF simulations or subassembly blockages could be carried out in such loops. However such experiments would, at a minimum, contaminate loops and would probably pose unacceptable dangers to continued operation of the facility. Therefore it seems unlikely that the FFTF can be used for safety tests.

The FFTF will presumably serve as the source of irradiated fuel pins for testing in any new facility. Since there are potentially great benefits to be obtained from testing fully enriched fuel, it is important to know whether enriched fuel can be irradiated at prototypic power densities during normal operation of the FFTF. A preliminary HEDL study has indicated that fully enriched fuel could be irradiated in a few inner reflector positions. Even in such locations, boron filters were required to reduce the power to prototypic levels. Around 37 pins could be accommodated within the filter in each reflector position. The expected power variation across the 37 pins would be approximately 30%. Thus it appears possible to irradiate a limited number of fully enriched fuel pins in the FFTF but with non prototypic power gradients. It would presumably be easier to irradiate partially enriched fuel and, of course, large amounts of irradiated prototypic fuel will be expected from the FFTF.

8. Nuclear Furnace

During 1972, a 44 MWt reactor was operated at the National Reactor Development Station (NRDS), Nevada for the testing of fuels in the nuclear propulsion or Rover program.⁽⁴⁾ This "Nuclear Furnace-1" used UC-ZrC graphite fuel with hydrogen coolant. Additional water moderation and a thick Be reflector were utilized in order to minimize the reactor size and thus power level. Since Rover fuel was being tested the power densities were high, typical values being $4\text{ kW/cm}^3 \approx 10\text{ kW/gm U}^{235}$.^{(4), (5), (6)} The system was designed to operate for only a few hours and coolant was exhausted to the atmosphere. A series of tests had been planned using different cores with the same reflector, containment, control, and cleanup systems, but the series was terminated after the first test with the demise of the Rover program. Much of the facility is still in existence at NRDS, Test Cell C, and it could presumably be reactivated fairly rapidly. Might this reactor, with its exceedingly high power densities, be used for significant tests of fast reactor fuel?

The reactor was thermal and approximately 94% of the fissions were produced by thermal neutrons.⁽⁶⁾ If such neutrons were eliminated by a Cd filter without significantly affecting the epithermal flux in the test region, the power density would be $600\text{ W/gm of U}^{235}$. The remaining fissions are, however, mostly due to low epithermal neutrons having a median energy $\approx 40\text{ eV}$ so that the filtered power distribution would not be as flat as desired. In addition, the reactor was not designed with any rapid transient capability.

In most of the fuel test positions, fuel elements measuring 19 mm in diameter and 1.3 m in length were used. It would appear that a small sodium channel, together with a few FTR fuel pins could be inserted in a similar test position. A hodoscope could presumably be accommodated because of the modular construction of the core.

It thus appears that the nuclear furnace facility could be reactivated rather quickly and could be used for testing of fully enriched fuel at prototypic power for a few hours. However it would have very limited, if any, burst capability and is therefore probably not worth further consideration.

On the other hand, some of the NRDS facilities, including hot cell complexes, control facilities and test areas, should be considered for possible use in a new safety test program. We have collected information on the existing facilities, but it is not a part of this report.

B. Capabilities of Existing Facilities Related to Experimental Needs

The following table summarizes the capabilities and deficiencies of some of the existing facilities as discussed above. Some of these, particularly TREAT and the SLSF loop in the ETR, can be used effectively in near-term experimental programs. TREAT can continue to provide an adequate facility for some failure threshold tests with the important exception that preconditioning cannot be performed. The SLSF loop can be useful for some loss-of-flow and failure propagation experiments, particularly those in which a single end-point result is adequate to answer a specific question. It may also be possible to investigate the importance of preconditioning in the ETR. The ACPR may also prove to be a useful facility for some short period and some PAHR experiments. We recommend that the continued use of these facilities be vigorously pursued insofar as applicable.

FACILITY	TYPE	APPLICABILITY	DEFICIENCY	CAPABILITY TO FULFILL EXPERIMENTAL NEEDS
TREAT	A	Meets requirements for at least single pin for failure threshold and location. Suitable for long, pre-irradiated pins, if available.	<ol style="list-style-type: none"> 1. No preconditioning. 2. Long period (inadequate for EOS tests). 3. Poor power flatness. 4. Restricted test time due to Class I operation. 5. Inadequate hodoscope resolution. 	Fuel failure threshold and location tests for mild TOP's excepting the effect of preconditioning.
SLSF/ETR	A,B	Steady state operation for up to 61 pins.	<ol style="list-style-type: none"> 1. Very limited power transient capability. 2. No visual data acquisition. 3. Poor power flatness. 	LOF accident sequence studies without power transient effects and without visual data acquisition. Some fuel element failure propagation tests.
PBF	A	Short period burst capability with steady state operation for 1-7 pins.	<ol style="list-style-type: none"> 1. Burst operation questionable. 2. No visual data acquisition. 3. Non-prototypical preconditioning. 	Short time scale information provided the facility is available and operational.
ACPR	A	Short period burst capability from low power for a single pin. Large (9") experimental region for PAHR tests.	<ol style="list-style-type: none"> 1. Essentially no steady state capability (2 W/gm). 2. No visual data acquisition. 3. Poor power flatness. 4. Short core length 	Short period information without steady state capability for short (.3 m) fuel pins.
SPR-III	A	Very short period, low energy bursts for small material samples (FCI, EOS).	<ol style="list-style-type: none"> 1. Low fluence and energy deposition capability. 2. Short core length 	None planned, currently.
EBR-II		Reliable steady state machine which could eventually be employed as a fast spectrum ETR type driver.	<ol style="list-style-type: none"> 1. No bursts permitted. 2. Short core length. 	None planned, currently.

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VI.

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VII. New Facility Options

Summary

When the design studies which are reported in this chapter were being undertaken, the four facility types (A, B, C, and D) had not yet been defined. Therefore, most of the calculations were aimed at exploring design options and capabilities of a wide range of intermediate spectrum drivers for class III operation and it has not yet been possible to focus on specific designs for type A, B, or D facilities. However among the range of parameters which have been examined, it is apparent that there are examples with characteristics resembling those of the type A and B facilities and an attempt has been made to identify their capabilities. In addition, the possibility of an accelerator-driven type D facility has been briefly examined.

The scope of the work performed in the first phase of the new facility conceptual design studies has been to investigate the basic characteristics of a wide range of design alternatives. The principal objectives have been:

(1) to gain an understanding of the complex neutronic behavior of multi region mixed spectrum STF's by first studying simple systems and gradually working up to the complex multi region driver/converter systems.

(2) to delineate the relationships and magnitude of the large number of trade-offs that exist between alternative design features.

The effort has been focused on the study of facilities capable of operating in the class III mode.

The study started with a review of the properties of various candidate materials for driver and converter fuels, cladding, and coolants for application in STF systems. Selection of the materials with the most potential for such applications was aided by a series of neutronic and thermal-hydraulic

scoping calculations. The materials that were finally selected for further detailed study and application in conceptual designs were:

Fuels - BeO/UO_2 , (UC-ZrC)-graphite, and UO_2 -Cr

Clads - Stainless Steel and Niobium

Coolants - Liquid Sodium and Gas (He or CO_2)

Iterations between thermal-hydraulic calculations, which included some post-burst transient heat transfer, and neutronic calculations narrowed down the range of driver fuel elements configurations, fuel thickness, coolant volume fraction, etc. that would be suitable in a class III facility. More detailed parametric neutronic calculations were then made on single region driver systems with test regions having the capacity for 19, 37, and 217 or more test pins which were either prototypic or highly enriched. The characteristics of these systems, max. to min. power ratios across the test region, Figure of Merit (FOM) values (minimum power density in test fuel divided by maximum power density in driver fuel), heat removal requirements, etc., were then evaluated. These results were then used to make fairly simple design optimizations in more complex systems containing converter regions to reduce the max. to min. power ratios and enhance the FOM.

In order to estimate the reactivity effect of placing a diagnostic instrument slot (e.g. hodoscope) through the reactor a few Monte Carlo calculations were performed on medium size systems. For one design, it was calculated that a 4 in. wide hodoscope slot would have a reactivity effect of about - 0.10 in k. Filling the slot with 50v/o steel reduced this by a half, an effect which is important to the consideration of collimators.

Some Doppler coefficient, prompt neutron lifetime, and test region fuel relocation-effect calculations were made to obtain an appreciation of the problems that face an STF designer.

Based on the results of the above studies the following list of tentative conclusions has been made:

(1) Comparison of Fuel Materials.

With respect to neutronic performance, the BeO/UO_2 fuel is superior to the Rover and cermet fuels in that it is more reactive and thus allows more flexibility in the choice of design variables. The most promising function of the cermet fuel is that of a converter. The BeO/UO_2 also has superior thermal characteristics to the other two fuels.

(2) Comparison of Clads and Coolants.

The combination of stainless steel cladding with sodium coolant is superior, in its neutronic and its thermal performance to stainless steel or niobium cladding with gas coolant. The economic implication of gas versus sodium cooling has not been evaluated.

(3) 19 or 37 Test Pin Facilities.

In facilities with test regions having 19 or 37 pins, burst energy densities of the order of 2000 J/gm in prototypic test fuel or up to 8000 J/gm in highly enriched test fuel appear to be feasible in single driver systems, even when the burst is initiated with the test region fuel operating at 200 w/gm. The max. to min. power ratios across the test region of 1.10 to 1.20 are typical which means the possibility of no converter or at most a fairly thin one. The total heat removal requirements for such facilities might be as low as 50 MW for the enriched test fuel and somewhat less than

200 MW for the prototypic fuel. The reactivity effects of fuel relocation in the test region are estimated to be negligible for the 19 and 37 test pin systems.

(4) 217 Pin (or more) Test Facilities

The problems of designing a facility to accommodate 217 (or more) test pins appear to be much greater than for the 19 or 37 pin cases. The max. to min. power ratios across the test region tend to be severe in the single region drivers which means that converter regions are required. If the converters are not to become a limiting factor in the attainable FOM they must be dilute in fissile content and hence non-multiplying or marginally multiplying. This tends to increase the critical size of the facility to the point where very little latitude is left for design contingencies.

With complex driver/converter zoning, Class III burst energy densities in the test fuel of several thousand J/gm for highly enriched fuel and at least a thousand J/gm for prototypic fuel might be possible. If, as suggested in Chapter V for the type B facility, highly enriched test fuel can be used for studies of gross fuel/steel motion then it appears likely that an intermediate spectrum driver can be used for such tests.

However, the reactivity effects of highly enriched test pin fuel relocations might be a dollar or more.

(5) Kinetic Behavior

Although very little has been done in the area of system kinetics there is reason to believe that burst energy densities of the desired amount can be obtained in Class III operation of the systems being studied. Alpha calculations have given prompt neutron lifetimes in the range of about 2 to 5 μ sec for the BeO/UO₂ configurations of interest. Reactivity insertion rates of up to \$50 per sec. might be required depending on the Doppler feedback levels.

A. Introduction

The TREAT fast reactor fuel testing program has made major contributions to the understanding of fuel failure mechanisms and related phenomena. Considering that the TREAT reactor was built and designed more than a decade ago its experimental program has been remarkably cost-effective by today's standards. It has been long recognized, however, that the TREAT facility has some basic limitations in its functional capabilities, namely:

- large non-prototypic power gradients across the test region due to the strong thermal spectrum component
- its inability to provide sustained prototypic power levels in the test section
- the limitations on the size of the test pin bundle
- the long burst width due to the relatively long prompt neutron lifetime

The recognition of the need for a more comprehensive in-pile testing program has stimulated studies of the experimental requirements of such a program and the facility designs that could meet those requirements. Such studies have been in progress at ANL for several years and have included fairly specific design studies such as the TREAT Improvement Project (TIP). More recently ANL has formulated a design proposal for a safety test facility that can be operated in the Class II mode, i.e., one in which prototypic power densities can be sustained in the test region for several minutes. Optional modes of operation in the facility would include a full burst capability from ambient temperatures (Class I) to a more limited burst capability from intermediate power levels.

Since ANL has already performed a design study of a pure Class I facility, the TIP, and is currently making a detailed design study of a Class II facility, it was decided to limit the present scope of the LASL studies to the

more difficult conceptual design problems of a Class III facility *, i.e., a facility in which prototypic power levels can be maintained for days and in which a significant burst can be initiated without first reducing power. A smaller effort is being applied to a feasibility study of using an accelerator-driven subcritical multiplier to provide energy densities of interest to certain aspects of the fast reactor safety program.

The ANL Class II facility proposal consists of a system having three distinct non-test regions, a thermal driver of BeO-UO₂ (mole ratio 8000/1), an outer converter region of (UC,ZrC)-graphite (C/U atom ratios in the range 125/1 to 770/1), an inner converter of UO₂-Cr cermet, and a central test region large enough to accommodate at least one FTR subassembly. The purpose of the converter regions is to "convert" the thermal neutrons to fast neutrons by the process of fission in dilute ²³⁵U fuel mixtures. Because of the gross differential between a fast and thermal spectrum, a large number of zones with graded ²³⁵U concentrations in the converter regions are required to minimize power peaks at interfaces due to large spectral mismatches.

The employment of a thermal driver in a safety test facility has both advantages and disadvantages depending on the planned mode of operation. A shortcoming common to all operational modes is the relatively long prompt neutron lifetime. In spite of the fact that in the ANL Class II facility proposal the test region spectrum is relatively hard, the system lifetime is mainly determined by the thermal driver and as a result the facility has virtually the same limitations as TREAT in the area of rise time and burst width.

For Class III operation the disadvantages of a thermal driver outweigh the advantages. A thermal driver can have a very strong inherent shutdown

* The discussions in this section do not refer to the categories of facility types defined in Chapter V of this report. The reason for this is that this categorization was introduced only recently.

capability due to the sensitivity of thermal neutron leakage to moderator temperature. Although this is a definite advantage in a Class I system bursting from room temperature, too large a shutdown capability can seriously degrade the burst capability of a Class III facility. An ideal shutdown mechanism for a Class III system would be one which is inherent and large, but is delayed long enough for the desired energy densities to be reached before the negative reactivity feedback quenches the impressed positive reactivity that produces the burst.

A Class III system requires a continuous heat removal capability which implies a need for cladding. To maintain the reactivity advantages ($k_{\infty} \sim 2$) that a thermal driver can have over epithermal drivers (BeO/UO_2 50/1 ratio, $k_{\infty} \sim 1.7$), the amount of poisoning due to cladding has to be minimal. The ANL Class II proposal achieves this by having large blocks (4 X 4 X 8 in) of $\text{BeO}-\text{UO}_2$ clad with zirconium, but such large dimensions tend to degrade the performance. The small size fuel elements required for efficient Class III operation would involve prohibitive volume fractions of cladding for thermal drivers requiring high k_{∞} values.

Another major disadvantage of a thermal driver in a Class III facility is the relatively rapid loss of reactivity as burn-up proceeds, mainly because of the much smaller critical mass of the thermal driver compared with a fast or intermediate spectrum driver. Since a major objective of Class III operation is sustained high power for several days or more to precondition the test fuel, reactivity losses due to burn-up are important considerations, and probably preclude the use of a thermal driver. Consequently, the LASL scoping studies for Class III facility design alternatives have been limited to non-thermal drivers.

Objectives and Scope

The primary objective of the first phase of the STF conceptual design studies has been directed towards understanding the basic neutronics and

complex inter-relationships between the various components of postulated system configurations. To accomplish this a large number of scoping calculations have been performed in the areas of neutronics and thermal-hydraulics.

The starting point in the study was the selection from a large number of alternatives the fuel, clad, and coolant materials that are most likely to satisfy the stringent thermophysical properties requirements dictated by the severe environmental conditions anticipated in a Class III facility. To this end the recent ANL compilation and evaluation of material properties proved most useful.

Having selected the major material candidates, the next stage was to determine useful ranges of internal configurations (^{235}U enrichment, diluent to uranium atom ratios, etc.) and external configurations (fuel diameter, clad type and thickness, coolant type and volume fraction, etc.) for fuel element design alternatives. This entailed the performing of parametric infinite medium neutronic calculations to determine k_{∞} values, Doppler temperature coefficients, and spectral characteristics for consideration in one dimensional neutronic models of systems of various complexities. Also, a parametric steady state and transient heat transfer study was made on a number of fuel element configurations covering a broad range of heat transfer coefficients for both gas coolant and sodium coolant. Optimization studies were made on fuel-clad gap thermal conductances to balance clad and fuel maximum temperature constraints.

Based on the knowledge gained from the infinite medium neutronic and the thermal-hydraulic studies a large number of one dimensional transport theory calculations was performed to obtain the range of system performance characteristics that might be attained in various reactor configurations, from single region drivers to multizone driver/converter combinations.

Most of the calculations were for a test region containing 217 fuel pins, both of prototypic composition and highly enriched. A number of interesting calculations was made on facilities with 37 and 19 test pins. Core heights of 3 ft. and 4 ft. were included as parametric variables.

Constraints on time and manpower prevented completion of all aspects of the phase I scoping studies in time for this report. Studies need to be made on the kinetics, control (including burn-up effects), and safety considerations of the various design configurations. Some prompt neutron life-time calculations have been made and have been included in a few "back-of-the-envelope" burst-kinetics calculations to determine the range of feedback characteristics that might be acceptable for bursting from steady state power. In addition, several two dimensional transport theory calculations have been made to ascertain upper limits to the amount of positive reactivity that may result from fuel relocations in the test region. The implications of these results on the safety of the facility are mentioned in the following sections.

No real considerations have been given yet to the ex-core problems of the various conceptual designs, shielding, heat removal systems, etc. It is expected that the type of coolant used to cool the Class III operating facility will certainly effect the economics and could affect the safety of the facility. Qualitative and quantitative considerations of these aspects will increase as the study progresses to the more detailed design proposals.

B. REACTOR ANALYSIS

1. CORE MATERIAL SELECTION

Introduction

The magnitude of the design problems of a safety test facility operating in a Class III mode can be inferred from a brief list of its functional requirements, namely:

- (1) Provide an operating LMFBR (or possibly GCFR) environment for a number of fuel pins (possibly several hundred) for a period of days or weeks. The environmental characterization would include neutron flux magnitudes and distributions (in both space and energy), temperatures and temperature distributions.
- (2) Without reducing the steady state power level, initiate a fast burst that will deposit in the test pins an energy density that will melt or even vaporize the fuel without exceeding the design limitations of the driver region which creates the environment.
- (3) Provide sufficient space for diagnostic observations (hodoscope and/or x-radiography) of the test region at any time during the experiment.

Clearly, a central design problem is the one of finding a viable driver configuration which will withstand such extreme impositions, not just once, but many times and which will behave consistently each time. Since most of the reactivity is provided by the driver region it must have a built-in inherent shut-down mechanism which will protect the facility from any positive reactivity effects caused by fuel relocations in the test region or any inadvertent accident conditions in the facility.

For safety reasons the fissile component of the test fuel should be U_{235} . The question is, therefore, what compound, solid solution, or dispersion containing U_{235} will best satisfy the stringent requirements demanded of an STF driver fuel? The desirable characteristics of a driver fuel candidate can be divided into four categories - neutronic, thermal, mechanical,

and compatibilities with clads and coolants. The desirable neutronic characteristics are:

- a low parasitic absorption cross section
- a high scattering cross section
- a low slowing down power to maintain a hard or hard-epithermal spectrum (this implies medium or high atomic weight diluent atoms)

The desirable thermal characteristics are:

- good refractory properties - a high melting or failure temperature (failure implying loss of integrity)
- a high heat enthalpy between operating and failure temperatures
- a high thermal conductivity to maintain low steady state operating temperatures.

The desirable mechanical characteristics are:

- good high temperature strength and stability
- good resistance to thermal cycling which implies a good thermal conductivity, a relatively low thermal expansion coefficient, and a high fracture strength

A material having all of the above properties does not appear to exist. Two of the best materials, from the heat enthalpy view point, are beryllium oxide and graphite, both of which are good moderators. A compilation of the thermo-physical properties of any fuel candidate is likely to reveal a mixture of such advantages and disadvantages. In selecting potential candidates many can be ruled out quickly because they fail to meet some minimum requirements in certain areas such as heat enthalpy or neutron capture cross section. Further short listing of candidates can be made by evaluating which (of the candidates that have met the minimum requirements) have the best overall characteristics. Eventually, however, fuel candidates having similar overall

characteristics (based on some weighted criteria) will have to be compared in terms of the functional capabilities of the facility. Even this may not be conclusive. For any particular set of design objectives there is an optimum configuration of core materials. Within the context of a Class III facility there is a broad range of energy requirements, test sample sizes, and enrichments, all of which may not be compatible within any one test reactor. The problem is a systems one and ultimately the final selections will be based on some quantitative cost-benefit and cost-effectiveness analyses which consider and weight every aspect of the problem. Factors such as required fuel development programs (time scales and costs) must be considered.

The scope of the preliminary studies reported here could not include a complete systems evaluation of the problem of selection of reactor materials. Based on a review of the recent compilation and evaluation by ANL of fuel and clad material for safety test facilities plus some thermal-hydraulic and neutronic scoping surveys, the fuel materials which appear most attractive for a facility operating in the Class III mode are:

Binary oxide, BeO-UO_2

Binary carbide-graphite composites, $(\text{UC, ZrC})\text{-C}$ (Rover fuel)

Cermet, $\text{UO}_2\text{-Cr}$

The binary oxide $\text{UO}_2\text{-ZrO}_2$ (with CaO stabilizer) which is the basis of the Power Burst Facility (PBF) fuel has potential for Class I operating facilities but its very low thermal conductivity makes it a poor Class III candidate fuel. The cermet, $\text{UO}_2\text{-Mo}$, would be an attractive candidate if the separated isotope, Mo92 were readily available; it is possible that progress in isotope separation will make this a reasonable option within the decade.

In the area of cladding, two material types are of major interest, stainless steel and a refractory metal such as niobium, molybdenum, or vanadium. These are chosen in context with the two types of coolant selected, namely liquid metal and gaseous coolants. The initial design constraint of

providing the test zone of the reactor with a hard or hard-epithermal neutron spectrum restricted the coolant to the above broad groups of non-moderators.

It is anticipated that refractory clads would be used in conjunction with gas cooling because of the high temperature drops across the gas boundary layer. On the other hand the much smaller film temperature drop associated with sodium means that the clad temperature is closely coupled to the bulk sodium temperatures and therefore the lower melting point clad, stainless steel, can be used. Because of the strong absorbing properties of niobium, there are large advantages in design flexibility to be gained by using stainless steel. On the other hand a sodium cooled system may be significantly more complex and hence more costly than a gaseous system.

a. Thermal Properties

The importance of heat enthalpy and thermal conductivity to Class III operations can be seen in the following elementary treatment. Under Class I conditions (burst from ambient temperatures) the maximum amount of energy ($\hat{E}$) that can be deposited into the test fuel during a fast burst is subject to the fuel temperature constraint equation:

$$\frac{1}{\text{FOM}} \cdot \frac{\hat{E} \rho_T}{\rho_D C_D} < T_F - 300^\circ\text{K} \quad (1)$$

where FOM is the figure of merit which is defined as the ratio of the minimum power density (watts/cc of fuel) in the test region to the maximum power density in the driver fuel. $\hat{E}$ is the maximum energy deposition in the test fuel in joules/gm. ρ_T is the density of the test fuel. ρ_D is the density of the driver fuel. C_D is the average specific heat of the driver fuel between 300°K and the failure temperature, and T_F is the fuel failure temperature.

Equation (1) states simply that the adiabatic temperature rise in a burst should not cause the driver fuel failure temperature to be exceeded. Note that the product of ρ_D and C_D is the volumetric heat capacity, a fact which

is important in the driver fuel selection process. In fast adiabatic bursts from ambient conditions we could make use of the volumetric heat capacity up to the failure temperature. The thermal conductivity of the fuel is not important except for the thermal shock considerations.

In the Class III operating mode, the fuel temperature constraint equation changes to

$$\frac{1}{FOM} \left[\frac{\hat{E} \rho_T}{\rho_D C_D} + Q \sum \Delta T_i \right] \leq T_F - T_C \quad (2)$$

where Q is the steady state power density in the test fuel (about 2 kw/cc). $\sum \Delta T_i$ is the sum of the temperature drops at steady state across driver fuel, fuel-clad interface, and the coolant film, normalized to unit power density in the driver fuel. T_C is the bulk coolant temperature at the point of maximum fuel temperature. Note that C_D is now the average specific heat of the fuel between steady state operating temperature ($T_C + Q \sum \Delta T_i$) and the fuel failure temperature. Thus the best fuel for Class I operation may not be the best for Class III. Equation (2) shows that the maximum allowable energy deposition in the test region, under the constraint of driver fuel failure temperature, depends not only on the specific heat of the driver fuel but also on the thermal conductivity of the fuel, and the heat transfer characteristics of the coolant and of the fuel-clad interface, with its usual expansion gap. The design of the fuel element configuration is of extreme importance in Class III operation.

Class III operation, by definition, entails test section operation at prototypic powers for days or weeks. Thus cladding of the driver fuel elements is likely to be mandatory although unclad Class III operation has not been finally ruled out. If the frequent bursting causes cracking of the driver fuel, which is fairly likely, then the clad will be required for structural strength and stability. The constraint equation for the clad failure temperature involves transient heat transfer phenomena and cannot be described by a

simple analytic equation like (1) or (2). This treatment is left to Section B3. In B3, however, optimizations are made on the thickness of the gap between the fuel and the clad so that the clad failure temperature is not reached long before the fuel failure temperature constraint occurs, or vice versa. The mechanism being optimized is the rate at which heat is transferred to the clad during the cooldown.

Figure 1 is a plot of the volumetric enthalpy between initial Class III operating fuel temperature, T_i , and the fuel failure temperature as a function of T_i , for the three primary fuel candidates. Conservative failure temperatures were chosen; they were 1900 K for the UO_2 -Cr, 2200 K for the BeO- UO_2 , and 2600 K for the Rover fuel. These temperatures are at least 200° below the observed melting points. The enthalpy values corresponded to compositions of interest for safety test facility applications.

It can be seen from the figure that BeO- UO_2 is the most promising Class I fuel of the three (initial temperature 300°K). In the Class III mode it is superior until the initial fuel temperatures are about 1400°K after which the Rover fuel appears more promising. Based on heat enthalpy alone the cermet fuel is inferior to the other two over the whole range. This is mainly due to its significantly lower failure temperature.

As mentioned previously, in Class III operation other factors take on importance, such as the thermal conductivity of the fuel and the type of coolant. Figs. 2, 3, and 4 show the influence of these factors on the maximum energy that can be deposited in the test region based on the constraint of maximum allowable driver fuel temperature in equation (2).

The starting point for the comparison of fuel materials and geometries was the steady state heat transfer study reported in Section 3. Based on a steady state operating power in the test region of 2000 watts/cc of fuel maximum fuel temperatures for the three fuels were calculated as a function

of driver fuel diameter and FOM with fairly conservative heat transfer coefficients for sodium and gas cooling, 4.0 and 0.4 watts/cm²-°K respectively. The burst energy depositions required to reach (adiabatically) the conservative fuel failure temperatures were then calculated as a function of the FOM. Figure 2 indicates that at a large FOM value of 10, of the order of 9 kJ/gm could be deposited in the test fuel before BeO-UO₂ driver fuel melts when sodium cooling is used, and about 7.5 kJ/gm lower when gas cooling is employed. The $\bar{E}$ values for the Rover fuel and UO₂-Cr fuels are significantly lower. As we move down the FOM scale the initial fuel temperatures will increase. The thermal conductivity of BeO-UO₂ is about 5% lower than the Rover fuel which is in turn about 10% lower than the cermet fuel. The sodium cooled curve for the Rover fuel is nearly always below that of the sodium cooled BeO-UO₂ fuel, but the gas cooled counterparts intersect at an FOM value around 6.0. It will be seen later that FOM values of 8 or so might be attainable when the test fuel is fully enriched, but if normal prototypic fuel is used then FOM factors in the range 2 to 4 only are feasible depending on the number of pins in the test region.

Figure 2 also indicated that if a facility is designed to have more than one type of material in the driver and converter regions (the ANL design has all three fuels), then with sodium cooling, for example, the design FOM values in zones containing BeO-UO₂, (UC, ZrC)-C and UO₂-Cr will have to be optimized in the ratio 1:1.2:1.35; ("the chain is as strong as its weakest link").

Figures 3 and 4 illustrate the effects of fuel pellet diameter on $\bar{E}$ for sodium cooling and gas cooling, respectively. For sodium cooling the difference between 6.4 mm (0.25 in) and 12.7 mm (0.50 in) diameter fuel is only about 10.4 kJ/gm or less in $\bar{E}$ for each fuel over the whole range of FOM values. Turning to Fig. 4 for the gas cooled situation the gains in $\bar{E}$ capability in going from a 12.7 mm to a 6.4 mm fuel diameter are much larger, amounting to 1.5 kJ/gm in the case of the cermet. Depending on where the specific design is on the FOM scale, differences in $\bar{E}$ of 0.4 or 1.5 kJ/gm could range in

significance from a useful bonus to a viable/non-viable design situation.

It must be emphasized that the foregoing analysis and discussion is semi-quantitative. However, it does illustrate the complexities and types of trade-off that are inherent in Class III conceptual design studies. For any particular set of design objectives there is an optimum configuration of core materials.

b. Neutronics Characteristics

A significant, and perhaps the most meaningful, neutronic trait of a combination and physical arrangement of materials (fuel, clad, coolant) is the infinite system multiplication constant, k_{∞} . As described in Section 2 (Neutronics Analysis) an extensive survey of the aforementioned primary candidate core materials, based on the indicator, k_{∞} , resulted in the following nominal ranges for fuel uranium concentrations and ^{235}U enrichments:

<u>Cr-UO₂</u>	3% $\sim$ UO ₂ v/o $\gtrsim$ 10%
	50 $\gtrsim$ ^{235}U % $\gtrsim$ 93.2
<u>BeO-UO₂</u>	20:1 $\gtrsim$ Be:U $\gtrsim$ 100:1
	30 $\gtrsim$ ^{235}U % $\gtrsim$ 93.2
(UC-ZrC)C	50:1 $\gtrsim$ C:U $\gtrsim$ 150:1
	50 $\gtrsim$ ^{235}U % $\gtrsim$ 93.2

This large range of potentially useful uranium concentrations and enrichments permits significant design flexibility in meeting a wide range of possible design and operating requirements.

For nominal (e.g., ~ 0.4 mm) clad thicknesses and coolant volume fractions (e.g., ~ 0.15), the clad, coolant combination of stainless steel and a liquid metal is suitable for this entire spread of fuels characteristics.

Neutronically, a gas coolant in conjunction with stainless steel clad restricts slightly the lower bounds on the ranges of fuels uranium concentration.

and enrichments. This is due to the increased neutron leakage for cores with gas cooling.

Refractory metal clads (e.g., Mo, Nb, W) impose significant neutronic penalties relative to stainless steel clad. This will certainly restrict core design flexibility, i.e., uranium concentrations and enrichments which are near the upper limits of the specified ranges are required to maintain the necessary reactivity margin.

Finally, an interesting possibility exists for the use of refractory metals in both fuel and clad applications. If advanced isotope separation techniques now under development should prove economical and timely, then the availability of a relatively low absorption refractory, e.g., ^{92}Mo or ^{184}W could lead to other potentially attractive fuel and clad options.

Fig. 1. Volumetric Enthalpy in kJ/CC Between T_i and Estimated Fuel Failure Temperature

Fig. 2. Max Burst Energy Deposition vs. FOM Reactor for Candidate Fuels.

Fig. 3. Max Burst Energy Deposition vs. FOM Factor

Fig. 4. Max Burst Energy Deposition vs. FOM Factor for Candidate Fuels

2. NEUTRONIC ANALYSIS

a. Multigroup Cross Section Generation

The neutronic calculations for the basic scoping studies were based on the broad group cross section set FTR 300S.⁽¹⁾ This is an ENDFB/2 set adjusted to give k_{eff} and cross section ratios comparable to ENDFB/3 for dilute fast systems. The 42 group structure is given in Table 1 and includes a thermal group. For most of the groups the lethargy interval is 0.5 except around some of the important resonances in oxygen, iron and sodium, in which cases narrower intervals were used. Set FTR 300S was constructed for FTR shielding studies which explains the structure below the 28KeV iron resonance where the S-wave interference dip acts as a window. Set 300S does not contain niobium and zirconium and so these were added from the ENDFB/4 data file.

The cost and turn-around time of a computer run obviously is dependent on the problem size. Since one and two dimensional transport calculations were planned for the neutronic studies it was decided to group collapse the 42 group set to a 20 group set using the 1DX⁽²⁾ multigroup diffusion theory code. The collapsed group structure is also shown in Table 1 and consists of lethargy intervals of unity for all but the thermal group. For the various driver and converter configurations, infinite medium geometry was used in the 1DX group collapsing calculations. Since the resulting macroscopic cross sections would be used in transport calculations, the transport cross section rather than the diffusion coefficient was averaged in the collapsing process.

The rationale for the infinite medium calculation was twofold. First it reduced the problem size for each of a large number of parametric driver and converter compositions. Second, the output of 1DX yielded a value of K_{∞} for each composition and this parameter provided a useful guide as to whether or

not the finite system would possess the desired excess reactivity, (go critical) after allowing for appropriate diagnostic instrumentation slots through the facility.

The LDX code uses the Bondarenko⁽³⁾ f-factor table approach in computing the resonance self-shielding cross sections, interpolations being made over a range of temperatures and effective potential scattering cross sections. Binary cell heterogeneity is introduced by a Bell⁽⁴⁾ type formula involving fuel and "moderator" dimensions.

b. Infinite Medium Calculations

As outlined in Section B1 of this chapter consideration of the properties of candidate fuels for safety test facility applications resulted in the short-listing of three potential fuel types - BeO/UO₂, UC,ZrC-graphite, and UO₂-Cr.

The LDX calculations were performed on a wide range of diluent to fissile atom ratios and enrichments for the above three fuels. For the most part two basic fuel-clad-coolant configurations were calculated:

(1) 12.7 mm (0.5 in.) diameter fuel, 0.381 mm (15 mil.) thick stainless steel clad with a 0.254 mm (10 mil.) gap between fuel and clad, and 15 v/o sodium coolant.

(2) Fuel, clad and gap as in (1) but with 25 v/o gas coolant. Only a few unclad gas cooled configurations were calculated because emphasis was being placed on potential Class III facilities.

In the heat transfer studies three fuel diameters were studied, 6.4 mm (0.25 in.), 12.7 mm (0.5 in.) and 19.1 mm (0.75 in.). It was evident that 19.1 mm diameter pins would result in steady state fuel temperatures which would, at least for gas cooling, severely limit Class III burst capability. In the neutronic studies a standard diameter of 12.7 mm was used. Comparative neutronic calculations were made for only a few 6.4 mm cases. It will be seen later that the main penalty incurred by the use of the smaller diameter fuel is to increase

significantly the critical size of the facility, because of the poisoning effect of an increased clad volume fraction.

BeO/UO₂ Fuel Configurations

Table 2 and Fig. 1 show the 1DX calculated values of k_{∞} for the 12.7 mm diameter BeO/UO₂ fuel in various configurations, all at a temperature of 300°K. The values for the gas cooled systems are slightly higher than for the sodium cooled ones due to the neutron capture of the latter. However, these are infinite medium characteristics. In finite systems the reduced leakage from the sodium cooled configurations more than makes up for the extra absorption. The same table also shows the considerable reactivity penalties incurred when niobium cladding is used; k_{∞} reductions in the range 0.1 to 0.45 are observed depending on the fuel molecular ratio and the enrichment.

In part 1d of this section it will be seen that when other factors are taken into account, namely FOM (figure of merit) factors and Doppler shutdown, BeO to UO₂ molecular ratios around 50/1 and enrichments in the range 40-60% are of principal interest.

Table 3 compares values of k_{∞} for the 6.4 mm and the 12.7 mm diameter BeO/UO₂ fuel pins in a sodium cooled, stainless steel clad configuration. Significant reductions in k_{∞} are observed for the smaller pins.

Since the inherent shutdown capability of a non-thermal spectrum driver may have to come from Doppler coefficients, macroscopic cross sections were computed at 1200°K, a conservative value for steady state operating conditions, and 2100°K, a plausible value for the limiting upper temperature under burst conditions.

Values of Δk_{∞} due to Doppler for the temperature ranges : 300° to 1200°K and 1200°K to 2100°K are given in Table 4A. Graphical displays of the quantities are more informative and are shown in Figs. 2 and 3. They show broad maxima between BeO to UO₂ molecular ratios of 50/1 and 100/1. In

the areas of interest there are no significant differences in the Doppler effects for sodium and gas cooled configurations.

Table 4B give the 20 gp. spectra for a wide range of BeO/UO_2 ratios and enrichments. In all but one case the percentage of thermal neutron flux is less than 1%. However, they all exhibit strong flux components below 100 eV.

(UC,ZrC)-Graphite Fuel Configurations

A similar set of calculations was performed for (UC,ZrC)-graphite Rover fuel configurations. The results are shown in Tables 5, 6, and 7, and in Figs. 4 and 5. Comparisons with the BeO/UO_2 fuel results are interesting. In spite of the fact that, for a given moderator to uranium ratio and uranium enrichment, the fissile atom density in the Rover fuel is higher by about 8%, the k_∞ is significantly lower (in the range 0.175 to 0.225 lower) than for the BeO based fuel. This is attributable to the superior neutronic properties of Beryllium, namely significant (n,2n) reactions and better scattering properties.

Again the use of niobium cladding severely depresses the reactivity of the configuration. Figure 4 shows a minimum value of k_∞ for the unclad gas cooled configuration in the region of $\text{C}/\text{U} = 150$. No such minimum occurs for the stainless steel clad Rover fuel. As the C/U ratio increases the effect of the stainless steel absorption increases, causing k to decrease monotonically.

The changes in k_∞ due to Doppler temperature broadening for sodium cooled configurations are shown in Table 6 and Fig. 5. Although the failure temperature for the Rover fuel is several hundred degrees higher than the BeO/UO_2 fuel the upper temperature for the Doppler calculations were standardized to 2100°K for purposes of comparison. The Doppler coefficients appear to be about 10 - 35% lower than those for BeO/UO_2 configurations due to the harder spectra of the Rover fueled systems for a given moderator to uranium ratio. Neutron energy spectra (twenty group) for various Rover fuel configurations are given in Table 7.

UO₂-Cr Fuel Configurations

Table 8 gives the results of a number of infinite medium 1DX calculations for UO₂ cermet with different metal matrices. It is clear that it is not possible to use Mo or Nb based cermets in hard spectrum single driver safety test facilities. In fact it was found that even as converters these materials impose severe reactivity penalties, resulting in substantially increased system sizes.

Because of the hard spectrum the Doppler coefficients for the Chromium cermet configurations are much smaller than for the Rover and BeO/UO₂ fuels. From 300°K up to 2000°K the change in k_{∞} for 10w/o UO₂-Cr at 30% enrichment is only - .011, a factor of about 9 lower than for the 30% enriched 50/1 BeO/UO₂ case.

c. Reactor Model

As a starting point for the STF scoping studies the central region of the facility was assumed to be the same as that proposed by ANL⁽⁵⁾. This consists of:

- (1) A 115.6 mm (4.551 in.) diameter test section containing a homogenized FTR subassembly either of prototypic composition or with the uranium dioxide fully enriched.
- (2) A 3.20 mm (0.126 in.) thick stainless steel annulus simulating the subassembly wall.
- (3) A 53.29 mm (2.098 in.) thick region which could accommodate additional test fuel in the form of partial subassemblies but which in this model is filled with sodium.
- (4) A 28.60 mm (1.125 in.) thick region consisting of the test region containment vessel and thermal insulating layers. This region is assumed to be equivalent to steel at 76.43% full density.

*FTR inner core subassembly (22.43 w/o Pu in U + Pu)

This 285.8 mm (11.252 in.) diameter central region is surrounded by a driver region or by a system of converter and driver regions. The system has a 300 mm thick nickel radial reflector at 90% density, and axial nickel reflectors having coolant volume fractions identical to the driver and converter regions. In the one dimensional studies these axial reflectors were taken into account by the use of axial buckling factors in the DTF-IV code. Calculations showed that a 300 mm thick 90% dense nickel reflector is effectively (99%) infinite.

The simple one dimensional scoping studies did not, and obviously could not, incorporate explicitly diagnostic instrumentation slots (such as for a hodoscope) or discrete control rods. These were allowed for in a parametric way by varying the desired k_{eff} in the radius search calculations. Typically a k_{eff} of 1.10 or 1.15 was used.

Monte Carlo Calculations of Slot Worth

In order to obtain an estimate of the reactivity loss due to placing a hodoscope slot or other diagnostic slot in the facility some Monte Carlo calculations were performed on simplified single driver facilities using the 3-dimensional continuous energy Monte Carlo Code MCNK.⁽⁵⁾ In the model a 115.6 mm wide slot penetrated the radial reflector and driver region on both sides of the test region. The height of the slot was 1314.4 mm, i.e., it extended through both the axial reflectors.

Two types of drivers were considered, a 25v/o gas cooled UO_2 -Cr driver having a 730 mm outer radius, and a 25v/o gas cooled 50/1 Rover fuel driver having a 900 mm outer radius. Three Monte Carlo runs were made for each configuration:

- (1) System without a slot
- (2) System with slot containing VOID
- (3) System with slot containing 50 v/o stainless steel.

The third run was made to ascertain how much reactivity might be gained or lost if collimators were to be required in the design.

The results of the Monte Carlo calculations are shown in Table 9. Relative to the unslotted reactor the Δk s due to the voided slots are essentially the same. It was somewhat surprising to note that filling the voids with 50v/o steel recovered half of the reactivity loss in both cases. It was anticipated that the absorbing effect of steel in the softer spectrum Rover driver would reduce significantly the effectiveness of the steel.

It is too early in the overall STF study to know what kind and how many slots will be desired or can be accommodated in a given design. Whereas a hodoscope slot might have a large amount of steel collimator material thus reducing the reactivity impact of the slot, an aperture for x-radiography might be in the form of a steadily diverging slot from one side of the facility to the other, having the same width as the test section when it intersects it (the slot would be discontinuous through the test section, of course).

d. Single Driver Region Systems

Although it was anticipated that an STF driven by a simple single composition region would not be a viable design, it was nevertheless considered useful to study such systems in order to obtain an estimation of the relative capabilities of each fuel candidate and a starting point for progressing towards more viable but more complex designs.

In the tables which summarize the single region driver results, a number of characteristics and capabilities are listed; the less obvious ones are defined as follows:

Test Region.	217P symbolizes a 217 pin test region containing an homogenized FTR subassembly (prototypic composition). 217E is the same except that the UO_2 in the FTR fuel is fully enriched (93.2%).
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R_c	is the outer radius of the driver region (in cm) which gives the value of k_{eff} listed, usually 1.10 or 1.15 to allow for hodoscope slots and control.
M_c	is the mass of ^{235}U , in kg, in the driver region only
P	is the continuous power in MW which must be removed from the system (including the test region) when it is operating under Class 3 steady state conditions corresponding to 200 watts/gm max. power rating in the test region.
Test Max/Min	is the maximum to minimum power density over the test region. For the BeO/UO_2 and the Rover fueled drivers the maximum occurs at the outer edge of the test region.
FOM	is the Figure of Merit which is defined as the ratio of the minimum power density (watts per cc of fuel) in the test region to the maximum power density in the driver region. As mentioned in Section B1 the FOM is the major factor determining the burst capability of the facility.

It is interesting to note that the ANL definition for the FOM is tied to the average power density in the test region, not the minimum. Our FOM values will thus be more conservative.

(i) BeO/UO_2 Drivers.

A number of single driver BeO/UO_2 systems were calculated using the DTF-IV (S4 approx.) code with the following variations:

- (1) BeO/UO_2 molecular ratios in the range 20/1 to 150/1
- (2) Three ^{235}U enrichments, 93.2, 50.0 and 30.0%

- (3) Gas cooled and sodium cooled configurations
- (4) Prototypic and enriched fuel in the test region
- (5) 3 foot and 4 foot high cores
- (6) Parametric eigenvalue $k_{eff} = 1.10$ and 1.15
- (7) Driver fuel temperatures at 300°K , 1200°K , and 2100°K
- (8) 12.7 mm and 6.4 mm fuel pellet pin diameters

The results of the DTF calculations for the BeO/UO_2 driven systems are given in Tables 10, 11, and 12, and in Fig. 5.

Examination of the results in Table 10 and Fig. 5 shows that an FOM of about 4 can be attained with a modest max. to min. power ratio of 1.17 (case 1) in highly enriched test fuel. From Fig. 2 of Section B1 it can be seen that an FOM of about 4 will allow several thousand Joules/gm to be deposited in a burst. However, the system as described in case 1 has at least two major weaknesses. First, the system would lack an inherent shut-down capability since the Doppler coefficient is negligible. Second, the FOM for prototypic fuel in the test region would be only around unity which would preclude any significant burst capability.

A higher FOM and a significant Doppler coefficient can be achieved by reducing the enrichment of the driver fuel to the 60-40% range, and by increasing the BeO to UO_2 molecular ratio. Case 13, for example has an FOM of 8.3 and a Doppler reactivity loss of about $2\frac{1}{2}\%$ Δk between steady state operating temperature and fuel failure temperature. Unfortunately, this carries with it two penalties, an increase in the max. to min. ratio and an increase in the critical size of the system. The outer radius of the driver is nearly 120 cm and the max. to min. has increased to 1.57. Furthermore if an excess k of more than 0.10 is required it is unlikely, based on the cross sections used, that the system could go critical (see case 14). If the core height is increased from 3 ft. to 4 ft. (case 14) the critical driver volume

is reduced by 37% and surprisingly the FOM increases to 9.1. This latter change is, as yet, not understood and might be associated with the approximations of the one-dimensional model.

The FOM is reduced by about a factor of three when prototypic fuel occupies the test region (cases 5, 10, 20, and 26) and the thermal heat rejection required to maintain prototypic power rates in the test region (~ 200 w/gm peak) increases by about a factor of four. The only improvement appears to be a reduction in the max. to min. power ratio.

Table 11 shows a comparison of the characteristics of systems with 6.4 mm (0.25 in.) and 12.7 mm (0.50 in.) diameter BeO/UO₂ fuel. The clad thickness and the sodium volume fraction are the same in each case. The major effect is an increase in the critical radius for the smaller diameter fuel, an increase which becomes large for the low enrichments (cases 3 and 30). The FOM appears to be relatively invariant for the two fuel diameters.

The characteristics for gas cooled BeO/UO₂ drivers are given in Table 12. Apart from increased critical sizes the results are quite similar to the sodium cooled systems. Some results are shown for 0.508 mm (20 mil) niobium clad fuel elements. Large penalties in critical size are evident. The max. to min. ratios are reduced only slightly by the absorption of low energy neutrons by the niobium.

(ii) (UC,ZrC)-Graphite Drivers

A number of DTF calculations were made with a single driver region containing Rover fuel. Fewer calculations were made because the relative trends were similar to the BeO/UO₂ fueled systems. In addition to gas and sodium cooling with stainless steel cladding, some gas cooled unclad configurations were made although these would probably be limited to Class 2 operating systems. A summary of the results is given in Table 13.

As suspected from the comparison of the k_{∞} values, the critical sizes

of the Rover fueled systems are significantly greater than those of the BeO/UO₂ systems. Because of the harder spectra and the larger core volumes for a given moderator to uranium atom ratio (C/U or Be/U) the test region max. to min. power ratios and the FOM values are significantly improved in the Rover systems. (compare case 44 with case 6). However, when the enrichment is reduced in order to obtain a reasonable Doppler coefficient the critical size increases remarkably (cases 44 and 45, and cases 49 and 50). With uncertainties in the validity of the cross sections for such systems and with uncertainties in the required amount of k_{ex} for control and diagnostic slots, the available latitude with Rover fuel configurations leaves little room for contingencies. Raising the core height from 3 feet to 4 feet of course would help considerably.

At first sight the systems with FOM values of 14.6 and 15.4 (cases 46 and 47) look very attractive even with their high max. to min. ratios which might be reduced by appropriate converters. Again, however, there is the problem of obtaining the desired Doppler and also, if unclad fuel elements for Class III operation are impractical, the addition of steel cladding would reduce greatly the available reactivity at such C/U ratios.

(iii) UO₂-Cr Cermet Drivers

As mentioned in Section B1 the cermet with the most potential for STF applications appears to be a UO₂ solution in chromium. This is certainly more true for their consideration as hard spectrum drivers rather than converters, the k_{∞} values for the Nb and Mo based cermets being so much lower.

The initial approach has been to investigate the potential of UO₂-Cr fuel configurations for drivers. FOM values as well as material properties, limit the range of investigation to UO₂ volume fractions in the cermet between about 0.075 to 0.125. Table 14 summarizes the results of the calculations. Again

we find a "squeeze play" between critical sizes and the FOM values. All the systems have an enrichment of 93.2%. The rationale for this is that the Doppler coefficients for the hard spectrum cermet in the range of interest are about a factor of twenty lower than the BeO/UO_2 fuels in their range of application. Furthermore, lowering the enrichment causes serious reductions in the excess reactivity as is evident from the k_{∞} values in Table 8.

Calculation 56 has a good FOM value, and a factor common to all the cermet drivers, the max. to min. power ratio in the test fuel is not only small but has a convex shape, which is appropriate for central subassemblies in power reactors. The critical size, mass, and power, however, are very large partly as a consequence of Nb cladding. Case 65 with a 4 foot high core shows more promise. It is of interest to note that some preliminary calculations with two dimensional TWOTRAN calculations indicate that the axial reflector savings assumed for the cermet DTF calculations (about 13 cm) may be too conservative; a k_{eff} of 1.16 was calculated with TWOTRAN compared to 1.10 for the DTF calculation, for a 3 foot core. The main disadvantages of case 65 may be the small expansion coefficient available from a taller core and its negligible Doppler coefficient. Expansion coefficients have yet to be calculated for any of the systems so far studied. Also, the question has to be answered as to whether expansion coefficients can be relied upon in burst facilities in which driver fuel cracking and ratcheting might occur. On the other hand very dilute UO_2 -Cr cermets might have superior behavior over ceramic fuels in this area.

Summary of Single Region Driver Results

The following tentative conclusions can be drawn from the results of the neutronic calculations of potential single driver (no converter) class III facilities; they pertain only to systems with 217 pin (or larger) testing requirements.

(1) Of the three fuel candidates the BeO/UO₂ configurations offer the most flexibility mainly because of their higher k_{∞} values. The Rover and UO₂ fuel configurations suffer from lower k_{∞} values which impose severe constraints on design flexibility.

(2) Considerable reactivity gains (and hence flexibility) can be achieved by increasing the core height to 4 feet or more.

(3) Significant Doppler coefficients are achievable with the ceramic fuels but cermet drivers would have to rely on expansion effects for inherent shutdown. Increasing the core height would reduce the latter.

(4) Sodium coolant is neutronically superior to gas coolant in that the effect of the extra absorption is far outweighed by the effect of lower leakage.

(5) Fairly high FOM values in the range 6 to 8 or even higher are feasible but only with highly enriched test fuel. When prototypic (FTR) fuel is used the feasible FOM values are reduced to about 2 or less and the total heat removal requirements are increased by about a factor of 4.

(6) In the case of the Rover and BeO/UO₂ configurations the higher FOM values carry with them the disadvantages of high max. to min. power ratios in the test region. The UO₂-Cr cermet fuel has excellent max. to min. ratios but the higher FOM values are bought at the cost of much larger critical sizes (this is true to a lesser extent with the other fuels).

e. Single Region Driver Plus Converter Systems

The strong conflict among four of the major design criteria - FOM, max. to min. power ratio, inherent shutdown capability, and system size - is a fact-of-life in Class III conceptual design studies. A large number of trade offs exist and involve many non-neutronic considerations. In order to achieve a fairly high FOM with an acceptable max. to min. power ratio, resorting to a fast converter region is likely to be necessary. Then the spectral mismatches, between the driver and converter, which can cause performance-limiting power peaks, have to be smoothed out gradually. Several or more zones of increasing converter fissile density are usually considered for this purpose. The greater the mismatch the more complex are the zoning requirements as is evident in the extreme case of the ANL design with its thermal converter and eighteen or so graded zones. Figure 2 of Section 1 makes the point that the FOM values in UO_2 -Cr fuel have to be significantly higher than for UO_2 -BeO due mainly to the lower maximum allowable temperatures for the cermet. Optimization of a BeO/ UO_2 driver and UO_2 -Cr converter tends to result in a converter which is, if anything, marginally multiplying in the finite system. This makes the system much larger than the driver-only situation.

A number of calculations were made with a combination of BeO/ UO_2 (or Rover fuel) drivers and UO_2 -Cr cermets. The first studies were done on simple 1 + 1 systems (single driver plus single converter region). The results are summarized in Table 15. In case 67 the positioning of a 7 cm thick cermet converter between the test region and a 100/1 BeO/ UO_2 , 50% enriched driver has the following effects:

- (1) The critical radius increases from 63.8cm (case 18) to 102.4 cm, and the total power from 101 MW to 265 MW.
- (2) The FOM value for the driver decreases from 8.4 to 7.6, mainly as

a result of depressing the flux levels in the test region.

(3) The max. to min. power ratio has decreased from 1.63 to 1.28

A study of the radial power distribution shows that the FOM for the converter corresponds to a point close to the boundary of the converter and driver. Appropriate fuel density grading in this area could increase significantly (20%) the converter FOM. Cases 68 and 69 show the effects of increasing the converter thickness by two 3 cm steps. The critical radius increases each time and the max. to min. ratios slowly decrease at the expense of reduced FOM values in both the converter and the driver.

f. Multi-Region Converter/Driver Systems

There are three obvious ways of improving the FOM values above those of the single driver - single converter system configuration:

(1) Gradethe converter with zones of decreasing fissile fuel density as the converter-driver boundary is approached

(2) Generally gradethe driver region to produce conventional power flattening i.e., increase the fissile density in the outer regions

(3) Lower the moderator to fissile atom ratio in the driver near the driver-converter boundary to preharden the spectrum.

Some first attempts at optimization are shown in Table 16. The 217 enriched fuel pin test region is surrounded by three converter zones and five driver zones. Modest test max. to min. power ratios (less than 1.08) are achieved with fairly high FOM values. The optimization, however, was very crude and significant improvements might be possible with fine-tuning. With so many parameters having a cross-impact upon cost and other factors, fine-tuning optimization is difficult and time consuming.

Both of the systems described in Table 16 are 3 feet high and the k_{eff} converged upon was 1.10. More flexibility can be achieved by increasing the

core height to 4 feet. In spite of the fairly high FOM values the thermal heat removal requirements are rather high, 200 MW or more, so that if 217 prototypic pins were to be tested under class^{III} conditions the total power might be prohibitively high, 1000 MW or so!

g. 19 and 37 Test Pin Studies

The scoping studies of the previous sections focussed on the conceptual design of facilities which could test one or more subassemblies in the Class III mode of operation. The standard model (adopted from the ANL proposal only as a convenient starting point) consisted of a central 217 pin test section separated, by a 3.2 mm steel wall, from an outer annular test region capable of accomodating six half-size subassemblies. In the scoping calculations performed to date the outer test region was assumed to contain only sodium.

Moving the inner boundary of the driver 9 or 10 cm towards the center, by removing the outer test region and reducing the number of inner test pins from 217 to 37 or 19, would not necessarily result in a decreased driver outer radius. The critical size change would depend on the relative reactivity of the removed test pins to that of the repositioned driver fuel and displaced sodium. For the smaller 217 pin test facilities the k_{∞} of the driver is greater than that for the test fuel (particularly when it is prototypic test fuel) so the critical size will decrease; compare cases 8 and 74 in Table 17. On the other hand the large facilities actually increase in size when 37 pins only are used; compare cases 15 and 80.

The results of some initial scoping calculations on 37 and 19 pin test regions are very interesting and are summarized in Table 17. Comparing case 74 (37E) with case 8 (217E) the FOM value has increased from 7.4 to 10.2 and

is accomplished by a most welcome change in the test max. to min. power ratio, 1.46 down to 1.22. As we see in case 75 the FOM for 37 prototypic pins is a more healthy 3.1 and the test max. to min. ratio is only 1.10. Equally encouraging is the fact that the required heat removal for prototypic preconditioning is about half that of case 10.

Further reduction in the number of test pins to 19 results in even greater improvements. With prototypic test fuel the max. to min. power ratio is only 1.08 and the FOM is as high as 3.4 which implies a class III burst capability of possibly 2500 J/gm with sodium cooling of the driver section. An important ramification of the results of cases 74 through 77 is the fact that they have high reactivity drivers ($k_{\infty} = 1.55$) and this means much more latitude for control flexibility and contingencies.

When a 4 foot high core was considered (cases 80 and 81) for a 50/1 BeO/UO₂, 30%E sodium cooled configuration with 19 test pins, a prototypic FOM value of 4.5 was calculated with a modest max. to min. power ratio of 1.14 (which could be reduced further by a thin converter). The critical radii are actually greater than for the 217 pin facility (case 15).

Cases 82 and 83 are calculations for a 4 foot high $7\frac{1}{2}$ v/o UO₂-Cr cermet driven 19 pin system. The critical size is larger for the 19 pin than for the 217 pin facility (compare cases 65 and 82). More noteworthy is the fact that there is no significant change in the FOM value, 11.8 and 11.7. This points to one of the main mechanisms causing FOM changes between 217 pin and 37 or 19 pin test regions, namely the 'by definition' effect of the max. to min. power ratio in the test region on the FOM. The FOM is defined in this report * as the ratio of the minimum power density in the test fuel to maximum power density in the driver fuel. Thus as the "dished"

* ANL defines their figure of merit as the ratio of the average power density in the test region to the maximum power density in the driver.

shape of the power density through a test region becomes flatter the FOM increases. In the case of a cermet driver the power profile has a very shallow inverted dish shape because of the hard spectrum. Another probable mechanism for the increase in the FOM for the BeO/UO_2 driver systems is the decrease in mass of structural material and sodium between the driver and the test region. This effect would be much greater with a soft spectrum BeO/UO_2 than for the cermet. As a result of these mechanisms it is likely that the differences between 37 pin and 217 pin systems with efficient converters will be slight.

h. Kinetics Considerations

The major and most demanding function of the safety test facility under consideration will be to deposit, in a short pulse, a large amount of energy into the test fuel without damaging the driver. In the Class III operating mode the burst will be superimposed on prototypic steady state power levels in the test fuel. The success and safety of such an operation will require the designing into the system of a number of inherent and programmed characteristics.

Clearly, if several thousand J/g are to be deposited within tens of milliseconds a large and rapid insertion of positive reactivity will be required and the prompt neutron lifetime of the driver should be small. It is unlikely that a mechanical shutdown device could be relied upon to render the system sub-prompt critical before the driver fuel would be destroyed. Some inherent negative feedback will be required to accomplish this, the most suitable mechanism in epithermal systems being driver fuel Doppler. However, there may be a need for a programmed mechanical shutdown device to prevent the driver fuel from being overheated by the subsequent power plateau but the consequences of small time delays are not as critical at this point.

DTF alpha calculations on single region BeO/UO₂ driver systems have indicated prompt neutron lifetimes of 1.5 μ sec, 5 μ sec, and 14 μ sec for the 50/1, 100/1 and 150/1 molecular ratios, respectively, the ²³⁵U enrichments being 50%. This compares with a value of about 300 μ sec in thermal BeO/UO₂ driver systems. When the temperature of the 50/1 driver fuel is uniformly raised from 300°K to 1200°K, and then from 1200°K to 2100°K, the calculated reactivity losses due to Doppler broadening are $2.8 \times 10^{-2} \Delta k/k$ and $1.3 \times 10^{-2} \Delta k/k$, respectively.

Some simple point kinetics calculations have been performed on systems having the above ranges of characteristics, to ascertain the reactivity insertion rates that will likely be required to yield the desired burst depositions. A β_{eff} of 0.007 and a single λ decay constant of 0.076^{-1} sec were used.

Starting from a steady state power level of 200 watts/cc in the driver fuel, which corresponds to an FOM value of 10, ramp rates in the range of 25 to 50 \$/sec and total reactivity insertions of around 2\$, produced energy depositions in excess of 4000 J/gm in the test fuel, burst widths of half-max around 10 msec. Linearized (with temperature) Doppler feedbacks in the range calculated for the BeO/UO₂ driver systems turned the power around and reduced the reactivity to below prompt critical.

Calculations with initial steady state power densities of 500 watts/cc in the driver, corresponding to an FOM value of 4, yielded burst depositions of more than 2000 J/g in the test fuel with the same range of ramp rates and feedbacks. As the initial power increases (FOM value decreases) it becomes increasingly more difficult to obtain the desired burst depositions. The above calculations were preliminary and of an exploratory nature. More sophisticated calculations are being planned with coupling of transient heat transfer with the system neutronics.

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TABLE 1. BASIC 42 GROUP CROSS SECTION AND 20 GROUP CROSS SECTION STRUCTURES

<u>E</u>	<u>Δu</u>	<u>42 GP SET</u>	<u>20 GP SET</u>	<u>E</u>	<u>Δu</u>	<u>42 GP SET</u>	<u>20 GP SET</u>
10.0 MeV				1.458 KeV			
6.065	0.5	1	1	884.3 eV	0.5	22	10
	0.5	2			0.5	23	
3.679	0.5	3		536.3	0.5	24	
2.231	0.5	4	2	325.3	0.5	25	11
1.353	0.5	5		197.3	0.5	26	
0.821	0.5	6	3	119.7	0.5	27	12
0.498	0.25	7		72.6	0.5	28	
0.388	0.25	8	4	44.0	0.5	29	13
0.302	0.5	9		26.7	0.5	30	
0.183	0.5	10		16.2	0.5	31	14
0.111	0.5	11	5	9.82	0.5	32	
67.4 KeV	0.5	12		5.96	0.5	33	
40.9	0.47	13	6	3.61	0.5	34	16
25.5	0.25	14		2.19	0.5	35	
19.9	0.28	15	7	1.33	0.5	36	
15.0	0.5	16		.806	0.5	37	17
9.12	0.5	17		.489	0.5	38	
5.53	0.5	18	8	.296	0.5	39	18
3.35	0.167	19		.180	0.5	40	
2.84	0.166	20	9	.109	0.5	41	19
2.404	0.167	21		.066	1.30	42(Th)	
1.458				.018		20(Th)	

TABLE 2. k_{∞} VALUES FOR 0.5 in. DIAMETER BeO/UO_2 FUEL IN VARIOUS CONFIGURATIONS
AND AT 300°K

BeO/UO_2 MOL. RATIO	URANIUM ENRICHMENT	15 mil. SS CLAD 15v/o Na COOLING	15 mil. SS CLAD 25v/o GAS COOLING	15 Mil. Nb CLAD 15v/o Na COOLING	20 mil. Nb CLA 25v/o GAS COOL
20/1	93.2	1.800	1.815	1.709	1.527
	50.0	1.537	1.551	1.407	1.235
	30.0	1.328	1.340	1.162	1.027
50/1	93.2	1.680	1.693	1.555	1.308
	50.0	1.416	1.430	1.217	1.068
	30.0	1.240	1.254	.977	.928
100/1	93.2	1.617	1.632	1.421	1.209
	50.0	1.365	1.382	1.064	1.024
	30.0	1.196	1.217	.822	
150/1	93.2	1.592	1.610	1.326	
	50.0	1.335	1.357	.961	
	30.0	1.151	1.177	.719	

TABLE 3. COMPARISON OF k_{∞} VALUES FOR 0.25 in. AND 0.50 in. DIAM.BeO/UO₂ FUEL

BeO/UO ₂ MOL. RATIO	20/1		50/1		100/1		150/1	
FUEL PIN DIAM.(in.)	0.25	0.50	0.25	0.50	0.25	0.50	0.25	0.50
93.2% Enriched	1.755	1.800	1.615	1.680	1.513	1.617	1.449	1.592
50.0% Enriched	1.486	1.537	1.331	1.416	1.217	1.365	1.134	1.335
30.0% Enriched	1.267	1.328	1.126	1.240	1.005	1.196	0.907	1.151

TABLE 4A. NEGATIVE Δk_{∞} IN 0.5 in. DIAM. BeO/UO₂ FUEL DUE TO DOPPLER BROADENING

BeO/UO ₂ MOL. RATIO		20/1		50/1		100/1		150/1	
TEMP. RANGE		300°- 1200°K	1200°- 2100°K	300°- 1200°K	1200°- 2100°K	300°- 1200°K	1200°- 2100°K	300°- 1200°K	1200°- 2100°K
15 mil. SS Clad 15v/o Na Cooled	$\epsilon = 93.2$	.0056	.0016	.0087	.0028	.0092	.0028	.0074	.0022
	$\epsilon = 50.0$	.0275	.0129	.0438	.0214	.0493	.0232	.0445	.0202
	$\epsilon = 30.0$	.0472	.0235	.0652	.0332	.0656	.0323	.0564	.0269
15 mil. SS Clad 25v/o Gas Cooled	$\epsilon = 93.2$	.0070	.0015	.0087	.0026	.0093	.0029	.0077	.0023
	$\epsilon = 50.0$	.0269	.0125	.0437	.0213	.0500	.0235	.0456	.0206
	$\epsilon = 30.0$	.0466	.0231	.0656	.0333	.0669	.0330	.0581	.0277

TABLE 4B. 20 GROUP SPECTRA FOR BeO/UO₂ FUELED CONFIGURATIONS

		20/1	20/1	50/1	50/1	100/1	100/1	150/1	150/1	50/1
		E = 93.2	E = 30.0	E = 93.2	E = 30.0	E = 93.2	E = 30.0	E = 93.2	E = 30.0	E = 93.2
GP	E _L	SS/Na	SS/Na	SS/Na	SS/Na	SS/Na	SS/Na	SS/Na	SS/Na	SS/GAS
1	3.68 MeV	1.65	1.30	1.32	1.10	1.17	0.97	1.05	0.91	1.34
2	1.35	11.06	8.64	9.07	7.52	8.94	6.73	7.32	6.34	9.22
3	0.498	13.24	10.54	10.99	9.12	11.15	8.14	8.80	7.66	11.08
4	0.183	13.29	11.04	11.13	9.36	11.85	8.28	8.88	7.77	10.95
5	67.4 KeV	12.90	11.07	10.92	9.31	9.95	8.22	8.77	7.71	10.71
6	25.5	10.88	9.72	9.51	8.25	8.50	7.32	7.77	6.87	9.35
7	9.12	10.52	9.84	9.63	8.52	8.67	7.62	8.05	7.17	9.51
8	3.35	8.32	8.29	8.14	7.41	7.24	6.70	7.04	6.33	8.63
9	1.458	3.02	3.19	3.18	2.96	2.80	2.71	2.84	2.58	4.13
10	536 eV	6.74	7.89	7.79	7.58	7.41	7.02	7.29	6.70	7.56
11	197.3	4.50	6.60	6.53	6.99	6.56	6.73	6.85	6.50	6.27
12	72.6	2.38	4.97	4.85	6.12	5.36	6.25	6.16	6.16	4.67
13	26.7	1.07	3.38	3.35	5.07	4.10	5.61	5.36	5.70	3.19
14	9.82	0.30	1.78	1.74	3.64	2.55	4.60	4.13	4.92	1.64
15	3.61	0.07	0.78	0.78	2.32	1.42	3.50	2.92	4.01	0.74
16	1.33	0.03	0.45	0.51	1.70	1.04	2.86	2.46	3.42	0.48
17	0.49	0.014	0.32	0.35	1.44	0.77	2.58	2.08	3.15	0.32
18	0.18	0.002	0.15	0.14	0.99	0.33	2.07	1.42	2.67	0.13
19	0.066	1 x 10 ⁻⁴	0.03	0.02	0.46	0.10	1.33	0.65	1.92	0.02
20	Th.	2 x 10 ⁻⁶	3 x 10 ⁻³	1.5 x 10 ⁻³	0.12	0.0103	0.72	0.164	1.49	1.3 x 10 ⁻³

TABLE 5. k_{∞} VALUES FOR 0.5 in. DIAMETER (UC, ZrC) - GRAPHITE FUEL INVARIOUS CONFIGURATIONS

C/U ATOM RATIO	URANIUM ENRICHMENT	15 mil. SS CLAD 15v/o Na COOLING	15 mil. SS CLAD 25v/o GAS COOLING	15 mil. Nb CLAD 25v/o GAS COOLING	UNCLAD 25v/o GAS COOLING
50/1	93.2	1.588	1.597	1.235	1.658
	50.0	1.316	1.325		
	30.0	1.107	1.114		
75/1	93.2	1.504	1.514	1.050	1.584
	50.0	1.232	1.240		
	30.0	1.033	1.040		
100/1	93.2	1.449	1.458	.920	1.540
	50.0	1.177	1.186		
	30.0	0.985	0.993		
150/1	93.2	1.377	1.386		1.520
	50.0	1.106	1.116		
	30.0	.920	0.930		

TABLE 6. NEGATIVE Δk_{∞} IN 12.7 mm (UC, ZrC)-GRAPHITE FUEL DUE
TO DOPPLER BROADENING

C/U ATOM RATIO		50/1		75/1		100/1		150/1	
TEMP. RANGE		300- 1200°K	1200- 2100°K	300- 1200°K	1200- 2100°K	300- 1200°K	1200- 2100°K	300- 1200°K	1200- 2100°K
.381 mm SS CLAD 15v/o Na COOLED	$\epsilon = 93.3$	.0050	.0013	.0055	.0016	.0060	.0018	.0057	.0017
	$\epsilon = 50.0$	.0247	.0116	.0301	.0144	.0337	.0160		
	$\epsilon = 30.0$	.0415	.0205	.0471	.0233	.0496	.0244		

TABLE 7. 20 GROUP SPECTRA FOR ROVER FUEL CONFIGURATIONS

ROVER 15v/o NA 0.381 mm SS		50/1	50/1	100/1	100/1
GP	E _L	93.2	30.0	93.2	30.0
1	3.68 MeV	1.37	1.11	1.16	.97
2	1.35	8.57	6.88	7.28	6.09
3	0.498	14.09	11.51	12.20	10.28
4	0.183	14.20	12.00	12.40	10.64
5	67.4 KeV	13.02	11.31	11.56	10.06
6	25.5	11.07	9.57	10.10	8.93
7	9.12	10.66	9.99	10.07	9.09
8	3.35	8.14	8.03	8.06	7.47
9	1.458	2.59	2.67	2.67	2.53
10	536 eV	6.57	7.83	7.74	7.65
11	197.3	4.82	6.58	6.34	6.91
12	72.6	2.65	4.91	4.51	5.77
13	26.7	1.28	3.45	3.03	4.70
14	9.82	.39	1.88	1.50	3.28
15	3.61	.098	0.85	0.63	2.00
16	1.33	.044	0.49	0.39	1.38
17	0.49	.021	0.35	0.25	1.13
18	0.18	4×10^{-3}	0.17	0.093	0.73
19	0.066	2×10^{-3}	0.041	0.014	0.31
20	Th.	4×10^{-6}	4×10^{-3}	7×10^{-4}	0.065

TABLE 8. VALUES OF k_{∞} FOR UO_2 -X CERMETS FOR DIFFERENT MATRIX METALS, X.

MATRIX METAL, X		Cr	Mo	Nb
10v/o UO_2 -X 12.7 mm DIAM FUEL, UNCLAD, 25 v/o GAS COOLING	$\epsilon = 93.2$	1.713	1.069	0.991
	$\epsilon = 50.0$	1.373	0.718	0.654
	$\epsilon = 30.0$	1.089	0.491	0.443
12.5v/o UO_2 -X 12.7 mm DIAM FUEL UNCLAD, 25v/o GAS COOLED	$\epsilon = 93.2$	1.784	1.204	1.139
	$\epsilon = 50.0$	1.451	0.839	0.745
	$\epsilon = 30.0$	1.163	0.530	0.496
10v/o UO_2 -X 12.7 mm DIAM FUEL 0.381 mm SS CLAD 15v/o Na COOLING	$\epsilon = 93.2$	1.618		
	$\epsilon = 50.0$			
	$\epsilon = 30.0$			
10 v/o UO_2 -X 12.7 mm DIAM FUEL 0.381 mm Nb CLAD 15v/o Na COOLING	$\epsilon = 93.2$	1.438		
	$\epsilon = 50.0$			
	$\epsilon = 30.0$			

TABLE 9. MONTE CARLO k_{eff} CALCULATIONS FOR FACILITY WITH
AND WITHOUT HODOSCOPE SLOTS

		<u>WITHOUT SLOT</u>	<u>VOIDED SLOT</u>	<u>50% SS SLOT</u>
ROVER FUELED DRIVER C/v = 50/1 E = 93.2	k_{eff}	1.040	0.938	0.987
	Error(1 σ)	0.93%	0.55%	0.60%
	Δk	-	-0.102 $\pm$.015	-0.053 $\pm$.016
10% UO ₂ -Cr FUELED DRIVER E = 93.2	k_{eff}	1.240	1.140	1.190
	Error(1 σ)	0.24%	0.50%	0.64%
	Δk	-	-0.100 $\pm$.009	-0.051 $\pm$.011

TABLE 10. SINGLE REGION BeO/UO₂ DRIVER CHARACTERISTICS.

12.7 mm DIAM. FUEL, 0.381 mm SS CLAD, 15v/o SODIUM COOLANT

CASE	CORE HEIGHT	TEST REGION	BeO/UO ₂ RATIO	ENRICH. E	k _∞	k _{eff}	R _c	M _c	P	TEST MAX/MIN	FOM
1	3 Ft.	217E	20/1	93.2	1.800	1.10	27.5	115	43	1.17	4.1
2	"	"	"	50.0	1.537	"	36.2	123	54	1.26	5.6
3	"	"	"	30.0	1.328	"	57.4	212	92	1.36	6.9
4	"	"	"	"	"	1.15	75.2	383	151	1.34	6.5
5	"	217P	"	"	"	1.10	65.2	290		1.16	2.0
6	3 Ft.	217E	50/1	93.2	1.815	1.10	33.0	83	44	1.34	5.8
7	"	"	"	"	"	1.15	35.8	100	52	1.32	5.6
8	"	"	"	50.0	1.551	1.10	49.5	116	67	1.46	7.4
9	"	"	"	"	"	1.15	60.1	178	100	1.43	6.9
10	"	217P	"	"	"	1.10	56.9	151	297	1.22	2.2
11	4 Ft.	217E	"	"	"	"	45.3	130	75	1.40	7.7
12	"	"	"	"	"	1.15	52.4	179	100	1.57	7.3
13	3 Ft.	"	"	30.0	1.340	1.10	118.9	423	244	1.57	8.3
14	"	"	"	"	"	1.15	∞	-	-	-	-
15	4 Ft.	"	"	"	"	1.10	82.4	270	157	1.50	9.1
16	"	"	"	"	"	1.15	151.7	926	476	1.53	8.5
17	3 Ft.	217E	100/1	93.2	1.617	1.10	38.5	75	40	1.49	7.1
18	"	"	"	50.0	1.365	"	63.8	101		1.63	8.4
19	"	"	"	"	"	1.15	81.8	169	138	1.67	7.8
20	"	217P	"	"	"	1.10	72.6	133	364	1.37	2.7
21	4 Ft.	217E	"	"	"	"	55.1	97	85	1.62	8.9
22	"	"	"	"	"	1.15	66.4	110	147	1.66	8.3
23	3 Ft.	"	"	30.0	1.196	1.10	∞	-	-	-	-
24	3 Ft.	217E	150/1	93.2	1.592	1.10	42.2	52	48	1.60	7.7
25	"	"	"	50.0	1.335	"	76.1	99	106	1.73	8.9
26	"	217P	"	"	"	"	86.2	126	451	1.47	2.9
27	"	217E	"	30.0	1.151	"	∞	-	-	-	-

TABLE 11. COMPARISON OF CHARACTERISTICS FOR 6.4 mm AND 12.7 mm DIAM. FUELBeO/UO₂ 0.381 mm SS CLAD, 15v/o SODIUM COOLING, 3 FT. HIGH CORE

CASE	FUEL DIAM MM	TEST REGION	BeO/UO ₂ RATIO	ENRICH E	k _∞	k _{eff}	R _c	M _c	P	TEST MAX/MIN	FOM
28	6.4	217E	20/1	93.2	1.755	1.10	30.8	129	48	1.17	4.0
1	12.7	"	"	"	1.800	"	27.5	115	43	1.17	4.1
29	6.4	"	"	50.0	1.486	"	43.4	164	58	1.27	5.5
2	12.7	"	"	"	1.537	"	36.2	123	54	1.26	5.6
30	6.4	"	"	30.0	1.267	"	93.0	498	179	1.36	6.7
3	12.7	"	"	"	1.328	"	57.4	212	92	1.36	6.9

TABLE 12. SINGLE REGION BeO/UO₂ DRIVER CHARACTERISTICS

12.7 mm DIAM. FUEL, 0.381 mm SS CLAD, 25 v/o GAS COOLANT

CASE	CORE HEIGHT	TEST REGION	BeO/UO ₂ RATIO	ENRICH. E	k _∞	k _{eff}	R _c	M _c	P	TEST MAX/MIN	FOM
31	3 Ft.	217E	20/1	93.2	1.815	1.10	30.4	135	48	1.16	4.2
32	3 Ft.	217E	20/1	50.0	1.551	1.10	41.8	163	56	1.23	5.8
33	3 Ft.	217E	20/1	30.0	1.340	1.10	7.19	316	113	1.36	7.1
34	3 Ft.	217E	50/1	93.2	1.693	1.10	37.6	103	50	1.33	5.9
35	3 Ft.	217E	50/1	50.0	1.430	1.10	59.9	149	83	1.45	7.5
36	3 Ft.	217E	50/1	30.0	1.254	1.10	206.4	1145	589	1.56	8.4
37	4 Ft.	217E	50/1	30.0	1.254	1.10	103.0	369	196	1.55	9.3
38	3 Ft.	217E	100/1	50.0	1.382	1.10	79.5	145	111	1.62	8.6
39	3 Ft.	217E	150/1	50.0	1.357	1.10	97.8	144	140	1.72	10.2
0.508 mm NIOBIUM CLADDING											
40	3 Ft.	217E	20/1	93.2	1.527	1.10	39.7	260	71	1.13	4.5
41	3 Ft.	217E	20/1	50.0	1.235	1.10	142.6	2012	440	1.21	6.2
42	3 Ft.	217E	50/1	93.2	1.308	1.10	82.1	523	166	1.27	6.4
43	3 Ft.	217E	100/1	93.2	1.209	1.10	∞	-	-	-	-

TABLE 13. PERFORMANCE CHARACTERISTICS OF SINGLE REGION (UC, ZrC)-C DRIVEN SYSTEMS

CASE	CORE HEIGHT	TEST REGION	C/U RATIO	ENRICH. E	CLADDING	COOLANT	k_{∞}	k_{eff}	R_c	M_c	P	TEST MAX/MIN	FOM
44	3 Ft.	217E	50/1	93.2	None	25v/o GAS	1.657	1.10	54.2	257	88	1.15	8.5
44P	3 Ft.	217P	50/1	93.2	None	25v/o GAS	1.657	1.10	58.9	308	376	1.07	2.2
45	3 Ft.	217E	50/1	50.0	None	25v/o GAS	1.397	1.10	135.8	550	297	1.24	11.3
46	3 Ft.	217E	100/1	93.2	None	25v/o GAS	1.537	1.10	84.2	325	135	1.26	11.4
47	3 Ft.	217E	500/1	93.2	None	25v/o GAS	1.617	1.10	114.5	167	164	1.61	14.5
48	3 Ft.	217E	1000/1	93.2	None	25b/o GAS	1.695	1.10	120.2	92	162	1.77	15.4
49	3 Ft.	217E	50/1	93.2	0.381 mm SS	15v/o Na	1.588	1.10	53.2	231	83	1.19	7.9
50	3 Ft.	217E	50/1	50.0	0.381 mm SS	15v/o Na	1.325	1.10	178.2	890	467	1.29	10.3
51	3 Ft.	217E	50/1	30.0	0.381 mm SS	15v/o Na	1.107	1.10	∞	-	-	-	-
52A	3 Ft.	217E	75/1	93.2	0.381 mm SS	15v/a Na	1.504	1.10	69.5	276	109	1.23	9.3
52B	4 Ft.	217E	75/1	93.2	0.381 mm SS	15v/o Na	1.504	1.10	57.4	246	95	1.23	9.7
53	3 Ft.	217E	75/1	50.0	0.381 mm SS	15v/o Na	1.232	1.10	∞	-	-	-	-
54	3 Ft.	217E	100/1	93.2	0.381 mm SS	15v/o Na	1.449	1.10	86.2	312	129	1.28	10.5
55	3 Ft.	217E	50/1	93.2	0.508 mm SS	25v/o GAS	1.580	1.10	72.9	481	123	1.16	8.1
55H*	3 Ft.	217E	50/1	93.2	0.508 mm SS	25v/o GAS	-	1.10	70.9	454	-	1.17	9.2 [†]

* HANSEN and ROACH 16 GP SET

[†] FISSION SOURCE RATIO

TABLE 14. PERFORMANCE CHARACTERISTICS OF SINGLE REGION UO_2 -Cr DRIVER SYSTEMS.

CASE	CORE HEIGHT	TEST REGION	v/o UO_2	E	CLADDING	COOLANT	k_{∞}	k_{eff}	R_c	M_c	P	TEST MAX/MIN	FOM
56	3 Ft.	317E	10.0	93.2	0.381 mm Nb	15v/o SODIUM	1.438	1.1	173.8	6436	535	1.02	9.5
57	"	"	12.5	"	"	"	1.543	"	82.7	1422	182	1.03	7.6
58	"	"	7.5	"	0.381 mm SS	"	1.515	"	225.1	6500	711	1.01	11.7
59	"	"	10.0	"	"	"	1.618	"	91.9	1405	189	1.01	9.1
60	"	"	12.5	"	"	"	1.706	"	66.8	935	133	1.01	7.5
61	"	"	7.5	"	"	15v/o GAS	1.542	"	∞	-	-	-	-
62	"	"	10.0	"	"	"	1.647	"	108.1	2015	245	1.02	9.4
63	"	"	12.5	"	"	"	1.735	"	73.7	1137	156	1.03	7.6
64	"	"	12.5	"	"	15v/o SODIUM	1.706	1.2	94.72	1879	252	1.00	7.3
65	4 Ft.	"	7.5	"	"	"	1.515	1.1	103.9	1818	229	1.00	11.8
66	"	"	10.0	"	"	"	1.618	"	68.6	1290	140	1.010	9.3

TABLE 15. PERFORMANCE CHARACTERISTICS OF (1+1) CONVERTER/DRIVER SYSTEMS

CASE	TEST REGION	CONVERTER REGION		DRIVER REGION		DTF RESULTS				
		CONFIGURATION	THICKNESS	CONFIGURATION		R_c	P	TEST MAX/MIN	FOM (CONV.)	FOM (DRIV.)
67	217E	$5^{V/o} \text{UO}_2$ - Cr E = 50% 0.381 mm SS CLAD, $15^{V/o}$ Na	7 cm	100/1 BeO/ UO_2 E = 50% 0.381 mm SS CLAD, $15^{V/o}$ Na		102.4	265	1.28	8.2	7.6
68	217E	$5^{V/o} \text{UO}_2$ - Cr E = 50% 0.381 mm SS CLAD, $15^{V/o}$ Na	10 cm	100/1 BeO/ UO_2 E = 50% 0.381 mm SS CLAD, $15^{V/o}$ Na		107.1	325	1.22	8.1	6.8
69	217E	$5^{V/o} \text{UO}_2$ - Cr E = 50% 0.381 mm SS CLAD, $15^{V/o}$ Na	13 cm	100/1 BeO/ UO_2 E = 50% 0.381 mm SS CLAD, $15^{V/o}$ Na		115.2	415	1.17	7.9	6.1
70	217E	$5^{V/o} \text{UO}_2$ - Cr E = 93.2% 0.381 mm SS CLAD, $15^{V/o}$ Na	10 cm	50/1 ROVER E = 93.2 0.381 mm SS CLAD, $15^{V/o}$ Na		91.2	175	1.12	7.7	7.8
71	217E	$7\frac{1}{2}^{V/o} \text{UO}_2$ - Cr E = 93.2% 0.381 mm SS CLAD, $15^{V/o}$ Na	15 cm	50/1 ROVER E = 93.2 0.381 mm SS CLAD, $15^{V/o}$ Na		62.7	136	1.03	7.9	7.9

TABLE 16. PERFORMANCE CHARACTERISTICS OF TWO CRUDELY OPTIMIZED MULTI-CONVERTER/DRIVER SYSTEMS

CASE		CONV. 1	CONV. 2	CONV. 3	DRIVER 1	DRIVER 2	DRIVER 3	DRIVER 4	DRIVER 5	R _c	P	TEST MAX/MIN
72	Zone Thickness cm.	8.0	5.0	2.0	5.24	23.12	12.85	12.85	10.27	93.52	-	-
	Converter UO ₂ Enrichment %	93.2	88.9	83.3	-	-	-	-	-	-	-	-
	Driver UO ₂ Enrichment %	-	-	-	67.7	61.5	69.3	82.1	93.2	-	-	-
	Regional FOMs	12.8	11.2	10.7	8.7	8.7	8.8	9.3	10.9	-	-	1.064
	Regional Powers, MW	6.9	6.5	3.3	11.4	74.7	54.5	60.2	49.8	-	267	-

Case 72 Specifications: 3 Ft. High Core, Converter - 5v/o UO₂-Cr Driver - 50/1 ROVER. Both 0.381 mm SS CLAD, 15v/o Na Coolant

CASE		CONV. 1	CONV. 2	CONV. 3	DRIVER 1	DRIVER 2	DRIVER 3	DRIVER 4	DRIVER 5	R _c	P	TEST MAX/MIN
73	Zone Thickness cm.	5.0	5.0	5.0	8.4	15.4	8.8	8.8	10.3	81.0	-	-
	Converter UO ₂ Volume %	5	4	3	-	-	-	-	-	-	-	-
	Driver UO ₂ Enrichment %	-	-	-	60	40	40	50	70	-	-	-
	Regional FOMs	10.1	9.7	9.7	8.7	7.9	8.0	7.8	7.7	-	-	1.077
	Regional Powers, MW	8.0	5.6	4.9	7.4	55	28	32	59	-	200	-

Case 73 Specifications: 3 Ft. High Core, Converter - 93%E UO₂-Cr, Driver 50/1 BeO/UO₂, Both 0.381 mm SS CLAD, 15v/o Na

TABLE 17. COMPARISON OF PERFORMANCE CHARACTERISTICS OF 19, 37, AND 217 TEST PIN FACILITIES

CASE	CORE HEIGHT	TEST REGION	DRIVER FUEL	ENRICH. E	CLADDING	COOLANT	R _c	P	TEST MAX/MIN	FOM
8	3 Ft.	217E	50/1 BeO/UO ₂	50.0	0.381 mm SS	15v/o Na	49.5	67	1.46	7.4
74	3 Ft.	37E	50/1 BeO/UO ₂	50.0	0.381 mm SS	15v/o Na	46.9	41	1.22	10.2
75	3 Ft.	37P	50/1 BeO/UO ₂	50.0	0.381 mm SS	15v/o Na	48.5	156	1.10	3.1
76	4 Ft.	37E	50/1 BeO/UO ₂	50.0	0.381 mm SS	15v/o Na	42.4	48	1.21	10.4
77	4 Ft.	37P	50/1 BeO/UO ₂	50.0	0.381 mm SS	15v/o Na	44.1	175	1.10	3.2
78	3 Ft.	19E	50/1 BeO/UO ₂	50.0	0.381 mm SS	15v/o Na	47.2	38	1.17	11.8
79	3 Ft.	19P	50/1 BeO/UO ₂	50.0	0.381 mm SS	15v/o Na	48.2	141	1.08	3.4
80	4 Ft.	19E	50/1 BeO/UO ₂	30.0	0.381 mm SS	15v/o Na	88.6	89	1.23	14.7
81	4 Ft.	19P	50/1 BeO/UO ₂	30.0	0.381 mm SS	15v/o Na	90.8	330	1.14	4.5
82	4 Ft.	19E	7½v/o UO ₂ -Cr	93.2	0.381 mm SS	15v/o Na	107.8	270	1.01	11.7
83	4 Ft.	19P	7½v/o UO ₂ -Cr	93.2	0.381 mm SS	15v/o Na	108.5	1150	1.01	2.7
84	3 Ft.	37P	50/1 ROVER	50	0.381 mm SS	15v/c Na	189.8	1740	1.04	3.6
85	3 Ft.	19E	50/1 ROVER	50	0.381 mm SS	15v/o Na	186.8	398	1.12	14.5
86	3 Ft.	19P	50/1 ROVER	50	0.381 mm SS	15v/o Na	189.6	1640	1.03	3.9
87	4 Ft.	19P	50/1 ROVER	50	0.381 mm SS	15v/o Na	105.3	718	1.03	3.9

Fig. 1. k_{∞} Values for BeO/UO₂ 12-7 mm Diam. Fuel,
15% Sodium Cooled, 0.381 mm SS or Nb Clad

Fig. 2. Negative Δk_{∞} due to Doppler Temperature Broadening
 BeO/UO₂ Fuel 12.7 mm Diam., .381 mm SS Clad, 15%
 Sodium Cooling

Fig. 3. Negative Δk_{∞} due to Doppler Temperature Broadening
 BeO/UO₂ Fuel 12.7 mm Diam., 0.381 mm SS Clad, 25%
 Gas Cooling

Fig. 4. Values of k_{∞} For (UC, Zr C) - Graphite Fuel 12.7 mm Diam., .381 mm SS or Nb Clad, 15% Sodium or 25% Gas Cooling (Unclad)

Fig. 5. Values of Δk_{∞} due to Doppler Temperature Broadening (UC, Zr C) - Graphite 12.7 mm Diam., .381 mm SS Clad, 15% Sodium Cooling

Fig. 6. Performance Characteristics of Single Driver BeO/UO₂ Systems 12.7 mm Fuel Diam., 0.381 mm SS Clad, 15% Sodium Cooled, 217E Test

FIG. 7. REACTIVITY EFFECTS OF VOIDS AND FUEL RELOCATION IN CENTRAL TEST REGION

Change in $k_{eff} (\Delta k)$	A	B	C	D	E	F
	0.0 (reference)	-0.027	-0.032	0.0 ± 0.001	+0.023	-0.042

3. Thermal - Hydraulic Analysis

To date, studies in this area have been concentrated in the following broad categories:

- development of analytical and calculational models and methods to be used both in preliminary, scoping parameter surveys and in detailed design studies.
- broad parameter surveys involving wide ranges of fuel rod designs, coolant options, fuel specific power levels (steady state) and burst energy depositions.

An area which interrelates directly with the fuel thermal operating limits is the mechanical stability of the fuel over the desired range of operating power densities and burst energy depositions. Since this latter fuel characteristic will ultimately be investigated during a fuels development program, thermal operating limits can only be estimated at this stage of the design study.

The most important result of the thermal analyses described herein may be summarized as:

- Liquid metal cooling is significantly superior to gas cooling for both steady state and burst operating conditions. This statement holds over the entire range of fuel pin designs, steady state power levels and burst energy depositions considered.

a. Steady State. The calculations described below were of a survey nature. While providing direction for the generation of the neutron cross section sets described in Section 2.2, they also quantified the effects of, 1) liquid metal vs. gas coolant and 2) differing fuel (pellet) dimensions.

The two-dimensional heat transfer code AYER was employed to determine steady state temperatures in this parameter study.¹ Fuel pins of varying dimensions, each surrounded by a (flowing) coolant zone, were investigated in R-Z cylindrical geometry. The axial direction was divided into 80 equal height intervals over the 91.44 cm (3') length, and the power generation profile over this length was input as a truncated cosine:

$$P(Z) = P_{\text{mid-plane}} [\cos((45.77-Z)0.0285)].$$

This distribution is representative of an FFTF-like system; i.e., a fast system; ~ 1 m core height; axial reflectors. The radial direction was divided into 10 intervals as shown in Fig. 1. Not shown in this figure is a nominal 1" fuel pellet geometry. Preliminary calculations for this fuel size resulted in prohibitively high centerline fuel temperatures even for sodium cooling, and further analyses for this fuel pin were discontinued.

In order to save computing time during these scoping calculations, materials properties (density, heat capacity, thermal conductivity) were input as constants although AYER accepts any user specified temperature dependence. Uncertainties in the resulting temperatures due to this effect alone should be generally less than 5%.

Materials Properties

Steady state temperatures for systems with flowing coolant are dependent on the thermal conductivities of the fuel rod components (fuel, gap, clad) and are dependent on the density, heat capacity and heat transfer coefficient of the coolant. The fuel-clad gap and clad thermal conductivities employed in this study were: 0.024 and 0.20 W/cm-K respectively. The gap conductance is representative of measured values for fast reactor fuel pins and the clad conductivity is typical of a metallic clad (i.e., Nb, ss). The fuel conductivities were parameterized as described in the following section.

Coolant properties were of two classes, (corresponding to gases or liquid metals) depending on the heat transfer coefficient regime.

Gas coolant: CO_2 at 23 m/s

$P = 80 \text{ atm}$; $\rho = 0.051 \text{ g/cm}^3$; $C_p = 1.03 \text{ J/g-K}$

Similar heat removal and transport may be realized with other gaseous coolants, e.g., helium, hydrogen. Hand

calculations for the above properties and previously described geometries indicated a maximum heat transfer coefficient:

$$h \approx 1.0 \text{ W/cm}^2\text{-K.}$$

If clad surface roughening were employed perhaps another factor of 2 increase in the heat transfer coefficient could be gained. However, as will be seen later, peak fuel and clad temperatures would be lowered only slightly by this measure.

Liquid metal coolant: Na at 10 m/s

$$P = 1 \text{ atm; } \rho = 0.85 \text{ g/cm}^3; C_p = 1.25 \text{ J/g-K}$$

Similar hand calculations indicated a heat transfer coefficient for sodium coolant of:

$$h \approx 10.0 \text{ W/cm}^2.$$

For liquid metal coolants, fuel and clad temperatures are very insensitive to factor of two changes about this nominal value of 10.0.

Parameter Variations

Fuel thermal conductivities were varied over a range which spans that of the three primary candidate STF fuels, viz. BeO-UO_2 , $(\text{UC-ZrC})\text{C}$ and Cr-UO_2 :

$$0.20 \leq k \leq 0.60 \text{ W/cm}^2\text{-K}$$

Clad-to-coolant heat transfer coefficients were varied over the ranges:

Gas Coolant

$$0.10 \leq h \leq 1.0 \text{ W/cm}^2\text{-K}$$

Sodium Coolant

$$3.0 \leq h \leq 10.0 \text{ W/cm}^2\text{-K.}$$

Peak driver fuel specific powers were varied as:

$$200 \leq P_{\text{mid-plane}} \leq 800 \text{ W/cm}^3\text{-fuel}$$

Results

The maximum fuel and clad and temperatures for the three fuel pin geometries (Fig. 1) are plotted as functions of (1) the fuel specific power at the axial midplane and (2) the clad-to-coolant heat transfer coefficient, in Figs. 2 through 7. Coolant outlet temperatures are given in Table I. These results are all based on a fuel thermal conductivity of 0.30 W/cm-k. Over the range of fuel thermal conductivities investigated,

$$0.20 \leq k \leq 0.60 \text{ W/cm-K,}$$

clad temperatures varied by less than 1% at constant specific power and heat transfer coefficient. Maximum fuel temperatures, however, did show a non-negligible temperature dependence on the thermal conductivity.

Figures 8 and 9 show this dependence of maximum fuel temperature on thermal conductivity for gas cooled and sodium cooled fuel pins respectively for the three fuel diameters

considered. These are also "fine structure" effects due to varying heat transfer coefficients and specific powers. The latter, however, are of second order and well within basic uncertainties in the calculations, such as temperature independent materials properties and the coarseness of the radial mesh.

Coolant pressure drop analysis

Scoping calculations were made to insure that the coolant velocities and flow areas and the pin dimensions used in the heat transfer calculations were consistent with reasonable coolant pressure drops.

Neglecting inlet, exit, acceleration, and other pressure drops, which should be relatively minor, the pressure drop for 2 meter long fuel pins was calculated from the formula:

$$\Delta P = 4f \frac{L}{D_H} \frac{\rho V^2}{2}$$

with $f \approx .046 Re^{-.2} \approx .005$

For the geometries and coolant densities and velocities used in the heat transfer calculations the calculated coolant pressure drops are given in Table II. The results indicate that pump specifications, etc. should be within the realm of current technology.

TABLE II
COOLANT PRESSURE DROP

Fuel Pin Diameter mm	Flow Channel Hydraulic Diameter mm	Pressure Drop CO ₂ atm	Na atm
20.5	1.71	0.66	2.5
14.1	1.18	1.0	3.9
7.9	0.66	2.1	7.8
GCFR		~3.3	
LMFBR			~8.1

Conclusions

The prototypicality of various FBR accident simulations would certainly be enhanced by an unlimited steady state capability at test fuel energy densities ≈ 12.5 KW/ft. Based on consideration of steady state temperatures only, the following points are significant.

- 0 Sodium cooling will permit the use of stainless steel or refractory (e.g., Nb) cladding.
- 0 Gas cooled fuel pins will require a high temperature clad unless the driver fuel radial dimension and specific power are restricted.

The calculations and results described herein are for fuel rods with a gas filled fuel-clad gap. It is possible that either mechanical (i.e., fuel cracking) or thermal considerations (i.e., optimizing the fuel-clad gap conductance) may suggest another fill material in the gap. This is discussed in more detail in the following section.

Table I Coolant Outlet Temperatures

Fuel Specific Power, W/cm ³	Temperature, K	
	Gas Cooled	Liquid Metal Cooled
6.4 mm Fuel Diameter		
200	540	325
400	780	355
600	1025	380
800	1270	410
12.7 mm Fuel Diameter		
200	590	330
400	890	365
600	1180	400
800	1480	440
19 mm Fuel Diameter		
200	615	335
400	935	370
600	1250	410
800	1580	450

Nominal 1/2" O.D. Fuel Pellet

Fig. 1. AYER radial mesh.

Fig. 2. Maximum fuel temperatures for 6.4 mm fuel.

Fig. 3. Maximum clad temperatures for 6.4 mm fuel.

Fig. 4. Maximum fuel temperatures for 12.6 mm fuel

Fig. 5. Maximum clad temperatures for 12.6 mm fuel.

Fig. 6. Maximum fuel temperatures for 19 mm fuel.

Fig. 7. Maximum clad temperature for 19 mm fuel.

Fig. 8. Relative maximum fuel temperature for gas-cooled pins.

Fig. 9. Relative maximum fuel temperature for liquid-metal-cooled pins.

b. Steady State Plus Burst. For this mode of reactor operation, scoping analyses of the influences of fuel pellet size and gas or liquid metal cooling were presented in B1 (Core Materials Selection). These were based on the steady state calculations described in the previous section and the heat capacity data published in ref. 2. In this section are described both theoretical and calculational models which may be used in optimizing the thermal-hydraulic core design. Finally, these models are applied to a specific fuel pin geometry and results of analyses are presented.

Optimum Gap Conductance Model

Although not unique, a model is described which enables a determination of that gap conductance which will balance the constraints due to the fuel and clad limiting temperatures so that one does not occur significantly before the other (for a specific combination of steady state power and burst energy deposition). Of course, any facility design will be required to provide a range of operating conditions. Nevertheless, this optimization model is valuable for scoping the range of "optimum" gap conductances corresponding to a broad band of operating conditions and indicating trends in this regard.

After more basic core design decisions such as fuel pellet diameter and coolant type (gas or liquid metal) have been made, and upper limits for the fuel and clad operating temperatures have been set, an optimum conductance, h_g , for the fuel-clad gap can be determined such that:

$$\frac{T_f}{T_{ff}} = \eta \frac{T_c'}{T_{cf}} \quad (1)$$

where:

T_f is the maximum steady state fuel temperature.

T_{ff} is the fuel failure temperature, or other limiting temperature.

η is a constant which may be used to preferentially weight the fuel or clad;

$\eta = 1$ was used in the analyses which follow.

T_c' is the maximum clad temperature; the prime denotes that this occurs post-burst.

T_{cf} is the clad failure temperature, or other limiting temperature.

The maximum clad temperature occurs when the rate of change, dT_c/dt , is zero. A heat balance on the clad at that time yields the relation:

$$h_c r_c (T_c' - T_x) \simeq h_g r_f (T'_{fs} - T_c'). \quad (2)$$

where:

h_c is the coolant heat transfer coefficient.

r_c is the clad outer radius.

h_g is the fuel-clad gap conductance

r_f is the fuel outer radius

T_x is the local coolant temperature

T'_{fs} is the fuel surface temperature, assumed constant at the maximum value reached during the transient.

The approximate equality of the left and right hand sides of Eq. 2 is due to the assumed constant maximum transient fuel surface temperature. Actually this is a small effect for the short times involved (until T'_c is reached).

The fuel temperatures T_f and T'_{fs} can be calculated for a specified fuel rod axial location, x , from the equations:

$$T_f = T_x + \frac{r_f^2 q_x'''}{2} \left[\frac{1}{2\lambda} + \frac{1}{h_c r_c} + \frac{1}{h_g r_f} \right] \quad (3)$$

and

$$T'_{fs} = T_f + \frac{\Lambda E_x}{C_f} - \frac{r_f^2 q_x'''}{4\lambda} \quad (4)$$

where

λ is the fuel thermal conductivity

C_f is the fuel specific heat

ΛE_x is local specific energy deposition in the burst; the same axial distribution relative to the peak (midplane) value as for q_x^{lll} is assumed.

q_x''' is the local steady state power density, given by

$$q_x'' = q_0''' \sin [x/L (\pi - 2M) + M] \quad (5)$$

x is the axial location of interest

L is the heated (active) length

q_0''' is the midplane (maximum) power density

M is a constant for approximating a chopped sine power distribution.

The clad thermal conductance is much higher than h_c or h_g and thus has been omitted from Eqs. (3) and (4).

For constant coolant flow and specific heat (neglecting mixing due to crossflow) the local coolant temperature can be found by integration using Eq. 5.

$$T_x = T_{in} + \frac{q_o''' \pi r_f^2 L}{m c_p (\pi - 2M)} \left\{ \cos M - \cos \left[\frac{x}{L} (\pi - 2M) + M \right] \right\} \quad (6)$$

where:

T_{in} is the coolant inlet temperature

m is the coolant mass flow per fuel rod.

c_p is the coolant specific heat.

After combining equations (1), (2), (3), and (4) and re-arranging, there result 5 non-dimensional parameters in a quadratic equation for the non-dimensional optimum gap conductance.

$$A^2 + A[(B+1)(1-\tau) + (1+D)] + [(B+1)(1-\tau) + D - C] = 0 \quad (7)$$

where:

$$A = \text{non-dimensional gap conductance} = \frac{h_c r_c}{h_g r_f}$$

$$B = \text{non-dimensional steady-state power} = \frac{2 T_x h_c r_c}{r_f^2 q_o'''}$$

$$\tau = \text{failure temperature ratio} = \eta \frac{T_{ff}}{T_{cf}}$$

C = non-dimensional energy spike

$$= \frac{\Lambda}{c_p r_f^2 q'''} = \frac{\Lambda}{c_p r_f^2 q_o'''} = \frac{2E h_c r_c \tau}{c_p r_f^2 q_o'''}$$

$$D = \text{Biot modulus} = \frac{h_c r_c}{2\lambda}$$

By using Eqs. (6) and (7) the optimum gap conductance can be found for any axial location along the fuel rod.

Axial locations that may be of interest include the core midplane, and where peak, steady state fuel and clad temperatures occur. Following the burst energy

deposition, the peak transient temperatures will be shifted toward the core midplane. The axial location of peak fuel temperature can be found using Eqs. (3), (5) and (6), differentiating T_f with respect to X , and setting the result to zero. The result, after some re-arranging is:

$$x \Big|_{\max T_f} = \frac{L}{\pi - 2M} \left\{ \tan^{-1} \left[- \frac{mc_p (\pi - 2M)}{2\pi L} \left(\frac{1}{2\lambda} \right. \right. \right. \\ \left. \left. \left. + \frac{1}{h_g r_f} + \frac{1}{h_c r_c} \right) \right] - M \right\}.$$

The location of the maximum clad temperature is found in a similar manner. Because the location shifts during the transient, maximum temperatures calculated for the above axial locations will be slightly in error, but not of sufficient magnitude to influence the applicability of the model.

Thus for specified fuel rod materials and dimensions, coolant flow rates and burst and steady state energies and powers, the previous equations can be used to determine: 1) h_g ; 2) T_f ; 3) T'_f ; 4) T'_c . This set of closed-form equations has been coded and has facilitated economical, useful parameters studies which have provided the bases and guidance for more detailed and complete numerical analyses.

Finite-Element Fuel Rod Model

The optimum gap model described above contains certain assumptions and limitations which can be removed by a numerical (computer) analysis of the heat transfer problem. Among them were: constant materials properties, constant fuel surface temperature post-burst, determination of axial location of temperature maxima and non-accounting for transient fuel expansion effects on the gas conductance, h_g . To assess these effects, a fine-mesh finite-element fuel rod model was constructed and used in the AYER computer code.

The model is one-dimensional (axisymmetric cylinder), consisting of 50 finite elements in the fuel, one in the gap and two in the clad. The power density and coolant temperature are specified by Equations (5) and (6). Temperature-dependent, material properties for fuels and clads were taken from reference 2. The model includes the effect of

the fuel rod's thermal expansion on the gap conductance. Included also, but not used in the cases studied, are latent heat effects in the fuel rod.

Once an "optimum" value for h_g is obtained from the analytical model prediction (for a given set of power densities, properties, coolant boundary conditions, etc.), the corresponding parameters, including h_g , are then input into the AYER model. Results of these calculations represent improved estimates of fuel and clad temperatures, both steady state and transient.

Results

Liquid Metal Cooling

For the purpose of performing parameter variations limited to thermal characteristics, the following materials and geometry were selected:

fuel and fuel dimension	= BeO, 12.7 mm o.d.
fuel-clad gap	= 0.25 mm (10 mils)
clad and clad dimension	= ss, 0.381 mm (15 mils) thick
coolant volume	= 15% of the cell volume

In order to obtain estimates of coolant flow velocities and heat transfer coefficients for the configurations of interest, scoping coolant pressure drop calculations were performed for the above described geometry. Fig. 1 shows the heat transfer coefficient, h_c , and friction pressure drop, Δp , for sodium flowing in-line through arrays of 14 mm o.d. rods with a

15% coolant volume fraction. The friction pressure drop was calculated as in (a. Steady State); however, this ΔP is only for a 91.44 (3') length. The heat transfer coefficient was determined from reference 3, neglecting the beneficial effect of circumferential heat conduction in the clad. Thus higher h_c 's should be realized in practice, but flow velocities greater than about 15 m/s and heat transfer coefficients above $50 \text{ KW/m}^2\text{-K}$ will be difficult to attain.

The most probable estimates for sodium (with 15% coolant volume) are:

$$h_c \approx 40 \text{ KW/m}^2\text{-K}$$

and $P \approx 2.2 \text{ atm (33 psi)}$

For these conditions the corresponding steady state coolant temperature rise is given in Table I as a function of core midplane power density, q_o''' .

Figures 2 through 5 are based on the "optimum gap conductance model". Figs. 2 and 3 show the variation of the optimum gap conductance with q_o''' and E respectively. These two plots indicate the range of desirable gap conductances for various reactor operating conditions.

Figures 4 and 5 show the resulting fuel and clad temperatures for the same parameter variations as in the previous two plots. Transient fuel temperatures will obviously become

excessive before the clad temperature limits are approached. Transient clad temperatures do not exceed 800 K, a value certainly consistent with stainless steel clad; and, steady state fuel temperatures are less than 1100 K over the entire range of operating conditions.

Figures 4 through 6 result from detailed "finite element fuel rod model" calculations, however, based on the optimum gap conductances taken from Fig. 3. The effect of fuel (BeO) thermal expansion on the gap conductance is shown in Fig. 6. This effect was modeled by assuming that as the gap closes the thermal conductivity remains constant. Since

$$h_g = \lambda / \delta,$$

where λ is the gap thermal conductivity
and δ is the gap thickness,

the gap conductance, h_g , increases with decreasing gap thickness. Although actual fuel rod conductances do not follow this behavior exactly, this model is probably quite conservative⁴.

Figure 7 shows the influence which the gap conductance model has on the calculated transient clad temperatures. The upper curve is based on the variable gap conductance model (Fig. 6) and the lower curve is based on a constant gap conductance. The peak clad temperatures differ by only about 70 K, even for these two extreme models.

Figure 8 shows transient clad temperatures for two sets of conditions, each at two axial locations. The combinations of E and q_0'' selected for this comparison represent substantial operating conditions for the driver fuel rods. As expected, the midplane ($x/L = 0.5$) clad temperatures are cooler than those downstream at the locations of maximum steady state temperatures ($x/L = 0.77$ and 0.81 respectively). The magnitude of this temperature difference is unimportant, however, being less than 100 K in both cases. The thermal conductances for the assembled gap were taken from Fig. 3. For the same operating conditions, clad temperatures are higher in this figure than those in Fig. 5, due to local coolant temperatures being slightly hotter than the 500 K used in the parametric studies leading to Figs. 2-5. This effect becomes more pronounced as the steady state power is increased.

Figure 9 duplicates case A in Fig. 8 but with an (unrealistically) reduced heat transfer coefficient, h_c , for the purpose of estimating the influence of such a reduction. The corresponding flow velocity and gap conductance have also been reduced in accordance with Figs. 1 and 2 respectively. The trends were as expected: clad temperatures are higher; the transient is slower; the temperature difference between the two axial locations is increased, due to the larger coolant bulk temperature rise.

In summary, using a liquid metal (e.g, sodium) coolant with reasonable flow velocities (~ 8 m/s) and pressure drops (~ 2.2 atm), the driver fuel rods remain below probable upper limit operating conditions for the entire range of steady state power densities and burst energy depositions considered. The results presented were calculated for a BeO based fuel, but both Cr and graphite based fuels, having similar thermal properties, should respond similarly to BeO. Calculations for Cr and graphite based fuels, paralleling those described herein for BeO fuel, are underway.

Gas Cooling

The identical fuel rod design was used in the analyses described in this section as in the previous discussion concerning liquid metal cooling. One difference in the geometry, external to the fuel rod was:

Coolant volume = 25% of the cell volume.

Paralleling the analyses for liquid metal cooling, Fig. 10 shows the dependence of coolant pressure drop and heat transfer coefficient on gas velocity. In Table I are presented the coolant temperature rise across the core for various fuel midplane specific power levels. Comparison of Figs. 1 and 10 and the temperature rises in Table I indicates that even for a $\sim 70\%$ larger coolant flow area, cooling via helium is significantly less effective.

Figures 11 through 14 are based on the "optimum gap conductance model". Figs. 11 and 12 show the variation of the optimum gap conductance with q_0''' and $\hat{E}$ respectively. These results indicate a significantly lower optimum gap conductance range for gas cooled rods than for liquid metal cooled. Figs. 13 and 14 show the resulting fuel and clad temperatures for the same parameter variations as in the previous two plots. These results indicate that (1) operating capability is definitely more limited for gas cooling, and that (2) either clad transient or fuel transient temperatures may limit the range of operating power levels and burst energy densities. Recall that for liquid metal cooling fuel transient temperatures (figs. 4 and 5) always became limiting before excessive clad temperatures were reached.

Results of an AYER calculation ("finite element fuel rod model") are given in Fig. 15. In comparison with liquid metal cooling, qualitatively one observes that the transient is stretched out in time and the clad temperature rise from steady state is greater. Both of these effects are due to the reduced heat transfer coefficients, h_g and h_c .

A quantitative comparison for the same power density and burst conditions shows that the maximum transient clad temperature is 1035 K versus 760 K for sodium cooling; the maximum steady-state fuel temperature is 947 K versus 760 K; the maximum transient fuel temperature is 1800 K versus 1625 K.

Discussion

The fuel-clad gap conductance is an important parameter in that it can be optimized with almost no feedback on the neutronics of the reactor, for suitable choices of materials in the gap.

Although it is not difficult to calculate fuel rod temperatures given a means for predicting the gap conductance, to make such predictions accurately is difficult. In most practical applications the gap conductance is measured experimentally.^{4,5,6} In particular, some of the conclusions from ref.⁴ are interesting and pertinent to these design studies.

- For initial fuel-clad radial clearances greater than 0.1 mm (4 mils) the conductance is essentially independent of the gap width.
- For smaller clearances the conductance increases with decreasing gap width. Where the gap is expected to be closed at power, the dependence on the initial cold gap is very strong and h_g 's as high as 30 KW/m² may occur.
- A reduction in the filling gas thermal conductivity gives rise to a disproportionately small reduction in h_g .

Available experimental data are not directly applicable to determining gap conductances for currently contemplated fuel pin designs. Thus, fuel rod operating conditions

must allow some margin for inaccuracies in the calculated temperatures. This uncertainty will be reduced by a fuels development program encompassing the measurement of gap conductances.

In order to design fuel rods with specific gap conductances there are at least three fill materials which alone or in combination seem capable of spanning the range of "optimum" conductances shown in Figs. 2, 3, 11 and 12. These materials are helium, pyrolytic graphite, and ZrO_2 ; their thermal conductivities and equivalent gap conductances are presented in Table II. This flexibility in gap conductances design appears feasible within the framework of a 0.25 mm (10 mil) gap.

Fig. 1. Sodium cooling performance.

Fig. 2. Gap conductance versus steady state midplane power density.

Fig. 3. Gap conductance versus midplane burst power density.

Fig. 4. Maximum temperature at core midplane versus steady state midplane power density.

Fig. 5. Maximum temperatures at core midplane versus burst power density.

Fig. 6. Gap conductance change with BeO thermal expansion.

Fig. 7. Temperature history for clad and fuel centerline, $q_o''' = 400 \text{ W/cm}^3$, $\dot{E} = 6 \text{ kJ/cm}^3$, $h_c = 40 \text{ kW/(m}^2\cdot\text{K)}$.

Fig. 8. Clad temperature history following power burst, $h_c = 40 \text{ kW/(m}^2\cdot\text{K)}$.

Fig. 9. Clad temperature history following power burst, $h_c = 20 \text{ kW/(m}^2 \cdot \text{K)}$.

Fig. 10. Helium cooling performance.

Fig. 11. Gap conductance versus steady state midplane power density.

Fig. 12. Gap conductance versus midplane burst power density.

Fig. 13. Maximum temperature at core midplane versus steady state midplane power density.

Fig. 14. Maximum temperature at core midplane versus burst power density.

Fig. 15. Clad temperature history following power burst.

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4. SAFETY CONSIDERATIONS IN A CLASS III FACILITY

Facility design features, modes of operation, and details of proposed experiments are either not known or at best tentative at this time. In this section therefore, their safety implications will be discussed mainly in qualitative terms but will include the results of some preliminary scoping analysis.

a. Steady State Operation

A Class III facility is designed for sustained operation at prototypic power levels in the test region and therefore a significant decay heat source will be present in both the test and driver regions, and consideration must be given to providing emergency cooling capability in the event of electric power or coolant pump failure. In this respect the natural convection heat removal capability of liquid metal coolants is an area where sodium has a definite advantage over gas cooling.

The magnitude of the decay heat level depends on the value of the Figure of Merit attained by the system. The high FOM values achieved in facilities with enriched test pins means relatively low fission densities in the driver region (possibly a factor of ten lower than in a power reactor) and hence low decay heat power densities. On the other hand with 217 or more highly enriched pins in the test region the concern over the effects of test fuel relocation are greater. At the other end of the scale, when prototypic test fuel is used, the driver power densities may only be a factor of two or three below the test region levels.

b. Burst Operation

In order to produce a burst which destructively heats test region fuel pins, a large reactivity insertion has to be made in the form of a step or ramp function. Clearly, well defined and regulated operating procedures are mandatory in such operations. In addition there must be an

inherent feedback mechanism(s) which limits the burst energy to the desired level by quenching the reactivity insertion. This must be followed by a reliable and fast acting mechanical shutdown system to reduce the post-burst power plateau to an acceptable level and thereby prevent fuel and/or clad overheating.

The inherent reactivity feedback needed for reactivity quenching might arise from Doppler temperature broadening, or from axial fuel expansion, or both. Axial fuel expansion is present to some extent in all reactors. The magnitude of the expansion and the resulting reactivity effect depends on the fuel material, the height of the fuel element, the mechanical restraints, and so on. The effect of fuel cracking partly as a result of thermal cycling, has been a concern of burst reactor designers, but it appears that a well designed fuel element will tend to form radial cracks which has a minimal effect on axial expansion, particularly when strongly clad. Doppler broadening on the other hand is perfectly inherent and depends mainly on the ^{235}U enrichment and the neutron energy spectrum. Its main disadvantage is its temperature dependence. In the intermediate spectrum drivers discussed in the previous section the Doppler feedback due to raising the driver fuel temperatures from 300° to 1200°K was about twice that from 1200° to 2100°K ; approximately a T^{-1} dependence. This means that up to two thirds of the available Doppler will be absorbed in taking the reactor up to steady state operation.

Experience has shown that adiabatic burst facilities (class I) can be operated reliably when the major feedback mechanism is either axial fuel expansion or Doppler. There is no reason to believe that this should not be the case with Class III operation depending on Doppler feedback. There is also no evidence to show that the same is not true for expansion feedback.

The results of the neutronics calculations in the previous sections indicate that sufficient flexibility exists to allow a tailoring of the feedback to the desired burst characteristics. In the event of loss of sodium coolant from the driver region it is unlikely that significant positive reactivity effects will result. The systems tend to have a high leakage and the sodium volume fraction is only 15v/o.

c. Test Region Reactivity Effects

Since the purpose of the test facility is to destructively heat the test fuel pins in a simulated accident condition, gross movements of the various components of the test region (fuel, sodium, clad, wire wrap and possibly subassembly wall) are likely to take place. The possible range of reactivity effects due to relocation of these materials must be considered in the design of a safety test facility.

Several two dimensional transport calculations were performed on a single driver system corresponding to the one dimensional calculation in case 49 of Table 13. The driver was a Rover fuel configuration with a C/U ratio of 100/1, a ^{235}U enrichment of 93.2%, a 0.381 mm SS clad and 15v/o sodium coolant. The S_N code TWOTRAN was used in the S_4 approximation. The radial regions were identical to the 1-D model and in the axial direction 30 cm of nickel and sodium (15%) was assumed. Half-core symmetry (about the axial midplane) was employed in all TWOTRAN calculations.

The reference TWOTRAN calculation contained the same test geometry and materials as the one-dimensional calculation, i.e. a highly enriched 217 pin FFTF subassembly. Relative to this case, five others (shown in Fig. 7) were run in order to look at the order of magnitude of the reactivity change associated with various changes in the test region. The results are also presented in Fig. 7 along with the test cavity material/geometry description.

Due to the half-core symmetry of the calculated geometry, the two compaction calculations are unrealistically conservative, i.e., the maximum expected reactivity change upon compaction would be significantly less than the values shown. Thus for a facility limited to 19 or 37 pin test bundles there appears to be no possibility of positive reactivity feedback of more than a few tens of cents for worst case recompactions. For class III facilities testing 217 or more fully enriched test pins, there appears to exist, at least theoretically, the potential for a worst case test fuel recompaction to add in excess of a dollar of positive reactivity feedback.

C. An Accelerator-Driven Subcritical Multiplier

In the preceding section, we have considered use of a reactor, that is a critical or supercritical neutron multiplying system, for inducing fission heating in the test fuel. An alternative approach is to employ an accelerator beam to produce heating in the fuel. Direct deposition of accelerator beam energy in the fuel has been considered and it has been concluded that this is not an alternative to in-pile heating. The energy deposition, in this case, is not primarily due to fission and will have a distribution in space and by material which is generally inappropriate for simulation of fission heating. If the test fuel had a high neutron multiplication, being, for example, nearly critical, then it could be arranged that the power due to fission in the test fuel was large compared to the accelerator beam power and that the desired power distribution was reasonably simulated. In this case, however, it would be expected that the neutron multiplication would be sensitive to fuel motion and difficult to control. In addition, there would be a criticality hazard on fuel compaction. For these reasons, we believe that a better arrangement would be to dump the beam into a fixed neutron multiplier, inside of or near which the test fuel is placed for heating by neutrons from the multiplier.

Such a multiplier could have a number of attractive features in comparison with the systems considered in the previous section. For example it would not require a temperature coefficient of reactivity and could be designed as a fast or intermediate system with fully enriched uranium fuel. Rapid reactivity control would not be required, for the power level could be modulated, below a certain maximum, by varying the fraction of beam power into the multiplier. Shutdown could be simply and reliably achieved by turning the beam off of the multiplier. Finally, large amounts of fuel could be accommodated so long as the required multiplication were low enough that fuel motion could not lead to criticality.

Consideration has been given to use of the accelerator at the Clinton P. Anderson Meson Physics Facility (LAMPF) located at Los Alamos. This is a proton linac having a beam energy of 800 MeV and an average design beam current of 1 ma; hence a beam power of 800 kW. The accelerator has been operated at full beam energy and it is planned to achieve design current early in FY 76. The accelerator delivers 120 macropulses per second with each pulse having a duration of 500 μ sec. The spacing between macropulses is 7.83 msec. In addition each macropulse contains 10^5 micropulses of 0.25 nsec duration.

The accelerator will be used for a variety of purposes, including particle and nuclear physics experiments, pion radiation therapy, and isotope production. In one of the experimental areas, the Weapons Neutron Research (WNR) facility, the proton beam will be used to generate neutron pulses for time of flight experiments. A neutron multiplier could be located in or near the WNR area.

What multiplications would be required in order to obtain prototypic power densities in test fuel? In the preceding section, it was seen that in order to attain power densities of 200 w/gm in a fully enriched sub-assembly, a system power of ≥ 50 MW was required (for BeO/UO_2 fuel). This power, in a multiplier exposed to 0.8 MW beam energy, would be achieved with an energy multiplication of 62.5. The required neutron multiplication would be somewhat less. In a non-fissile beam target, about 20 neutrons per proton are obtained, so that a current of 1 ma ($= 6 \times 10^{15}$ p/sec) would produce about 1.2×10^{17} neutrons per second without fission multiplication. A fission power of 50 MW corresponds to about 1.5×10^{18} fissions per second or with 2.5 neutrons per fission, a neutron emission rate of 3.75×10^{18} neutrons per second. The required neutron multiplication is thus around

30, corresponding to a multiplication factor, $k \approx 0.97$. Thus it could be argued that for experiments requiring modest amounts of fully enriched fuel, such that the reactivity gain, Δk , due to fuel motion were certainly less than 0.03, the accelerator could be used for driving fuel at prototypic power density. By varying the fraction of the beam dumped into the multiplier, it would be possible to vary the power in the test fuel or to maintain constant power by compensating for reactivity increases due to fuel motion.

A burst capability would clearly require higher powers and higher multiplications. Indeed a rapid burst would require operation of the multiplier near or above prompt critical. Such a multiplier would then resemble a driver such as was described earlier and the advantages of coupling it to an accelerator are not apparent. Moreover, the potential hazards of a burst facility at LAMPF and adjacent to the town of Los Alamos would be of concern. We therefore have not further considered the design of an accelerator driven facility with burst capability.

A facility without burst capability but with the ability to handle large volumes of fuel could be useful for conducting PAHR experiments and for the study of boiling pools. For such experiments, as discussed under Facility D in chapter VI, power densities ~ 10 w/gm are of interest. The small multiplier, discussed earlier, would produce this level operating at 2.5 MW and with a multiplication ≤ 2 . However for such experiments, a larger multiplier with a large test volume would be more appropriate. Neutronic calculations have not yet been done on a multiplier configuration, but we believe that a multiplier having a power level ~ 10 MW and a neutron multiplication ~ 10 should be further considered as a type D facility.

If such a multiplier were to be used at LAMPF, the time dependence of the proton current, in particular the macropulse structure, would impose a

most undesirable modulation of the neutron source in the multiplier and therefore of the power in the test fuel. This effect could be avoided by collecting the protons during a macropulse in a proton storage ring and then dribbling the protons from the storage ring into the multiplier at a constant rate. Indeed a formal proposal has been made by LASL to construct a proton storage ring at WNR. The purpose of the proton storage ring, as initially conceived, was to collect the protons from say a macropulse and to then dump them in a much shorter time ($\sim 0.25 \mu\text{sec}$) in order to produce a shorter neutron pulse for neutron time of flight experiments. However it is believed that the storage ring could be operated in a complementary mode, smoothing out the macropulse structure as required for use in the multiplier. A current cost estimate for the proton storage ring is ten million dollars with construction to be completed early in FY 1979. However the project has not yet been authorized. At present it appears that something very similar to the proposed storage ring would be required for smoothing the LAMPF pulse structure and would be essential for operation of a type D facility at LAMPF.

VIII. Data Acquisition

Summary

In this chapter consideration is then given to techniques for observing fuel, clad, and sodium motion during (as well as before and after) a transient, and the data acquisition needs for the facility types A, B, C, and D are briefly reviewed. It is concluded that either a fast neutron and γ hodoscope or a flash x-radiographic system can probably furnish acceptable resolution in space and time for observing fuel and clad motion and possible sodium voiding, in experiments with moderate numbers, say ≤ 37 , of pins. For full subassemblies, resolution of gross fuel or clad motion is possible, but details, at the single pin level are probably not observable. In Section B, flash x-radiographic experiments, at the Los Alamos Scientific Laboratory PHERMEX facility, on PWR fuel pins are reported and the characteristics of a system for STF use are discussed. In Section C, the capabilities of hodoscopes and related methods for detecting neutrons/gammas emitted by fuel, clad, or coolant are examined and a particular hodoscope detector is suggested. Alternative imaging techniques including a Fresnel zone plate for acquisition of three-dimensional information are considered in a preliminary manner.

In addition, tests should clearly be instrumented with thermocouples, pressure transducers, and sodium flow meters. We have not reviewed such instrumentation for this report, largely because of an impression that it can be obtained by state of the art techniques. In the future, we plan to evaluate some possible new techniques, including acoustical methods and the incorporation of radioactive or activatable tracers in test material.

A. Introduction

It is obvious that the degree of detail and refinement to which experiments can be characterized has enormous impact upon the choice of experiments and upon the design of facilities. It is also obvious that any measurements within a test region embedded in a driver reactor are difficult and that detailed measurements are enormously difficult.

At the present time it is possible to identify methods (many of which are in use) to make some measurement in each of the necessary areas, but it is not possible to identify methods which can clearly satisfy all that analysts desire. Some judgements must therefore be made as to which techniques are presently or potentially valuable and the implications of the availability of these techniques upon the experimental program and upon the facility design must be assessed. This latter point often conflicts with facility design and compromises must be reached in many areas.

Within the time and manpower available we cannot claim to have done more than start upon delineation and solution of the data acquisition problems. What we have attempted to do to date is to delineate the needs and to make a first review of potential new or improved techniques which could have major impact upon the facility selection and design.

The first section of this chapter is a review and discussion of needs. The second section addresses itself to the new and improved techniques. Neither of these can be regarded as complete at the present time, but this is particularly true of the latter section.

1. Delineation and Discussion of Needs

In Chapter V of this report the data acquisition requirements are specified for the six principal areas of investigation,

- 1) Fuel failure and initial cladding and fuel motion,
- 2) Gross fuel/steel motion within a subassembly,
- 3) Gross material motion involving more than one subassembly,
- 4) Coupled hydrodynamic/neutronics,
- 5) Post accident heat removal (PAHR), and
- 6) Radioactive material transport.

The data acquisition requirements are then regrouped into the four selected candidate facilities.

Facility A - Few pin experiments related to pin failure and initial cladding and fuel motion (Area 1).

Facility B - Experiments related to gross fuel/steel motion within a subassembly and to a limited class of radioactive material transport phenomena (Areas 2 and 6).

Facility C - Experiments involving material motion in more than one subassembly as well as those involving coupled hydrodynamic/neutronic phenomena, and radioactive material transport (Areas 3, 4, and 6).

Facility D - PAHR-related experiments involving large masses of fuel and materials with relatively low internal heat generation (Area 5).

In each case the desirable, acceptable, and necessary (minimum) levels of data acquisition are given.

The data needs for Facilities A and B fall in generally similar areas although the details differ significantly. The needs for Facility C are generally similar in principle, to those of B, but it is felt that the massiveness of the test region probably precludes very many measurements. The needs of Facility D are generally for "conventional", but difficult measurements and do not overlap those of the prior three facilities. Here we have focused upon the needs exhibited by the first three facilities since these are likely to be the more urgent although we recognize that Facility D, PAHR, studies may well require instrumentation development.

For Facilities A, B, and C the general needs are:

- 1) Fuel, clad, and coolant locations and densities as a function of time throughout the tests.
- 2) Temperatures, pressures, and flow rates, both in the test region and in the inlet and outlet, and
- 3) Heat flow through and displacement of "structural" components such as a subassembly wall.

These three categories are given in roughly their order of importance although there may be different priorities for some tests.

These data must be obtained subject to several conditions and constraints as follows:

- 1) the neutron and gamma background of the reactor,
- 2) through the sodium of the test (with the recognition that specific tests might be with gas coolant or gas as a sodium vapor

simulant) and the intervening test and reactor structure and

- 3) without providing excessive constraints (physical, operational, monetary) upon the reactor design.

Within these common areas there are very large differences in the specific requirements particularly in

- 1) the time response (directly related to the rate-of-change of power in the test),
- 2) the duration of the test,
- 3) the radial spatial resolution,
- 4) the axial spatial resolution,
- 5) the radial viewing dimension, and
- 6) the axial viewing dimension.

A further need which is not yet well formulated is

- 7) the depth of viewing in the test sample.

If we accept the most stringent need for any test as applying to all tests then it is probable that the requirements cannot be met, but it is more prudent at this stage to accept the limiting needs as targets than to deliberately lower our sights.

The Facility A requirements are most stringent in terms of spatial resolution and time response. A spatial resolution of 0.5 mm radially, and 2 mm axially is desired with an additional wish to obtain 0.1 mm radial resolution for observation of single pins. The time resolution desired is 0.5 ms. These specifications are relaxed by about one order of magnitude in going to the acceptable and necessary levels.

The desired axial viewing dimension is 1.2 m and the radial dimension is set by the 37 pin test sample size. The test duration is of the order of 1 min. Viewing is desired in two planes.

The Facility B requirements are generally less stringent except in viewing dimensions. The desirable time response and resolution correspond roughly to the "necessary" level for Facility A, but an axial viewing dimension up to 4.5 m is desired. The radial viewing is approximately that of a subassembly. The test duration desired is a little longer at 5 min.

For Facility C it seems adequate to accept less stringent criteria than for A and B although this is, at least partially, a recognition that the test region size places practical limitations rather than a belief that detailed data would not be very valuable.

Facility D, as stated earlier, falls in the general area of more conventional instrumentation.

An approximate, broad-brush, summary of limits of needs can therefore be taken as

- 1-ms time response,

- 5 min test duration,

- 1 mm radial resolution,

- 2 mm axial resolution,

- 1 subassembly radial viewing dimension and

- 4.5 m axial viewing dimension.

Viewing should be in two planes.

Not all of the limits need be met simultaneously.

Furthermore, although it is undoubtedly useful to be able to view the entire cross section of a test it is not clear that it is necessary to view more than the "surface".

2. Data Acquisition Techniques

Here we discuss some specific techniques which have potential for application to the most difficult area of instrumentation, that of viewing material motion. Some of these techniques have not been considered (in detail at least) for application to fast reactor safety tests and in some we discuss possible improvements.

We have concentrated upon obtaining numerical and experimental information for the techniques in which there is appreciable LASL experience and expertise since this has not been feasible in studies conducted by others. We recognize that tests should include such instrumentation as thermocouples, pressure transducers, and sodium flow meters but we have not reviewed these here. Time limitations have also precluded discussion of post-test examinations, and such potential methods as acoustical holography, radioactive traces, and in-test detectors. These we propose to evaluate in the continued study.

Our evaluations here are in the area of X-radiography, and measurement of emitted neutrons and gamma-rays. All require penetrations (slots) through the driver region of the reactor.

B. Flash Radiography

1. Introduction

Flash radiography provides a possible alternative to the hodoscope for the acquisition of visual data in fast reactor safety experiments. To apply this method, a through slot in the test reactor must be provided. A high-energy, high-intensity pulsed x-ray source would be placed on one side of the reactor and a suitable detector system located on the opposite side. Collimation would be needed to ensure that the detector sees only the test assembly and not the driver core. The detection system should be activated only during the x-ray pulse (to minimize background problems). Capability for repetitive pulsing would be needed.

In the following we describe some feasibility experiments carried out using the LASL pulsed x-ray machine PHERMEX as the source and commercial radiographic film as the detector. We next discuss opto-electronic detection systems, and obtain a set of requirements for an x-ray source suitable for reactor safety experiments. Finally, we discuss possible ways of achieving such a source.

2. PHERMEX Experiments

PERMEX (Pulsed High-Energy Radiographic Machine Emitting X-rays) was designed and built by LASL to perform flash radiographic studies of detonating explosives and explosive-driven metal systems. It consists of a 30-MeV standing wave electron linear accelerator operating at 50 MHz. The electron beam is focussed onto a 1.79-mm-thick tungsten target. The effective x-ray source is less than 1 mm in diameter. The normal pulse length is 0.2 μ s. In each pulse, approximately 75 R of bremsstrahlung is produced in the forward direction 1 m from the target. The radiation is highly directional, the intensity falling to one half of its maximum value at 5 deg. from the forward direction.

In the experiments, it was desired to simulate various numbers of fast reactor fuel pins, up to a full subassembly. Since no fast reactor fuel pins were readily available, use was made of PWR oxide fuel pins. A fixture was made to hold up to 61 of these pins in a hexagonal array with pitch-to-diameter ratio of 1.24. Since the PWR pins are larger (10.8 mm o.d.) than typical fast reactor fuel pins (5.85 mm o.d.), the 61-pin array was approximately equivalent in size and fuel content to a 217-pin array of fast reactor elements. The fuel pin array was contained in a brass tube 127 mm in diameter with 3.2-mm-thick walls, which could be partially filled with water to simulate liquid sodium. One PWR pin, usually placed at the center of the array, had three fuel pellets removed and replaced with expanded polystyrene foam to simulate a void. See Fig. 1a. Measured along a diameter, each PWR pin consists of approximately 7.6 g/cm^2 uranium and 2 g/cm^2 zirconium plus oxygen.

In the first set of radiographs, the test fixture was set up 4.56 m from the x-ray target. A film pack consisting of Kodak Industrial X-ray Film types AA and T between 1-mm lead intensifying screens was placed 0.69 m behind the test fixture. In this geometry, the image is magnified by a factor of 1.15. In the first radiograph only the central pin (containing the void) was in place. Additional radiographs were made as successive rings of pins were added. In these latter cases, exposures were made both with the x-ray beam normal to the flat sides of the hexagonal array and with the beam incident along a line joining opposite corners of the hexagon (so that the pins are aligned in rows in the direction of the beam, and there is no fuel in the path of the x-ray beam in the gaps between the rows of pins). The effect of steel and titanium absorbers between the x-ray source and the test fixture was also investigated.

In all cases the voided pin and the water level were clearly visible. For arrays of 19 or fewer pins, many of the individual fuel pellets could be distinguished. Figures 2 and 3 show the results for a 19-pin array, in the two orientations described above. Figure 4 shows the results for the full 61-pin array, with the radiation normal to one of the flat sides of the hexagon. Absorbers placed between the x-ray target and the test fixture reduced the exposure to the film but had little effect on the sharpness or contrast of the image.

The results of the first set of radiographs were very encouraging. However, the experimental conditions were unrealistic, for two reasons. First, it would be necessary in practice to place the detector farther from the test object. Second, it is very likely that the x-ray beam would have to pass through several containment walls on its way into and out of the test region. Therefore, a second set of radiographs was made using the geometric arrangement shown in Fig. 1b. Here the x-ray source and detector are each 3 m from the test object, giving a magnification of 2. Two 25.4-mm steel plates were placed between the source and object, and two similar plates were placed between object and detector, to simulate containment walls. A more sensitive film, Kodak Industrial X-ray Film type KK, was added to the film packs to partially compensate for the loss in intensity due to the steel plates. Figure 5 shows the results for a single pin, Figs. 6 and 7 show two views of a 19-pin array, and Figs. 8 and 9 show similar views of a full 61-pin array. The above radiographs were all made on type KK film, which was the only type film to receive sufficient exposure to produce a useful image. We estimate that the four steel plates (total thickness 102 mm) reduced the film exposure by a factor of 9.6.

Although the radiographic quality is not as good as that of the first set of pictures, it is nevertheless quite acceptable. The water level can

always be seen, but the individual fuel pellets can no longer be distinguished. The void in the central pin is visible in all cases, although in the view of the 61-pin array along the line joining opposite corners of the hexagon, (where one is looking through a row of 9 pins), it is just barely detectable. In this case, it would be much easier to detect fuel which had moved into the coolant channels between pins than it would to detect fuel missing from a pin. The resolution of the image at the film position appears to be approximately 2 mm (equivalent to 1 mm at the object plane, since the magnification is 2). This unsharpness results from the resolution of the film (~ 1 mm for type KK film with Pb screens), geometric unsharpness (~ 1 mm at the film plane for a 1-mm source diameter and a magnification of 2), and from the effects of scattering in both the test object and in the steel plates between the object and detector.

To further investigate the image quality, the film exposure behind a row of fuel pins, divided by the exposure which would have been obtained with no pins in place, was determined for various numbers of pins. The results obtained for the two sets of radiographs are shown in Fig. 10. Also shown is a curve calculated from the x-ray absorption coefficients of the pin materials. The excess exposure over that calculated is probably due to scattered radiation; it limits the density variation which can be detected, particularly for the higher density portions of the test object. For the density equivalent to 9 PWR pins ($\sim 86 \text{ g/cm}^2$; close to the maximum density for a full fast reactor subassembly), an 11% (1 PWR pin) density variation is at the threshold of detectability under the conditions of the second set of radiographs. Under the same conditions a 5% density variation can be seen in an object equivalent to 4 PWR pins ($\sim 38 \text{ g/cm}^2$; near the maximum density for a 37-pin array).

In summary, the PHERMEX tests indicate that flash radiography can provide visual data having spatial resolution of the order of 1 mm. The ability to distinguish variations in areal density decreases as the number of pins increases, due to the effects of scattered radiation. For a 37-pin array, a change of $\sim 5\%$ of the maximum areal density can be detected, whereas for a full 217-pin subassembly, such a change would have to exceed $\sim 11\%$ of maximum to be detected. For a full subassembly, specialized techniques, such as the observation of fuel motion into coolant channels between rows of fuel pins, might be useful.

In the above discussion, no consideration was given to the possible effects of radiation produced in the test object during the radiograph. This question is now taken up.

In severe fast reactor transients, the maximum power density in the fuel is expected to be in the range $10^5 - 10^6$ watts/gram. In estimating the background radiation at the x-ray detector produced by fissions in the test object, we assume a full subassembly containing ~ 36 kg of fuel operating at an instantaneous power of 10^6 watt/gram. During a typical flash radiograph of 2×10^{-7} sec duration, such a subassembly produces 7200 J ($\sim 2.4 \times 10^{14}$ fissions). It is also assumed that the x-ray detectors cannot see the driver core, and that driver radiation scattered from the test object is of less importance than radiation produced within the object.

To determine the background radiation associated with this energy release, an experiment was carried out using the unreflected fast critical assembly Godiva IV. This critical assembly is made of 93% ^{235}U metal in the form of a cylinder approximately 17 cm in diameter and 15 cm high. A radiographic film pack of the type used in the PHERMEX experiments, together with neutron and gamma dosimeters, was set up at a distance of 3 m from the center

of Godiva. There were no absorbers between Godiva and the detectors. The critical assembly was then operated for 300 sec at 24 watts (7200 J).

The dosimetry gave the following results:

Fast neutrons	- 9.8 rem (2.8×10^8 n/cm ²)
Thermal neutrons	- 0.045 rem (4.3×10^7 n/cm ²)
Gammas	- 0.145 R.

The film pack consisted of a 1-mm lead screen, followed by 1 sheet each of types AA, KK, and T film, followed by a 1-mm lead screen, followed by a second set of three films. The data listed below are the first set of three films. Results for the second set were similar.

Film Type	KK	AA	T
Observed density	1.45	0.30	0.20
Fog + base density	0.46	0.18	0.16
Net density	0.99	0.12	0.04
Equivalent PHERMEX "R" (at film location)	0.178	0.170	(net density too low for accuracy)
Average	0.174		

If we assume that the film has the same sensitivity to the Godiva gamma spectrum as it does for PHERMEX bremsstrahlung, then the fast neutron component was equivalent to $0.174 - 0.145 = 0.029$ R of PHERMEX radiation.

In actual practice, the radiation from the fissions would be attenuated by whatever containment walls it passed through on its way to the detector. For example, 51 mm of steel attenuates PHERMEX radiation by a factor of 3.1. For the softer fission gammas, the attenuation would be higher.

Thus, the background density under the assumed conditions would be ≤ 0.3 . A uniform background density of this magnitude would not interfere with the radiographic measurements. We conclude that the background is not important (at least for film detectors).

3. Flash X-ray Cinematography of Reactor Core Material During a Transient

a. Introduction

A practical multiple flash x-ray radiographic capability which satisfies the desired characteristics given in section IV.B requires a careful optimization of the entire source-reactor-detector system. The test facility must allow a reasonably unobstructed view through the driver core and have the minimum possible attenuation for the x-ray energies of interest. A considerable saving in cost and complexity of the system will result if the x-ray source intensity can be minimized. This could be accomplished by a corresponding increase in the sensitivity of the imaging system. However there is a limit to this approach which is determined by the neutron and gamma-ray background produced by the fuel pins under study during the maximum desired power excursion as well as the statistical accuracy of the image.

Static x-ray radiographs of simulated test fuel taken with the Phermex source and x-ray photographic film have been described in the previous section. These radiographs indicate that the desired spatial resolution can be attained for a cluster of up to about 37 fuel pins. For a larger number of fuel pins x-ray scattering appears to limit the contrast of the radiographs. These studies indicate that an electronic imaging system with a signal-to-noise ratio of 100 to 1 and standard spatial resolution would be adequate.

The present conceptual design of the desired source-imaging system is considered to be conservative. Only reasonable extensions of the present state-of-the-art are assumed. Therefore only a modest development effort would be required to attain the performance characteristics presented herein. If desired, improved performance could undoubtedly be attained with a greater development effort. Areas in which significant improvements could be made with reasonable further effort are identified in the text.

b. Pertinent Reactor Configuration

In its preliminary design the reactor consists of a central test region surrounded by a 0.32-cm thick stainless steel container. This in turn is surrounded by a 5.3-cm region of sodium and approximately 2.5-cm thick stainless steel containment vessel. From this radial distance of about 15 cm the reactor driver core is separated to provide a sufficiently large viewing slot on diametrically opposite sides of the reactor for the x-ray radiography of the central test region. Estimates of the material thicknesses and x-ray transmissions at minimum absorption for the material exterior to the test region are as follows:

<u>Material</u>	<u>Thickness (cm)</u>	<u>Transmission</u>
Stainless steel	5.6	0.23
Sodium	10.7	0.73

The total transmission is $(0.23)(0.73) = 0.17$. To be conservative and to allow for possible design changes the transmission of the portions of the reactor external to the test region will be taken in the following to be 0.1.

c. Imaging System and X-ray Source Strength

The most practical imaging system¹ which meets the desired specifications consists of a fast (decay time of a few ns) plastic or liquid scintillator viewed via f/1.0 optics by a gateable image intensifier followed by a videcon. Such a system has a sensitivity for selected components of ~ 0.01 rad or approximately 1 erg/g at the fluor for 100 to 1 signal-to-noise ratio. The same distances used for the static x-ray radiographs will be assumed, i.e. 3m from the source to the center of the test region and 3 m from the test region to the fluor. The required source strength to provide 0.01 rad for the most transparent portions of the object is:

$$\text{Source strength} = S = \frac{0.01 \times 6^2}{T} = 3.6 \text{ R @ 1m/Pulse}$$

From x-ray yield data² 30-MeV electrons impinging on a W target give ~ 10 R @ 1m/ μ Coulomb. At this rate the charge per pulse is

$$Q = \frac{3.6 \text{ R @ 1m/pulse}}{10 \text{ R @ 1m}/\mu\text{C}} \approx 0.4 \mu\text{C/pulse}.$$

It is of interest to estimate the spatial resolution and the statistical accuracy afforded by this detector system and estimated source strength.

The efficiency of the fluor is about 1% (i.e. the fraction of the absorbed energy emitted as blue light). The optics system has typically an efficiency of about 5%. Therefore the energy incident on the photocathode of the image intensifier at the brightest spots is

$$\begin{aligned} U &= (0.01 \text{ rad}) \times (10^2 \text{ ergs/cm}^2\text{-rad}) \times (10^{-2}) \times (5 \times 10^{-2}) \\ &= 5 \times 10^{-4} \text{ ergs/cm}^2 = 5 \times 10^{-11} \text{ J/cm}^2 \end{aligned}$$

With blue photons the wave length $\lambda \approx 0.4 \mu\text{m}$ and the energy per photon is about 3 eV or 4.8×10^{-19} J/photon. Therefore the number of photons per cm^2 at the brightest portions on the photocathode is

$$N = \frac{5 \times 10^{-11} \text{ J/cm}^2}{4.8 \times 10^{-19} \text{ J/photon}} \approx 10^8 \text{ photons/cm}^2$$

The resolution of a typical image intensifier is about 15 line pairs/mm on the photocathode. Thus there are 150 line pairs/cm and about $(150)^2 = 2 \times 10^4$ resolution elements/ cm^2 . With the typical photocathode efficiency of 0.1 photo-electron/incident blue photon the number of photo-electrons emitted per resolution element from the brightest positions of the photocathode is

$$n = \frac{10^8 \text{ photons/cm}^2 \times 0.1 \text{ photoelectrons/photon}}{2 \times 10^4 \text{ resolution elements/cm}^2}$$

$n = 5 \times 10^2$ photoelectrons/resolution element. This corresponds to about 5% statistics in the brightest regions and about 15% statistics in the darkest regions. An increase in the sensitivity of the image intensifier if possible is not warranted because the statistical accuracy of the image will be further impaired as mentioned in the Introduction.

The diameter of the photocathode of the image intensifier is 18 mm. With ~ 15 line pairs/mm resolution the number of line pairs along a diameter is about 250 line pairs across face of image intensifier (allowance is made for the deterioration of the resolution near the outside edge of the photocathode). For the desired 0.5 mm radial spatial resolution specified in Section IV.B the maximum radial dimension for each recording system would be 125 mm or the approximate radial dimension of a 217-pin cluster. The specified axial resolution is a factor of 2 to 10 larger than the radial resolution. A cylindrical lens may be used to compress the axial dimension by a similar factor so that the axial field of view could be between 25 cm and 125 cm which is nearly consistent with the acceptable axial extent.

Standard videcons are limited by the read-out time (16 ms) to a picture rate of ~ 60 frames/s. A number of image intensifier-videcon packages gated in succession would be required to maintain the desired repetition rate specified in Section IV.B. It is anticipated that a variable pulse rate could be used with the repetition frequency increasing to $\sim 10^3$ pulses per second during the peak of the power excursion. The exact number of required recording channels is dependent on the actual experiment and cannot be specified at the present time.

1. Neutron and Gamma-Ray Background

During the maximum specified power excursion of $\sim 10^6$ watts/g the fuel under test will be a serious source of background radiation. A portion of this radiation will be incident on the fluor and could degrade if not obliterate the radiographic image.

The effects of the fast neutrons can be minimized without seriously affecting the x-rays by:

1. Using a thin fluor with Pb converter to maintain the x-ray sensitivity.
2. Using a non-hydrogeneous fluor such as NE-226.
3. Using a hydrogeneous neutron scatterer in the reactor shield between the test core and the fluor.
4. Using an optimized combination of all of the above.

The gamma-ray background from the test core is expected to be less energetic than the penetrating source x-rays. Proper choice of the Pb converter thickness may provide the necessary filtering of the photons.

This is one of the areas in which a modest investigation to determine the optimum conditions could contribute significantly to the minimization of the cost and complexity of the required x-ray source.

e. X-ray Source

A 30-MeV bremsstrahlung x-ray spectrum appears to be near optimum. A similar spectrum was used to obtain the static x-ray radiographs discussed above. The desired repetition rate of the flash radiographs and the few second duration of the proposed experiments as specified in Section IV.B effectively rule out the adaptation of Phermex techniques to the present requirements. Fortunately, if the background problems can be minimized by careful design of the imaging system, the required source strength can be kept reasonably low.

The minimum source strength of about $0.4 \mu\text{C}/\text{pulse}$ estimated above can be provided by state-of-the-art rf type electron linear accelerators. For the $0.2 \mu\text{s}$ burst duration used for the static x-ray radiographs the required peak current assumed constant during the pulse is

$$I = \frac{0.4 \mu\text{C}/\text{pulse}}{0.2 \mu\text{s}/\text{pulse}} = 2\text{A}.$$

This near nominal current could reasonably be increased by possibly a factor of 4 or 5.

The severity of the background can be reduced not only by an increase in the electron charge delivered per pulse but also by decreasing the pulse duration. A reasonable possibility of accomplishing both approaches is to use an electron storage ring. With this device following the rf linear accelerator it may be possible to inject a few amperes of current in a few microseconds and extract the accumulated charge in a few tens of nanoseconds thereby improving the signal-to-noise by a factor of $\sim 10^2$.

Further study on this and other possibilities such as the magnetic induction accelerator³ will be required to determine the optimum accelerator configuration.

Summary

A multiple flash x-ray radiographic capability appears to be a feasible data acquisition option for a Safety Test Facility. It could provide invaluable information regarding fuel pin and assembly failure. Based on near state-of-the-art technology its overall cost would be but a small fraction of that of the whole facility.

References

1. The technology described in this subsection exists today and is used routinely by personnel in J-12 in their testing activities. This information was obtained from George Berzins, J-12, via private communication.
2. LA-3241, p. 109.
3. J. Leiss, NBS, private communication.

C. Detection of Neutrons and Gamma Rays Emitted by Test Material

1. Application of the Hodoscope for STF Diagnostics

A useful starting point in reviewing the applicability of the hodoscope system is to summarize what we know about the system on TREAT: Prior to the design of the fast neutron hodoscope as it exists on TREAT, a number of other techniques, including a gamma-ray pinhole, were tried and abandoned.¹ Other early attempts to measure pin failures and sodium voiding included spiking the sodium coolant and clad with gadolinium and boron to enhance its gamma-ray output, and spatially observing the characteristic lines from the clad material. Because approximately one-half of the gamma energy emanating from the test package was delayed gammas, the signal-to-noise was such ($S/N < 1$) that these methods were abandoned. By 1964 it became apparent that fuel motion detection via fast neutrons was feasible, and the hodoscope system as we know it was constructed.

Because of (n, γ) reactions in the steel and the fact that the energy emitted as prompt neutrons is approximately one-half that of prompt-plus-delayed gammas, the gamma-ray background on TREAT has been a continuing problem. In current operation, up to 5 cm of lead is used² in front of the 5% ZnS(AG) - plastic "Hornyak buttons." In spite of these attempts to minimize the effect of gamma-ray pulse pile-up in the scintillators, non-linearities occur at TREAT power levels of $\sim 100 \text{ MW}$ ¹ (TREAT capability is 20 MW - 16,000 MW, with burst length of 200-300 msec FWHM). This power level corresponds to only about 10 KW/pin where the length of the fuel pins is about 30 cm. Other problems such as gain shifts and detector calibration uncertainties also plague the TREAT hodoscope system, with $\sim 10 \gamma/s/n$ present at all power levels. At present consideration is being given to using pulse shape discrimination, backup fission counters using ^{234}U , and NaI(Tl) for sodium voiding detection. Adequate

resolution of the recreated images remains a problem, however, in an consideration for the upgrading of the data accumulation/readout system on TREAT.

To optimize detector design for an STF hodoscope, it is useful to estimate the neutron fluxes that can be expected to occur during both steady state and burst operation. Assume a 3 m long hodoscope positioned against the test fuel containment, with slots of 1.0 cm (axial) by 0.2 cm (radial). Assume also that the areal fuel density per pin is $\frac{10 \text{ gm}}{\text{cm}^3} * 0.5 \frac{\text{cm}}{\text{pin}} = 5 \frac{\text{gm}}{\text{cm}^2\text{-pin}}$, and that there are 3×10^{10} fissions/J and 2.5 neutrons/fission (neglect absorption). Thus, under steady state conditions of 200 w/gm we have a neutron flux at the detectors of

$$200 \frac{\text{w}}{\text{gm}} * 3 \times 10^{10} \frac{\text{f}}{\text{w-sec}} * 2.5 \frac{\text{n}}{\text{f}} * \frac{0.2 \text{ cm}^2}{10^6 \text{ cm}^2} * \frac{5 \text{ gm}}{\text{cm}^2\text{-pin}} = 1.5 \times 10^7 \frac{\text{n}}{\text{cm}^2\text{-sec-pin}}$$

Under burst conditions of 2000 J/gm in 20 msec, the maximum power density becomes $7.5 \times 10^9 \frac{\text{n}}{\text{cm}^2\text{-sec-pin}}$. Presumably $\sim 10 \text{ } \gamma\text{'s/n}$ would also be present at all power levels on the STF. In future work, Monte Carlo calculations will be performed to better estimate the neutron and gamma signal and background for a variety of hodoscope configurations and slot sizes.

From the standpoint of sensitivity to material motion, a high neutron or gamma count rate is desirable. For example, a count rate of $10^3/\text{msec}$ would provide 3% statistics for reconstruction of "hodoscope pictures" on a one msec time scale.

Thus based upon a) the relatively high gamma to neutron flux on TREAT, b) the need for recording hodoscope data under both long-term steady state and burst conditions, c) the possibility of having to record data for from one to ten pins, and d) the need to maintain a high count rate to insure good statistics, the hodoscope detection system should have the following characteristics:

- 1) The detection system should have a relatively fast time response in comparison to the 200 nsec system on TREAT.
- 2) The detection system should be able to discriminate the fast neutrons from the ~ 10 times more intense gamma flux, and, if possible, use the gamma flux for in-parallel measurements of sodium voiding and clad motion.
- 3) The detection system should be able to operate over a wide dynamic range extending from $\sim 1.5 \times 10^7 \frac{n}{cm^2\text{-sec}}$ for a single pin at steady state conditions to $\sim 7.5 \times 10^{10} \frac{n}{cm^2\text{-sec}}$ for a 217 pin subassembly (assuming 10 pins are viewed by the slot) undergoing a 2000 J/gm-20 msec burst.
- 4) The detection system should be designed to optimize the count rates at all neutron and gamma flux levels to insure good statistics and adequate sensitivity to fuel and clad motion and to sodium voiding, while minimizing the need for special data recording systems.

In view of the mechanics involved in doing an experiment on TREAT, the STF hodoscope system should also incorporate the following features:

- 5) On-line and off-line image reconstruction techniques capable of handling large amounts of recorded data (up to 1000 channels) of varied resolution on a reasonable time scale; both "quick-look" and high resolution data readout systems should be employed.
- 6) The hodoscope must be dimensioned so as to permit easy access to the detectors and adequate detector shielding; the system must also allow for easy alignment and calibration.

One approach which is feasible and meets all of the above detection requirements is to employ fast (~ 30 nsec) fission counters with programmable sensitivity to detect the fast neutrons, and, in series on the same hodoscope

slot, use a fast, neutron-insensitive fluor - PM tube combination for gamma pulse detection. Thus, two different types of detectors would be used for simultaneous measurements of both fuel and clad motion and sodium voiding.

Figure 11 is a schematic of this combined detector system. In such a detector, the range of sensitivity is determined by the number and thickness of the foils. The maximum thickness that a given foil can be made (due to self adsorption of the fission fragments) is $\sim 10 \text{ mg/cm}^2$. Thus, by counting the fission fragments from up to 10 foils³, the effective sensitivity of the detector can be made to vary by one order of magnitude. The switching (varying of sensitivity) can be easily achieved electrically on a microsecond time scale by employing a rate-sense circuit. A varying sensitivity to optimize the results from a given test can be programmed in advance by using a small digital computer⁴. Although as many as 10 foils might be used in a counter, the collector plates would not necessarily require separate circuits [see Fig. 16(a)].

The counter would probably employ a gas mixture of 90% argon and 10% carbon dioxide at a pressure of $\sim 20 \text{ psia}$ ^{3,5}. The plate spacings would probably be $\sim 0.5 \text{ cm}$ to maximize the fission fragment to alpha background signal^{3,5,6}, making the detector $\sim 5 \text{ cm}$ long. The dead time of such a detector would be $\sim 30 \text{ nsec}$, with a signal to noise ratio of approximately five.

To eliminate elastically scattered neutrons the foils could be made up of fissile materials which have effective thresholds in the energy region between 0.5 and 2.0 MeV. For example, the detection efficiency of a 10 mg/cm^2 foil of ^{237}Np ($\sigma_f = 1.7 \text{ barn}$ for 2.0 MeV neutrons) becomes

$$\epsilon_a = 1 \frac{\text{fragment}}{\text{interaction}} * \frac{6 \times 10^{23}}{237} \frac{\text{atoms}}{\text{gm}} * 10^{-2} \frac{\text{gm}}{\text{cm}^2 \text{-foil}} * 1.7 \times 10^{-24} \frac{\text{cm}^2}{\text{atom}}$$

$= 4.3 \times 10^{-5} / \text{foil}$. If fission fragment pulses from 10 such foils are

being summed during a steady state single-pin experiment, the count rate would be

$$1.5 \times 10^7 \frac{n}{\text{cm}^2\text{-sec-pin}} * 1 \text{ pin} * 0.2 \frac{\text{cm}^2}{\text{detector}} * 4.3 \times 10^{-4} \frac{\text{pulses}}{n}$$

$= 1.3 \times 10^3$ pulses/detector/sec, or enough for a 3% statistical resolution "hodoscope picture" every second. If 10 pins were being viewed (i.e., a full subassembly) the count rate would be 1.3×10^4 /sec.

For an experiment involving a burst that adds ~ 2000 J/gm to the fuel of a single pin in a time of 20 msec, the count rate is

$$7.5 \times 10^9 \frac{n}{\text{cm}^2\text{-sec-pin}} * 1 \text{ pin} * 0.2 \frac{\text{cm}^2}{\text{detector}} * 4.3 \times 10^{-4} \frac{\text{pulses}}{n} = 6.5 \times 10^5/\text{sec},$$

or enough for a 3% statistical resolution "hodoscope picture" every two msec. A burst-test involving a full subassembly would result in a similar count rate if the fission fragments from only one foil were counted at the height of the burst. As the fuel cooled, more collector plates could be electronically switched into the data accumulation system.

Thus, for an experiment involving between 1 and 217 pins undergoing both steady state and burst conditions, the number of foils to be counted could be programmed in advance and controlled by a digital computer to maintain a count rate ranging from 1.3×10^3 /sec to 3×10^6 /sec. For a detector dead time of 30 nsec, the observed count rate at 3×10^6 /sec is $\dot{m} = 2.75 \times 10^6$ /sec, resulting in a system dead time of $100 \text{ m } \tau = 8.3\%$.

For those experiments involving only a partial subassembly, the length, width, and height of the hodoscope and the slot size could be adjusted to improve the conditions outlined above. Special consideration must also be given to those experiments involving only a few pins in which a fast scram is initiated in the driver fuel to effectively cool the test fuel from 2×10^5 w/gm to ~ 20 w/gm in a few msec, while the fuel maintains its high velocity

(~ 1000 cm/sec). In all instances the design and fabrication of the hodoscope will be coordinated with the reactor designers to minimize the loss of reactivity and the effect on burst width.

For gamma detection, a liquid scintillator such as NE-226 seems suitable⁷. Its efficiency to gammas (for 0.5 cm thick scintillator) is $\sim 4 \times 10^{-4}$, with $\tau = 3.3$ nsec. The scintillator does not contain hydrogen and is relatively insensitive to neutrons. Thus, assuming 10γ 's/n as observed on TREAT, we can expect gamma fluxes ranging from $1.5 \times 10^8 \gamma$ -photons/cm²-sec-pin for the steady state (200 w/gm) case up to $7.5 \times 10^{10} \gamma$ -photons/cm²-sec-pin for bursts. The count rate ranges from $1.5 \times 10^8 * 0.2 * 4 \times 10^{-4} * 1 \text{ pin} = 1.2 \times 10^4/\text{sec}$ to $7.5 \times 10^{10} * 0.2 * 4 \times 10^{-4} * 10 \text{ pins} = 6 \times 10^7/\text{sec}$. In this latter case the observed count rate $\dot{m}$ is $\dot{m} = \frac{6 \times 10^7}{1 + (6 \times 10^7)(3.3 \times 10^{-9})} = 5.0 \times 10^7/\text{sec}$ for a dead time of $100 \dot{m} \tau = 16.5\%$. The gamma S/N due to the presence of the fission foils is $\frac{10 \gamma\text{'s/n (signal)}}{1 \times 10^{-3} \gamma/\text{n(noise)}} = 10^4$.

By combining the results from the two independent detector systems outlined above, the ratio of neutrons to gammas at all power levels can be determined and accurate detector dead time corrections made. The presence of two independent detectors also simplifies data reduction and detector system troubleshooting.

The gamma sensitivity of the hodoscope system to sodium voiding can be determined as follows: Assume a count rate of $10^6/\text{sec}$ in each detector and that a "hodoscope picture" is reconstructed every msec. Thus, we have 10^3 pulses/msec or 3.3% statistics. The difference in transmitted flux becomes

$$\left(\frac{I_1 - I_2}{I_0} \right) = .033 = e^{-\mu t_1} - e^{-\mu t_2} \approx -\mu(t_1 - t_2) = -\mu \Delta t.$$

$$\text{Thus } 0.043 \frac{\text{cm}^2}{\text{gm}} * 0.97 \frac{\text{gm}}{\text{cm}^3} * \Delta t = .033 \text{ or } \Delta t \approx 0.8 \text{ cm.}$$

In future work transport calculations will be performed to obtain better estimates of the gamma sensitivity of the hodoscope. Two additional types of detectors were examined and found to be less suitable than the system outlined above. The first involved the use of solid state detectors to count fission fragments from a fission foil. To optimize the count rate, two detectors were considered for each foil. However, the pulse height from solid state detectors begins to deteriorate after the detectors have been exposed to $\sim 10^8$ fission fragments/cm². Thus, at steady state conditions in which 10 fuel pins are being viewed, the fission fragment flux is

$$1.5 \times 10^7 * 10 \text{ pins} * 0.2 \text{ cm}^2 * 5 \times 10^{-5} \frac{\text{fragments}}{\text{foil}} = 1.5 \times 10^3 \text{ fragments/sec.}$$

for a detector lifetime of $\frac{1 \times 10^8}{1.5 \times 10^3} = 6.6 \times 10^4 \text{ sec}$ or $\sim 18 \text{ hrs.}$

To achieve the same sensitivity as the fast fission counter would require 10 detectors/channel which would probably have to be replaced after each test. At a 1975 cost of \$300/detector, this detection system thus becomes prohibitively expensive for a 1000 channel system.

The feasibility of employing pulse shape discrimination (PSD) was also examined. The advantage of this system is that neutrons and gammas can be detected within the same scintillator, thereby necessitating only a single scintillator - PM tube combination. The neutron and gamma count rates are then read-out on separate scalars. The optimum fluor for this type of detector would be NE-213, a liquid with a decay constant of 3.7 nsec.

The main principle of the PSD system involves differentiating the signal response of the scintillator to neutrons and gammas. Unfortunately, adequate differentiation can occur only if the signal is followed for $\sim 600 \text{ nsec}^8$.

Under the optimum conditions, the time response of the PSD system thus becomes $\sim 1 \mu\text{sec}$, or approximately 30 times slower than the fast fission counter. For

a 0.5 cm thick NE-213 scintillator, the detection efficiency for neutrons is $\epsilon_n = 1.8 \times 10^{-3}$. Thus, for a detector area of 0.2 cm^2 , and $\tau = 1 \text{ } \mu\text{sec}$ the PSD system becomes 50% dead at an observed count rate of $\dot{m} = \frac{50}{100 \times 10^{-6}} = 5 \times 10^5 / \text{sec}$.

Without being able to adjust the sensitivity of the scintillator, the count rate during a subassembly-burst experiment could be as high as

$$7.5 \times 10^{10} \frac{n}{\text{cm}^2\text{-sec}} * 0.2 \text{ cm}^2 * 1.8 \times 10^{-3} \frac{\text{pulses}}{n} = 2.7 \times 10^7 / \text{sec}$$

or approximately two orders of magnitude higher than the optimum PSD system can handle.

2. Application of Multiple Hodoscopes

For experiments in which a cluster of fuel pins is being examined, the outer ring of pins unavoidably conflicts with the signal from the inner pins. Indeed, to see fuel voiding in every pin in a 217 pin subassembly with one hodoscope would require seeing through up to 16 pins, whereas the number is reduced to 8 pins with two colinear hodoscopes. In addition to providing a better signal-to-background, the simultaneous use of two neutron and gamma-ray hodoscopes would permit clad relocation, sodium voiding, fuel voiding, and pin failure propagation measurements to be performed in parallel. For example, in the case of a 37-pin cluster (one central pin and 3 surrounding rings), the optimum view for clad disruption and sodium voiding near the central pin requires a view along both sides of a 3-deep row of pins. For fuel voiding measurements in the central pin, the optimum view is provided at right angles to the above orientation, in which configuration of the view is only through the equivalent of one pin. In the case of a 217 pin cluster the central pin is shadowed by 8 pins along one line-of-sight and by 4 pins at right angles (see Fig. 12). Using two hodoscopes, the worst case (217 pin subassembly) is 8 pins deep from both directions (hodoscopes at 3 o'clock and 6 o'clock and

pin at 11 o'clock), but in this case the pin is on the outermost ring. Thus for simultaneous clad relocation, sodium voiding, fuel voiding, and pin-failure propagation studies on single or multiple subassemblies, the usefulness of the diagnostics system is markedly enhanced by the employment of two hodoscopes.

The use of two hodoscopes, with perhaps different resolutions, presents the problem of reconstructing the time-dependent structure of a three-dimensional object (the cluster of pins) from two of its time-dependent projections. Details can be obtained from the projections considered collectively that are not apparent in any one of them alone.

There has been considerable interest in this type of image reconstruction and a number of different techniques have been proposed⁹⁻¹¹. The techniques that exist generally fall into two basic classes; one in which the reconstruction is performed in normal signal space (algebraic approach) and one in which it is performed in Fourier Space (Fourier approach). Because most reconstruction algorithms are more easily implemented in one space than in the other, an in-depth review of these algorithms would be necessary to optimize the information from the two-dimensional hodoscope "pictures." Efforts underway in Group C-8 at LASL could easily be extended to include such a review¹², as well as lending assistance in the development of useful digital image reconstruction and data readout techniques.

3. Application of Fresnel Zone Plate to Gamma-Ray Imaging

The advantages of obtaining a three-dimensional image of the individual fuel pins and subassemblies undergoing an excursion are obvious and have led to an examination of the Fresnel zone plate as a diagnostics tool. Both on-line (vidicon camera) and off-line (film) systems are considered for data readout.

In 1961 Mertz and Young¹³ realized that a Fresnel zone plate could be used to construct a hologram using incoherent radiation and they applied the

principle to imaging X-ray stars. The technique is also applied to imaging gamma-ray sources¹⁴⁻¹⁵, in which case the zone plate does not produce diffraction effects but is only used to cast a geometric shadow onto a NaI (Tl) crystal (see Fig. 13).

A zone plate consists of alternating opaque and transparent rings whose bounding radii are in the ratio of the square roots of successive whole numbers. If the shadow of the zone plate is photographed on transparency film at a suitable reduction in size, a miniature zone plate image results which simulates the hologram of a point source. This similarity to an interference hologram formed with coherent light is due to the fact that the interference pattern of a plane wave and spherical wave is just a zone plate pattern. Upon illuminating the pseudo hologram with plane coherent light, one reconstructs the image of the original point source at the focus of the hologram. Images may also be constructed for a collection of point sources and small objects. If the objects are located at different distances from the zone plate they will cast different size shadows, and images will be constructed at different distances from the hologram, thus giving rise to a three-dimensional image.

The theoretical limiting resolution of the zone plate is determined by the width W of the smallest or outermost ring; $W = r_1(\sqrt{n} - \sqrt{n-1})$, where r_1 = radius of the central opaque zone and n = number of zones. For a zone plate of radius r_n and for sources at a distance A greater than r_n , the detection efficiency of the zone plate ϵ_1 is $\epsilon_1 \approx \frac{r_n^2}{8A^2}$. This includes the fact that the zone plate area is 50% open. In practice the maximum zone plate radius is limited because the lead plate must be constructed with a finite thickness to stop the gamma-rays, thus limiting the field of view for the smaller rings. The limiting object resolution R is related to W by the relative distances between the object, zone plate and crystal: $R = \frac{A+B}{B} W$, where B = distance between the zone plate and the scintillator.

Two additional efficiencies must also be considered; that of the film or NaI (Tl) scintillator ($\epsilon_2 = 7 \times 10^{-3}$ at 2 MeV gamma energy for a ~ 0.5 cm thick crystal) and the efficiency which represents the added exposure needed to obtain a zone pattern with adequate contrast needed to reconstruct a meaningful hologram. This latter efficiency ϵ_3 has been estimated¹⁵ to be $\epsilon_3 \approx 10^{-2}$.

For 2 MeV gamma rays the thickness of a tungsten zone plate should be about $e^{-2} = 0.14$ corresponding to $t \approx 2.4$ cm. Consider a 30 zone plate of 12.4 cm in diameter located 3 meters from the test region. Assume $B = 1$ m and $r_i = 1.6$ cm. The width of the outermost ring then becomes $W = 1.6$ cm $(\sqrt{30} - \sqrt{29}) = 1.5$ mm, and the limiting object resolution becomes $R = \frac{A+B}{B}$. $W = 6$ mm. The optics efficiency for viewing the fluor is $\epsilon_4 = 0.05$, while the solid angle efficiency is $\epsilon_1 = 1/2 \frac{\pi r_n^2}{4\pi A^2} = \frac{(6.2 \text{ cm})^2}{8(300 \text{ cm})^2} = 5.3 \times 10^{-5}$.

The total efficiency of the zone plate/fluor system thus becomes

$$\epsilon_t = \epsilon_1 \cdot \epsilon_2 \cdot \epsilon_3 \cdot \epsilon_4 = 5.3 \times 10^{-5} \times 7 \times 10^{-3} \times 5 \times 10^{-2} \times 10^{-2} = 1.8 \times 10^{-10}.$$

The prompt gamma ($\sim 3\%$ of total power) source strength of a 12.4 cm long portion of a 0.5 cm diameter fuel pin at 10^5 w/gm (2000 J/gm in 20 msec) becomes $10^5 \frac{\text{w}}{\text{gm}} * 10 \frac{\text{gm}}{\text{cm}^3} * \pi r^2 \text{cm}^3 * 3 \times 10^{-2} = 7.3 \times 10^4$ watts. The area of observation is $\pi r_n^2 = 120 \text{ cm}^2$. Thus, the source strength/unit area at the image intensifier becomes $7.3 \times 10^4 \frac{\text{J}}{\text{sec}} * 1.8 \times 10^{-10} = 1 \times 10^{-7} \frac{\text{J}}{\text{cm}^2\text{-sec}}$. For a signal to noise of $S/N = 100$ the sensitivity of an ordinary intensified vidicon camera is $10^{-9} \frac{\text{J}}{\text{cm}^2\text{-frame}}$.

Thus we have enough gamma source strength from a single 12.4 cm long pin during a burst to have $\sim 10^2$ frames/sec, implying that ~ 10 pins would be needed to obtain images with one msec time resolution.

Because the camera rep rate is ~ 16 msec, more than one vidicon would have to be used during a 20 msec burst. For steady state experiments, the power density would be $\sim 10^3$ lower, necessitating an unusually long intensifier imaging ("soak") time.

In a manner similar to the one outlined above, a neutron Fresnel zone plate could be utilized. In this case a ~ 2 cm thick plate made up of graded boron-tungsten could be used. In place of the NaI (Tl) scintillator, a plastic fluor such as NE-102 could be used.

In summary it appears feasible to employ a Fresnel zone plate system for burst experiments involving several pins. At the expense of spatial resolution, larger zone plates could be considered which would enhance the test fuel area being viewed. Because ASA 3000 film has a sensitivity of

$$3 \times 10^9 \frac{\text{ev}}{\text{cm}^2\text{-frame}} * 1.6 \times 10^{-19} \frac{\text{J}}{\text{ev}} = 4.8 \times 10^{-10} \frac{\text{J}}{\text{cm}^2\text{-frame}} \text{ for a signal/noise} = 10,$$

it may also be feasible to use an electro-optical shutter for fast framing directly onto film. Such a system would eliminate the rather limiting 16 msec rep rate inherent in all vidicon cameras.

4. Application of the Pinhole Camera to Gamma-Ray Imaging

The properties of the pinhole camera are described in almost any elementary optics textbook, and because of its simplicity and proven usefulness¹⁵ deserves consideration as a diagnostics tool for gamma-ray or neutron imaging (see Fig. 14).

For gamma-rays a 2.4 cm thick tungsten pinhole seems adequate (e^{-2} for 2 MeV gammas). For such a thick pinhole $d = \frac{1}{(1-f)} \cdot \frac{S}{2} \cdot \frac{t}{h}$, where the resolution element at the source is approximately the same size as the pinhole diameter d . For the typical case of a thick pinhole we have:

field of view; $S = 10 \text{ mm}$

fraction of solid

angle subtended on axis; $f = 0.9$ (90%)

distance to source: $h = 1 \text{ m}$

thickness of pinhole: $t = 2.4 \text{ cm}$

$$d = \frac{1}{0.1} \frac{10 \text{ mm}}{2} \frac{24 \text{ mm}}{1 \times 10^3 \text{ mm}} = 1.3 \text{ mm}.$$

$$\frac{10 \text{ mm}}{1.3 \text{ mm}} = 7.5 \text{ resolution elements across the source diameter.}$$

The efficiency of a pinhole is given by (see Fig. 14) $\epsilon_1 = \frac{d^2}{16R_2^2}$;

where R_2 = distance between the pinhole and the fluor.

For $R_2 = 3 \text{ m}$ (magnification of $m=3$), we have $\epsilon_1 = \frac{(1.3 \text{ mm})^2}{16(3000 \text{ mm})^2} = 1.2 \times 10^{-8}$.

The efficiency of NaI (Tl) to 2 MeV gammas is $\epsilon_2 \approx 7 \times 10^{-3}$. With an optics efficiency of ~ 0.05 for transmitting the blue light onto an image intensifier or film, we have, for the geometry above, an overall system efficiency of $\epsilon_t = 1.2 \times 10^{-8} * 7 \times 10^{-3} * 5 \times 10^{-2} = 4 \times 10^{-12}$, or ~ 40 times less than that of the Fresnel zone plate. Thus, it would appear that the pinhole system would be limited to burst experiments involving many pins. However, it might be possible to recover this difference in sensitivity and obtain 3-dimensional images by using a non-redundant pinhole array containing up to 27 pinholes¹⁵. Such a technique like gamma-ray imaging with a zone plate, would be an extension of high resolution optical imaging techniques and awaits laboratory experiments to determine its feasibility and limitations.

The above zone plate and pinhole system calculations assumes an intensified vidicon camera with a sensitivity of $10^{-9} \frac{\text{J}}{\text{cm}^2\text{-frame}}$. However, measurements have been performed with carefully selected intensified vidicons having a ~ 100 times more sensitivity than the ordinary intensified camera¹⁵. An additional factor of $\sim 10^2$ in sensitivity might also be obtained by employing intensifiers with two channel plates within the same envelope.

In summary, it appears feasible to employ the pinhole camera for multiple-pin burst tests, although work should proceed to determine the efficiency and signal to noise ratio for the potentially more useful non-redundant pinhole array.

5. Off-line Image Enhancement

Because reconstructed photographs, radiographs, or three-dimensional images obtained by the methods outlined above may vary in contrast and resolution, methods of off-line image enhancement (deblurring) have been examined. Although optical processing is very useful, particular attention is given to LASL's capability for digital image processing.

Since the advent of the laser, and of such tools as spatial filtering and holography, optical data processing has become the standard technique for problems such as distinguishing aerial photographs exhibiting cloud cover, terrain analysis and land-use determination, fingerprint identification, and character and word recognition. The techniques are inexpensive and fast, with all operations performed in parallel on two-dimensional images. Indeed, electron micrographs of biological test specimens have been deblurred to their theoretical diffraction limit of resolution by optical holographic techniques in the time required to insert a slide in front of a laser beam¹⁶. Thus, for a limited number of data channels, optical image processing is a useful and relatively inexpensive tool.

For most automated applications, involving a large number of data channels, two dimensional image enhancement must be performed on digital computers. In LASL's effort¹⁷ image digitizing is done with a microdensitometer, which can take samples at steps as small as 1 micron spacing between raster point centers. The travel speed is such that a full 35 cm x 43 cm film can be sampled (100 micron centers) in one hour. The digitized data is written onto magnetic tape and the tapes are taken to the CCF for processing. Associated with the CCF computers and specialized hardware (see Fig. 15) is a library of software, the Los Alamos Digital Image Enhancement Software (LADIES). Among those subroutines currently in the LADIES library are those for convolution

and digital filtering, density scaling, data packing and unpacking, and sub-routines related to specific PHERMEX radiography problems such as film characteristics and X-ray angular intensity variations.

Three types of hardware are used for displaying the processed images: The Photometric Data System (PDS) display utilizes a light emitting diode whose intensity is modulated to expose the film. The method possesses the highest precision and greatest flexibility; it is also the slowest. Another film writing system used for output is the FR-80 film recorder made by Information International, Inc. This system possesses precision and flexibility, but its resolution is not as good as the PDS; on the other hand it is faster and provides a quicker turn around. Data output is via 35 mm film. For a "quick look" display a Comtal Corporation on-line CRT display is used. At present the display is being interfaced to the HYDRA network by a PDP-11 mini-computer, and ultimately will allow on-line interaction with image processing operations via the HYDRA network.

The main disadvantage of digital image processing is the enormous amount of core storage and magnetic tape that is needed. Indeed, NASA data requirements, in which $\sim 55,000$ images/day would have to be recorded and processed, preclude the use of the all-digital scheme. Similar problems would confront users of a 1000 channel hodoscope in which an image was recorded every one msec for several minutes during a reactor test. Assuming that the channels are combined to produce only one composite image every msec, there would result 3000-5000 images that would have to be processed for each reactor test. By combining the advantages of the digital computer with those of optical data processing, it is possible to achieve a reasonable and versatile totally on-line system capable of handling the potentially large amounts of data that would be produced in a 1000 channel hodoscope system. Two such hybrid optical/digital systems are described in the following section.

6. Data Recording and Playback

This section outlines several methods that might be useful for data recording, processing, and playback of information from a hodoscope. Additional methods of digital to analog image processing will be examined in later reports.

Count rates of $\sim 10^4$ /sec can be easily recorded on standard 8-track tape decks with 2 MHz frequency response. These machines can record 33,000 bits/inch (per 8 tracks) and operate at 120 inches/sec tape speed. Assuming 6 bits/word (or pulse), the capacity of the recorder becomes 8×10^5 counts/sec. Six bits per count provides $1/2^6 = 1.6\%$ resolution.

Faster tape decks are also available for recording during a burst such as the Terra Bit system¹⁸, which can record up to 10^6 bits/sec at tape speeds of 1000 inches/sec. However, the most straight forward method of recording burst data on tape would be to use binary circuits for data compaction and sigma compression.^{4,18,19} These techniques allow a relatively high data rate ($\sim 10^6$ /sec) to be transmitted and recorded at a relatively low bandwidth (~ 1 KHz). Thus, by using data compaction techniques in conjunction with magnetic tape and video disk recording, both low resolution and high resolution data playback is feasible. For example, the video disk would permit an "instant color replay" in slow motion for a low resolution "quick-look," while the magnetic tape could be processed off line for high resolution data analysis and image enhancement. A schematic of this type of data readout system for a hodoscope channel is shown in Figures 16(a), 16(b) and 16(c). If two hodoscopes were employed, a split screen T.V. display could be used.

Another method for high speed data processing employs an on-line image-forming light modulator²⁰ (OLIFLM) device as an integral part of an optical data processing (OPD) system (see Fig. 17). Some of these on-line devices,

which in effect provide input data to the ODP system, accept optical inputs, while others accept electrical input signals. When the input to the OLIFLM is a modulated electrical input signal rather than modulated light, computer generated spatial filters and holograms can be employed and computer interfacing becomes more convenient. The entire software and hardware package then becomes an optical-digital processor, whose input may be optical or video information, film scans, contents of a digital computer core, disk, or drum memory, or the contents of a storage tube that serves as an analog memory.

The optical-digital processor system operates as follows: a digitized video image is provided to a digital computer serving as a control interface unit (DCCI). The DCCI sends the video signal to an OLIFLM which uses it to modulate a writing electron gun, causing a charge pattern corresponding to an input signal (assumed to be of a two dimensional video format) to be deposited on a crystal (KD_2P) surface. This charge pattern creates an electrical field distribution across the crystal. By the Pockels effect, this field distribution causes a laser beam projected through the crystal to be modulated point by point, which after propagation through a crossed polarizer/analyzer can be converted to an amplitude modulated optical signal that is also fed into the DCCI. The DCCI then interacts with a larger programmable digital computer (PDP-11) for data processing, spatial filtering, and image enhancement. The information is then returned to the DCCI for analog display in a variety of modes (hard copy, T.V. display, etc.). The input information is usually scanned onto the crystal during the frame time of a standard 525 line television (~ 30 msec). Reading consists of pulsing the laser with the vertical sync pulse. For application to the hodoscope system, the electron beam could be modulated (spatially as well as its intensity) by the count rate in each channel.

Frequently two laser light-modulating OLIFLM's are used, one in the input plane of the Fourier transform lens and one in its spatial filter plane. By employing a beam splitter and display tube between the two OLIFLM's, the Fourier transform plane and correlation plane data is available for display or analysis.

It appears feasible to adapt the electron beam modulated data readout systems outlined above to an on-line application involving up to 1000 channels of digital information from a neutron or gamma hodoscope. Such systems while not a panacea, would offer promise and flexibility for handling the massive amounts of data which would be accumulated by one or more hodoscope systems. The digital information (number of pulses) in each hodoscope channel could also be recorded on magnetic tape for off-line processing.

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VIII. C

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a. Test Object

b. Test geometry

Figure 1

Fig. 2. Radiograph of 19 PWR pins,
looking at flats of hexagon.
Magnification = 1.15, no
absorber.

Fig. 3. Radiograph of 19 PWR pins,
looking at apex of hexagon.
Magnification = 1.15, no
absorber.

Fig. 4. Radiograph of 61 PWR pins,
looking at flats of hexagon.
Magnification = 1.15, no
absorber.

Fig. 5. Radiograph of 1 PWR pin.
Magnification = 2,
102 mm steel absorber.

Fig. 6. Radiograph of 19 PWR pins,
looking at flats of hexagon.
Magnification = 2,
102 mm steel absorber.

Fig. 7. Radiograph of 19 PWR pins,
looking at apex of hexagon.
Magnification = 2,
102 mm steel absorber.

Fig. 8. Radiograph of 61 PWR pins,
looking at flats of hexagon.
Magnification = 2,
102 mm steel absorber.

Fig. 9. Radiograph of 61 PWR pins,
looking at apex of hexagon.
Magnification = 2,
102 mm steel absorber.

Figure 10

Figure 11. Schematic of fast fission counter and gamma-ray detector.

Fuel $\rho = 10 \text{ gm/cm}^3$
 Na $\rho = 1 \text{ gm/cm}^3$
 Neglect clad
 0.63 cm dia. rods
 217 pins
 0.75 cm pitch

Figure 12. Arrangement of a 217 pin subassembly showing fuel and sodium areal densities.

Figure 13. Schematic of a Fresnel zone plate diagnostics system (from ref. 14).

Efficiencies:

1. Pinhole

$$\epsilon_1 = \frac{a^2}{16R_2^2}$$

2. Fluor

$$\epsilon_2 = (1 - e^{-ut}) \times 2 \times 10^{-2}$$

3. Optics

$$\epsilon_3 = \frac{d^2}{16L^2} \times \frac{1}{m^2} \times T$$

Realistic

$$10^{-10}$$

$$10^{-3}$$

$$10^{-2}$$

Camera Sensitivity ($S/N \sim 10$):

$$S = 10^{-10} \text{ J/cm}^2 \text{ frame}$$

d = lens diameter

m = lens magnification

T = lens transmission

Needed Brightness:

$$S_0 = S / \epsilon_1 \epsilon_2 \epsilon_3 = \begin{cases} 10^5 \text{ J/cm}^2 \text{ frame} \\ 10^{18} \text{ Mev/cm}^2 \text{ frame} \end{cases}$$

Figure 14. Schematic of a pinhole diagnostics system.

Figure 15. Block diagram of LASL's off-line digital image enhancement system (from ref. 17).

Figure 16(a). Schematic of fast fission chamber and associated electronics. The collector plates are shown connected in arrays of one, two, three, and four plates, for a total of four output channels.

Figure 16(b). Schematic of detector electronics showing binary circuits, rate sense detection and programmable input. The binary circuits permit high data rates to be recorded and transmitted at low bandwidths (data compression) while maintaining resolution.

Figure 16(c). Schematic of data readout system showing one channel of the hodoscope system being displayed on a T.V. screen. A split screen T.V. display could be used if two hodoscopes were being employed simultaneously.

Figure 17. Optical - digital processor block diagram (from ref. 20).

END

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