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SPERT PROGRAM REVIEW

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~~D. Nishimoto 8-23-85~~
~~Lee Plansky 5-8-91~~
~~ILL 10-4-91~~
~~6-23-93~~
~~Rusty Grant 2-6-95~~

W. E. Nyer
S. G. Forbes

October 19, 1960

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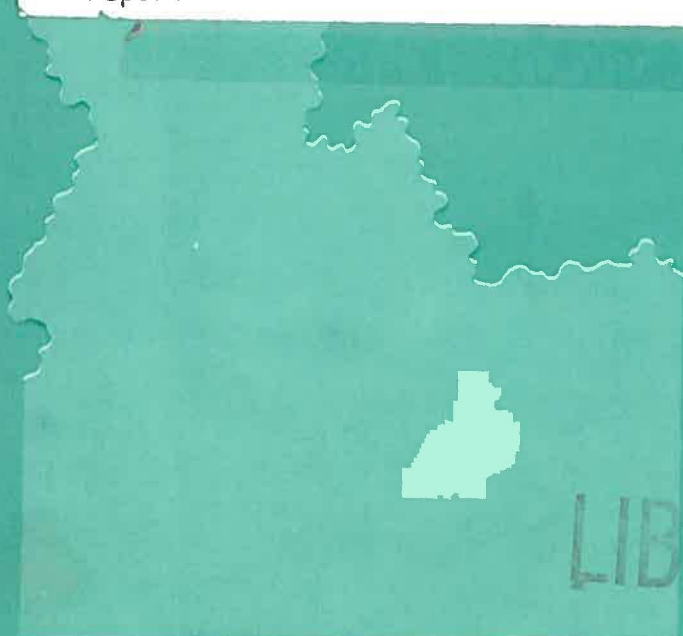

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SPERT PROGRAM REVIEW

W. E. Nyer and S. G. Forbes

Foreword

This report was originally issued as an internal document June 23, 1959, in response to a request from J. B. Philipson of the Atomic Energy Commission, Idaho Operations Office, to R. L. Doan of Phillips Petroleum Company. Its purpose was to provide a basis for determining the directions of the future program. Discussion was specifically requested on the origin and development of the Spert Program, summary of type of tests conducted, a summary of the theoretical advances made, and recommendations as to future program. At the request of the Idaho Operations Office, the report is being reissued with only minor changes as IDO-16634.

A list of references containing information supplementary to this report has been added in Appendix B.

PHILLIPS PETROLEUM COMPANY
Atomic Energy Division
Idaho Falls, Idaho
Contract AT(10-1)-205

IDAHO OPERATIONS OFFICE
U. S. Atomic Energy Commission

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I. INTRODUCTION AND SUMMARY

Recent developments in the Spert Program appear to warrant changes in the program direction superseding some of the plans suggested in IDO-16415. In order to provide a sound basis for the consideration of future activities, a review of the history and accomplishments of the program in its relation to the Atomic Energy Commission's Reactor Safety Program is desirable. For completeness of this report some of the comments of earlier reviews and proposals will be briefly repeated.

The Reactor Safety Program is centered on those problems associated with the operation of nuclear reactors which involve the possibility of extreme hazards. Related to these problems are three broad areas of investigation: Reactor Dynamics, Chemical Reactions, and Reactor Containment. The Spert Program is aimed at studies in which the reactor and its behavior are essential factors. Thus, it provides the means by which the investigations in the several areas may be unified and it is the main instrument for studies in the area of Reactor Dynamics. The problems usually considered in the latter area are Runaway, Plant Stability, and Afterheat. At the present stage of development of the program and the understanding of safety problems, Runaway is considered the major problem and will receive the most attention in this report.

The earliest large-scale experimental work on this subject was the Borax series of tests, whose principal objectives were related to the feasibility of operating boiling reactors but which included the study of the self-limiting properties of such systems when subjected to sudden and large additions of reactivity. The destructive test in 1954 essentially completed the portion of this program dealing with violent kinetic studies. However, the tests had clearly revealed that valuable safety information could be obtained by such methods and the Commission determined that this kind of investigation should not end with the completion of this test. Accordingly, at the request of the Atomic Energy Commission, Phillips Petroleum Company undertook to continue work of this sort with a broadened outlook. The emphasis in the new program was to be placed on the general study of reactor safety.

This work has since gone forward along the lines laid out in the original Phillips' proposal and developed in subsequent program reviews. Section II discuses, in some detail, the history of the Spert program in terms of the broad objectives and accomplishments. Much of the work initiated under these plans is still in the earliest stages and should continue without significant alteration. However, a shift in emphasis for future work in some portions of the program appears desirable because by the end of 1958 some of the initial objectives had been achieved. In particular, the specific task of determining the maximum step and ramp additions of reactivity that can safely be introduced into a few selected types of reactor cores has been completed. The analytical work carried out in connection with these experiments contributes a great deal to the most important initial objective, which was to obtain a thorough understanding of reactors under transient conditions. As a consequence, many general properties of excursions which apply to all reactors are now believed to be well understood and the need to explore general burst behavior is correspondingly lessened for the range of periods studied thus far. More attention can now be devoted to certain aspects of the program in progress which are deserving of greater emphasis, and to those phases of the work which, heretofore, have not been pursued.

Excursion studies will continue for the purpose of determining the dynamic reactivity coefficients, identifying specific shutdown mechanisms, establishing empirical safe limitations on steps and ramps for various core types and operational conditions, assessing the importance of specific factors in self-shutdown, and determining the particular form of the shutdown equations appropriate to various core-types.

The proposed program for the various reactors is as follows:

Spert II will follow the present plan of examining the influence of the prompt neutron lifetime and, secondarily, of studying the influence of the special factors connected with the use of heavy water. Provision has also been made for the study of the importance of direction of coolant flow through the core. Spert III will likewise proceed as planned to assess the importance of pressure and temperature, as well as other special factors to be encountered in power plant operation of boiling and pressurized water reactors. The planned initial use of Spert IV for investigation of self-induced oscillations will go forward. The major noticeable difference in the program on these reactors for the next two years, as a result of the change in emphasis, will be in the decrease of the number of transients required to obtain general kinetic behavior patterns and in the growth of detailed studies, including pile oscillator work and static measurements. It is of great importance to the overall objective of the work that the increased attention to specific shutdown mechanisms can be accompanied by definite efforts to improve shutdown characteristics by capitalizing on mechanisms which are not presently important.

For Spert I the proposed changes in the program are more extensive. Some of the specific factors to be examined for the general program, such as the effect of the delayed neutron fraction, have less urgency and can now be regarded as demonstrations of the state of our understanding rather than as explorations of new territory. Some factors, such as the dynamic coefficients, have become relatively more important and a survey of these should be initiated at an early date. An appropriate proportion of effort on static experiments will be required.

As previously mentioned, the program has contributed significantly to obtaining a thorough understanding of transient behavior over the range of conditions that have been studied, that is for excursions with periods longer than 5 msec. These tests were essentially non-destructive. Extrapolation of some of the general features of the results to excursions with shorter periods is possible but severely limited because additional phenomena of importance in self-shutdown are likely to become evident. It is to be expected that the magnitude of the dynamic shutdown coefficient will decrease in this region, possibly quite sharply. Metal temperatures may become high enough to make metal-water reactions a serious concern and shock waves may be generated. In contrast with the heavily-explored region of periods greater than 5 msec, there are virtually no experimental data in the potentially destructive period region below 5 msec.

It is in this region where all the ramifications of the overall reactor safety problems become important: Reactor Dynamics, Chemical Reactions and Containment. It is even more important to fully understand reactor transient behavior under these conditions than under non-destructive conditions. Destructive tests have always been viewed as appropriate to the overall objectives of the program but had been deferred because of the program dislocations that would have resulted from conducting such experiments with the only available facility. The completion of the exploratory phase of the Spert I program and the construction of other Spert reactors have alleviated this problem. Also, as a result of the Spert I experiments, the required planning can now be based on a firmer background of experience and understanding than initially existed. The early objective of determining the conditions dividing destructive tests from non-destructive tests has been accomplished for some cores and permits a degree of extrapolation to others. The next step to be taken should be directed toward answering the many questions related to destructive incidents. Accordingly, it is recommended that experiments in the destructive region be programmed for the Spert I reactor. The objectives would be twofold: to extend the understanding of reactor behavior to the region below 5 msec, and to understand the total reactor safety problem under circumstances representative of serious accidents under conditions which include Chemical Reactions and Containment as important features.

The theoretical work will continue to be directed toward the understanding of both specific physical shutdown processes and the general kinetic properties of reactors. Sufficient progress has been made in the analysis of the influence of various factors on reactor kinetics to permit the formulation of general safety criteria for reactor designs to be undertaken even though some analytical predictions and the dynamic values of the parameters still require experimental confirmation.

The technical accomplishments and consideration upon which these recommendations are based will be reviewed in Section III and the recommendations will be given in detail in Section IV.

II. HISTORY OF THE PROGRAM

As a result of the Commission's decision to establish a Reactor Safety Program, representatives of the Washington staff of the Commission, Idaho Operations Office, Argonne National Laboratory and Phillips Petroleum Company met at Argonne in August, 1954, to discuss possible continuation of excursion studies, with the emphasis to be on the general field of reactor safety. The Washington staff presented a memorandum(1) in which some views as to the nature of the program were suggested for consideration. The following objectives were set forth therein:

"(1) Conduct experimental and theoretical studies as necessary to obtain a thorough understanding of the behavior of reactors under transient conditions, with particular regard to phenomena which might result in an explosive release of energy, such as extreme pressure surges due to excessive rates of energy release, or possibly chemical reactions. Experiments with plate fuel elements in an open tank with and without forced circulation are of immediate interest.

(2) Determine experimentally the maximum equivalent step increase in reactivity and the maximum ... [excess] ... reactivity as a function of time rate that can be safely introduced into a few selected types of reactor cores. Appropriate theoretical studies should be a part of this program for planning of the experiments and analysis and interpretation of the results."

These objectives influenced all future planning and are evident in the following quotation from the initial instructions (September 15, 1954) to Phillips Petroleum Company to proceed with this work(2).

"The Reactor Development Division has defined the work to be undertaken as follows:

> 'The initial programmatic work should consist of designing and constructing an experimental reactor with aluminum clad fuel elements capable of withstanding forces associated with transient tests. Theoretical and experimental studies should be conducted to determine maximum equivalent step increase in reactivity and maximum excess reactivity as a function of time rate that can be safely added to this core and later theoretical and experimental studies should be extended to obtain a thorough understanding of transient phenomena which produce explosive release of energy as due to excessive rates of energy release or possible chemical reactions. Funds.....are being budgeted in FY 1956 to continue these studies with other types of fuel elements. In addition, the contractor will accept and be responsible for transient testing of reactors submitted by other organizations. A survey of requirements for such testing of other reactor cores will be initiated by Washington staff."

The substance of this directive and a later clarification(3) was that the Borax experiments with a strengthened aluminum core, and specifically including destructive tests, were to be repeated and extended and that a long-range program was to be formulated.

The planning of the long-range program was greatly aided by discussions at meetings of the Spert Advisory Panel established by Phillips Petroleum Company and composed of representatives of various reactor projects. One of the major contributions of this group was to assist in establishing the need and specifications for Spert III.

The initial program proposal for Spert I, submitted to the Idaho Operations Office by Phillips Petroleum Company on January 20, 1955(4), was approved(5) with the exception of the destructive test which was deferred due to the general feeling that it was too early in the program for such tests. By common agreement, the prototype testing had also been dropped from the plans because of limited value and interference with long-range work.

At about the same time, Phillips was requested(6) to submit a proposal for transient testing of the homogeneous reactors because of the intended transfer of the Kewb tests to the Spert Site. A downward revision in costs at the Santa Susana Site contributed to a later decision to continue the testing there(7). However, the Phillips' Proposal was completed and submitted on February 16, 1955(8).

By this time a feeling for the factors likely to be important in kinetic behavior had begun to emerge as is evident from the plans, noted in the above documents, to study the effects of temperature, pressure and starting power.

The first extensive statement of the experimental factors to be studied was made at the second meeting of the Spert Advisory Panel in Idaho Falls, May 16, 1955(9) with respect to the plans for a second Spert reactor. Part of this is quoted below.

"It is felt that ... [the main] ... experiments to do ... [were] ... exploration(s) of the variables affecting reactor behavior in order to determine a model. This work could be broken down into three groups: experiments on nuclear systems, experiments on hydrodynamic systems, and experiments on thermal systems. Topics under these programs would be (1) changes in flux distribution and their effects on nuclear and hydrodynamic systems (shutdown coefficient, peak power, peak pressure, and total energy would be of interest); (2) changes in neutron lifetimes, both in the reflector and in the core; (3) fuel plate spacing which changes the shutdown coefficient; (4) ramp rate studies (this involves the addition of Δk at a fixed rate); (5) cladding changes; (6) pressurization."

Concurrently, a theoretical, albeit elementary, framework for the experimental program began to take shape with the first application of Fuchs' work to Spert systems (Internal Report, July 1, 1955; reissued later as IDO-16393). It should be noted that this was completed before the experiments started. The recognized shortcomings of this approach were stated but refinement was not considered desirable until experimental results were available to establish the validity of the approach and to indicate the direction which efforts at refinement should take. Taken together, these statements of theoretical background and experimental factors form the framework of the entire subsequent Spert program. They were combined and presented in fuller detail in the extended program proposal submitted to the Idaho Operations Office in October, 1955[10]. The program as planned therein has been followed without essential deviation. The changes that did occur were in the nature of additions and shifts of emphasis rather than setting new objectives. Thus, the objective of establishing a long-range program was completed when Idaho Operations Office accepted and approved this proposal which provided for two additional reactors, Spert II and Spert III, for extending the range of experimental investigations.

In the meantime, Spert I had been designed, constructed and placed in operation by July, 1955, in accordance with the other established objectives.

Thus, within the space of a year the following things had been done to accomplish the objectives of the initial directives:

(1) the facilities for repeating and extending the Borax work had been provided;

(2) the experimental work was well underway;

(3) theoretical work to provide a framework for the future was completed;

(4) the experimental factors to be examined were explicitly stated;

(5) the long-range program was formulated;

(6) the design criteria for the additional facilities were established;

(7) the conceptual design for Spert III was completed;

(8) the technical staff had been expanded to eight people.

In retrospect this period may be viewed as one of preparation. The experimental program was then directed, as requested by the Commission, along the twin lines of specific studies on aluminum cores and general studies of reactor kinetics, which emphasized the study of factors likely to be important in kinetic behavior.

In the succeeding two years, from July, 1955 to September 1957, the construction of Spert II and Spert III was begun and considerable experimental data were accumulated with the Spert I cores. Under subcontract to Phillips, the Ramo-Wooldridge Corporation carried out theoretical studies devoted largely to problems of bubble formation and reactor stability[11]. In addition, some information about general properties of power bursts was obtained. Experimental work to check the stability calculations is planned.

With the accumulation of technical data the program entered a new phase. It became possible to more sharply define the specific objectives of the different facets of the program and to suggest new lines of investigation consistent with the overall plan. This was done in the program review, IDO-16415, submitted to the Commission in September, 1957[12]. The general problem of defining reactor safety was attacked and the relation of reactor kinetic studies to such problems was discussed in terms of the general kinds of tests to be performed. The specific technical advances were reviewed and new experiments proposed. The discussion of these points is too lengthy to reproduce here in full, but the essential program recommendations were two-fold. First, the established effort to study specific factors in self-shutdown was to be pursued with some additions to the scope of the program. Secondly, as a result of the Spert discovery of very large self-induced oscillations, the program was to be broadened to include stability studies. For the latter purpose a new reactor facility, Spert IV, was proposed and the pile oscillator program was to be initiated. The oscillator studies had the long-range objective of relating stability and excursion experiments. A number of test cores were proposed in connection with these objectives. At this time, approximately 50 Phillips personnel were engaged full-time in Spert work, including 10 from the Division Engineering Branch. Regular assistance of several people was supplied by the Theoretical Physics and Applied Mathematics Branch. An engineering subcontractor assigned from 7 to 10 employees to the NRTS for engineering work on Spert II and Spert III.

Subsequently, the test program largely followed the sequence proposed in IDO-16415 which is reproduced in Table I, but at a slower pace than indicated because of construction delays and increases in the experimental program.

Delays in construction of Spert II prevented the initiation of the program in late 1958 as planned. Instead, the work will start in late 1959. Similarly, construction delays moved Spert III work back to late 1958, but criticality was achieved December 19, 1958, and the indicated sequence was begun. Spert IV will probably not begin operation until the fall of 1960. Studies in the Spert I reactor with the variable plate spacing cores, the insulated A core, and the APPR core, were carried out as planned and the duPont service tests were completed. Pile oscillator studies were begun but the APPR core has been used initially for this purpose rather than the A core because of scheduling problems. The remaining items in the table are for the future and will be discussed in Section IV.

The staff requirements for carrying on the research, development, design and procurement efforts have continued to grow and at present 70 people are assigned full time to this work at the NRTS, including about 10 from Division Engineering. Computational assistance is also supplied by other groups. Additional staff will be required for full-scale operations of the Spert facilities.

The chief development in the program since September, 1957, was the formulation of a simple theory which correlated all of the step and ramp experimental observations and which contributed significantly to the understanding of kinetic properties of reactors in general. The variations in the specific shutdown mechanisms that will occur in other reactor systems are expected to alter the general behavior characteristics only to an extent readily predictable on the basis of a few measurements. The next phase of the program should emphasize the verification of the predictions of the models and the extension of the investigations to unexplored areas.

III. TECHNICAL REVIEW

A. Introduction

A brief summary of the technical work appears in Part D. (pp. 25-26) of this section, and those interested in the broader aspects of the program may well pass over the intervening, more detailed discussion.

In Parts B. and C., the technical work will be reviewed in considerable detail since the evaluation of the present status and the planning for the future course of the program are based on technical considerations. The results of much of the experimental work have been summarized in other reports and this phase of the program will be treated rather briefly. The theory has seen its greatest development more recently in the program, and for this reason these results have not been collected in a single document, although the details of most of the analytical work to be discussed are contained in Quarterly Reports. It is, therefore, deemed advisable at this point to dwell at some length on these analytical developments. The general implications will be discussed in the Analytical Section.

B. Experimental Work in Spert I

1. Step Tests

The initial experiments repeating and extending the Borax tests used a core similar to that used in Borax but more rugged mechanically. This feature was incorporated to permit tests to be performed in the unexplored region between the shortest (13 msec) period, non-destructive, subcooled Borax test and the 2.6 msec destructive test. The tests were also extended beyond Borax in the long period region. Several hundred transients of the step type were performed with the first core and the salient results can be summarized as follows:

a. The reactor behavior was very reproducible, and extrapolation of results from longer periods to shorter periods could be made with reasonable certainty so that tests of this type could be performed without undue hazard.

b. In spite of some design differences between Spert and Borax, the behavior of both reactors was essentially identical, indicating the general applicability of the test results to reactors of similar construction.

c. It was found that with the strengthened Spert core, 7 msec transients could be performed routinely with only minor mechanical distortions of the core, whereas 20 msec was the limit for the Borax core. Core damage at 5 msec was limited to modest distortions, and the onset of fuel plate melting could be predicted to occur at about 3.5 msec.

d. The 135 Mw-sec nuclear energy release observed in the Borax destructive test was in agreement with the extrapolation of the

Spert data, indicating that no anomalous nuclear behavior was associated with the destruction of Borax. Since the destructive effects from the Borax test appear to be compatible with a simple steam explosion of this energy, it seems unnecessary to postulate other energy release mechanisms such as a chemical reaction.

e. Extension of the tests to longer periods than those examined by Borax revealed a significant change in the dependence of peak power on the reactor period above and below prompt critical.

f. The self-limiting mechanism for periods longer than about 50 msec was definitely shown not to be steam formation. There is strong evidence that the longest period Borax subcooled transients were not shut down by steam. The fact that no apparent change in behavior occurred as the period was shortened from above 50 msec to below 50 msec and that the reactivity compensation per unit energy release was essentially constant for all periods, suggested a common shutdown process was responsible for the self-limiting behavior at all periods. Thus, for the first time, the hypothesis of steam shutdown in the Borax tests was open to question. The question was not one of whether steam was formed during a transient, but rather one of whether steam formation actually occurred prior to the peak of the power burst.

g. The reactivity change required to halt a power rise is appreciably less than that originally injected, for transients up to prompt critical and somewhat beyond.

h. As the initial reactor temperature is raised toward the saturation (boiling) point, the nuclear excursions become less severe. The mechanical manifestations, such as water ejection and transient pressures, become more pronounced, however, because of the enhanced formation of steam. As the initial temperature is increased, the reduction in peak power appears first in the short period region and moves progressively to longer periods as the saturation temperature is approached.

i. Increasing the hydrodynamic head by raising the depth of water over the core produced no measurable effect on the subcooled tests but increased the peak power and energy released in the boiling tests.

j. Modification of miscellaneous factors, such as the amount of dissolved gas in the water moderator or the surface tension of the water, produced no measurable changes in the behavior up to the peak of the first power burst, although some post peak changes were observed.

2. Ramp Tests

In addition to the step transients in Spert I, a number of ramp tests were performed in which reactivity was added continuously at a constant rate to the just-critical system, since this form of accident initiation is more typical of the types of accidents likely to occur in actual practice. The ramp tests were the first departure from the types of experiments performed in Borax. In these tests the parameters of particular importance are the initial power and the rate of reactivity addition (ramp rate). During a ramp transient the rate of logarithmic power rise, α, is initially zero, increases to a maximum, and then decreases to zero again at the time of maximum power. Two readily determined indices of ramp transient behavior are the maximum power and the maximum in the rate of logarithmic power rise, α_m. The ramp experiments on the first core included ramp rates from 0.01% Δk/sec to 0.35% Δk/sec and initial powers from 10^{-4} watts to 10^5 watts with the following results.

a. The initial reactor power is relatively unimportant in a ramp accident. Increasing the initial power by a factor of 10^9 reduced the peak power by only a factor of ten. This extremely weak dependence of ramp burst behavior on initial power indicates that, from the point of view of inherent safety, a startup source needs only to be large enough to eliminate problems arising from the statistical fluctuations in the initiation of a divergent chain. Thus, blind startup problems may be less severe than has generally been assumed.

b. The peak power is approximately proportional to the ramp rate, which makes the ramp rate the dominant factor in accidents of this type.

c. If a ramp burst is characterized by the maximum value of α, it is essentially equivalent to a step burst having the same value of α. The problem of establishing the relationship between step and ramp accidents is therefore reduced to that of finding α_m for a given set of ramp conditions. The analytical methods which have been developed for this purpose will be discussed in a later section. This relation between ramps and steps permits direct application of all the step data and analysis to the ramp case.

d. The highest ramp rate used, about 0.35% Δk/sec, was estimated to be an order of magnitude below that which would lead to core damage during the first burst.

e. Although in every case the first burst during these ramps was safely self-limiting, continued withdrawal of the rods led to violent power oscillations which rapidly grew to destructive proportions necessitating reactor scram to prevent core damage.

3. Instability Tests

A series of instability tests was undertaken to investigate more fully the violent oscillations observed during the ramp tests. These instability tests were conducted by injecting a predetermined amount of reactivity at a modest rate and observing the reactor power behavior after the injection was completed. For small reactivity injections the reactor operated stably at an equilibrium power determined by the reactivity added above the zero power critical condition. For larger reactivity injections unstable behavior developed and the reactor power went through a series of power bursts. About 50 instability tests were performed with the following results.

a. Large power oscillations appeared whenever the reactivity held in the form of moderator voids exceeded 1.5% Δk.

b. These power oscillations were an order of magnitude larger than those observed in Borax and frequently approached the largest bursts observed in the step tests. Thus, in some situations instability may constitute a serious hazard potential.

c. The moderator voids collapsed rapidly, and essentially completely, prior to each power pulse. The fractional collapse of voids in these tests appeared to be much greater than that reported from Borax.

d. The interval between bursts was about 0.3 sec for tests initiated from room temperature and about 0.9 sec for boiling tests.

e. The tendency toward oscillation increased with the amount of reactivity above zero power critical and with the depth of water over the core.

f. Reproducibility of detailed behavior was poor. The sequence of oscillations appeared to be random although some regularities were observed. In some cases sustained oscillations of roughly constant amplitude were observed. In other cases the oscillations died out for as long as half a minute and then reappeared. In several subcooled tests with 9 feet of water over the core the power peaks increased abruptly from sustained pulses of about 200 Mw peaks to erratic pulses exceeding 2500 Mw peaks.

g. The mode of reactivity addition was not responsible for these oscillations since essentially the same results were obtained by incremental rod withdrawals to the final position in which the reactor power was allowed to come to equilibrium between increments.

h. Long-term instability and the effect of the hydrodynamics of the system could not be satisfactorily investigated in Spert I.

4. Variation of Core Parameters

The experimental work summarized above was conducted on the Spert A core. From these tests the general kinetic properties of a reactor having a specific set of design characteristics were sufficiently well established to permit the next step in the program to be taken. This was to extend the investigations to the study of the effects of changing individual parameters believed to be important in kinetic behavior. Accordingly, the next phase of the experimental program included cores which would provide a wide range of metal-to-water ratio. The core parameters principally affected were the void coefficient and to a lesser extent the prompt neutron lifetime. Altogether, five highly enriched plate-type cores were used in the investigations: the A core previously discussed, three configurations of the aluminum B core which were made up from fuel assemblies equipped with removable fuel plates to permit variations in the metal-to-water ratio, and the P core composed of APPR-type stainless steel fuel assemblies. These cores ranged from undermoderated to overmoderated and represented a substantial portion of the spectrum of heterogeneous, research reactor core designs.

The experiments on the B and P cores were principally of the step type. However, some ramp studies were performed on one configuration of the B core for comparison with the A core results. Because of the limitations of the Spert I reactor vessel, the stability tests were restricted to a determination of the instability threshold for the most highly moderated configuration of the B core, since this represented the greatest departure from the static parameters of the A core. Although the transient work on these cores was much less extensive than for the referent A core, it was necessary to perform fairly detailed static measurements of the local and average void coefficients. The results of the experiments on all five cores are summarized below.

a. The average void coefficients for all cores were negative. The magnitude of the negative void coefficient was greatest for the undermoderated core and decreased uniformly with increasing atomic ratio of hydrogen to uranium. The overmoderated core showed a local positive void coefficient in the core center near the water channels for the control rods.

b. The prompt neutron lifetime increased with the increasing H/U ratio.

c. The step behavior for all cores was remarkably similar. Curves of peak power, energy release, reactivity compensation at peak, plate temperature at peak and pressure as functions of α were similar in shape. The principal differences between such curves for different cores were displacements on the α and amplitude scales.

d. The displacements on the α scale were consistent with the respective prompt neutron lifetimes for the various cores.

e. The amplitude displacements were consistent with analytical predictions of the effects of void coefficient and prompt neutron lifetime on burst behavior.

f. The minimum safe period for the stainless steel core for tests at ambient temperature was found to be about 5 msec, essentially the same as that found for the aluminum A core. The advantage of the higher melting point of stainless steel is largely offset by its poor thermal diffusivity which increases the temperature peaking in the center of the fuel plate. The 5 msec transient produced severe buckling and blistering of the fuel plates.

g. The maximum stable power for the most highly moderated aluminum B core was about 18 Mw, or about 80 kilowatts per liter. This represents an increase of a factor of three in thermal neutron flux and a fifty per cent increase in the heat flux over the A core.

h. The ramp test results were essentially the same as those for the A core with only those differences expected from the changes in void coefficient and prompt neutron lifetime.

5. The Pile Oscillator Program

The pile oscillator program currently under way is in the initial phase of the experimental measurements. The zero power transfer function for linear (small) oscillations has been obtained. This work will be extended to measurements at higher power levels where coupling effects will appear, and to large oscillations where the nonlinear nature of the kinetics equations becomes important. The principal objective of this part of the Spert program is to determine to what extent pile oscillator tests may be utilized to predict the dynamic behavior of reactors in accident situations.

6. Shutdown Mechanism Experiments

A few experiments have been performed for the specific purpose of investigating the physical processes responsible for the self-limitation of the Spert I reactor, particularly in the short period region. One set of experiments consisted of coating the Spert I A core with plastic in an effort to modify the heat transfer characteristics of the plates, and thereby to effect some isolation between processes which require heat transfer, such as transient boiling, and processes which do not depend on this factor, such as the formation of radiolytic gas. The results of transient tests on this insulated core were somewhat ambiguous, but indicated that either (a) heat transfer was relatively unimportant in the self-shutdown process, or (b) the plastic coating had not achieved the calculated thermal insulation of the plates. Preliminary tests in closed capsule experiments gave results exactly contrary to (a); that is, these tests indicated that heat transfer was an essential element in self-shutdown and that no appreciable shutdown effects could be attributed to processes not involving heat transfer. The capsule experiments appear to be unambiguous, and more detailed experiments of this type are currently in progress in an effort to resolve the conflict in results. The Spert III reactor will also provide an opportunity to investigate the role of transient boiling in reactor self-shutdown by pressurizing the system to suppress boiling.

C. Analytical Work

1. General Considerations in Transient Behavior

The ultimate purpose of the theoretical work in reactor kinetics is to develop analytical methods for the prediction of reactor dynamic behavior under a given set of conditions from a knowledge of the system design. It is equally important that the influence of specific design parameters on dynamic behavior be sufficiently explicit in the analysis to permit a quantitative evaluation of the effects of design modifications on the safety and operability of the reactor. The latter point is emphasized here because it has to a considerable extent determined the analytical approach to reactor kinetics problems at Spert.

Before considering specific aspects of the theoretical effort, it is useful to discuss some general properties of the reactor kinetic equations. For the present purpose, the reactor can be considered as a lumped-parameter system and the kinetic equations may be written as:

$$\frac{dn}{dt} = \frac{k-1-\beta}{\ell} n + \sum_i \lambda_i C_i \tag{1}$$

$$\frac{dC_i}{dt} = \frac{k\beta_i}{\ell} n - \lambda_i C_i \tag{2}$$

$$k = f(s) \quad , \tag{3}$$

where the common symbols have their usual meanings and f(s) is used to express the fact that the reactivity, k, is a function of the state (s) of the system. In general, the state of the system will be determined by external factors such as the position of the control rods and by internal factors such as the core temperature. These internal factors are frequently affected by the reactor behavior and in this manner coupling is introduced between the reactivity, k, of the system and the reactor power, n. The solution of a problem in reactor dynamics thus consists of finding the proper form for f(s) including the coupling, when it exists, and solving the resulting equations (1), (2) and (3) for the power behavior as a function of time. From this solution other quantities, such as transient pressure and fuel plate temperature, can often be found.

The simplest dynamic problems are those in which coupling can be ignored and the reactivity of the system is determined entirely by the external factors. Three familiar examples of problems of this sort are the following.

(1) The reactivity is suddenly changed from critical to a constant amount above critical. A short time after such a change, the reactor power rises exponentially with a period given by the Inhour Equation.

(2) Reactivity is added to the initial critical reactor at a constant rate (the ramp rate). Approximate solutions to this problem have been used. The first of these attempts, the Newson equation, is still in frequent use in hazards reports to describe startup accidents by predicting lower bounds for reactor periods and upper bounds for energy releases.

(3) The reactivity is varied sinusoidally above and below criticality. The amplitude and phase of the resulting power oscillations as functions of the amplitude and frequency of the reactivity oscillations constitute the zero power transfer function for the reactor.

These cases are useful in describing the limited class of accidents which are assumed to take place without self-shutdown effects occurring, and which are ultimately controlled by the reactor control system. In a sense this is merely a way of attempting to design an adequate control system, but generally the ignoration of shutdown effects severely limits the achievement of even this objective. Nevertheless, it is this restricted approach that is often employed in hazards discussions.

More properly, the consideration of accidents should take into account failures of the control systems. In such accidents coupling cannot be ignored. The problem is then to write the appropriate form of the coupling equation and to solve the resulting kinetic equations.

There are two approaches to the problem of writing the coupling equation. In the first approach, specific mechanisms are postulated at the outset for the flow of energy and for the generation of reactivity effects. In principle, the calculations of reactor kinetic behavior through the coupling equations (3) and the set (1) and (2) is then straightforward. The second approach permits separation and postponement of the question of mechanisms by assuming a methematical form for the coupling equation. Again, in principle, the calculations are straightforward. Actually, even this approach can be interpreted as being based on a particular mechanism as a limiting form.

The latter approach has the advantage that it may suggest the mechanisms which should be investigated when a correlation has been achieved between the equations and the experimental data. The influence of various factors is more easily discernible in this approach.

In either case, the reactor characteristics that were required in the description of the reactor behavior when only the kinetic equations (1) and (2) were considered, are no longer sufficient when coupling is included. Consideration must now be given to the two-part problem of first determining changes in the physical state of the reactor from the power history and, second, relating these state changes to reactivity changes. To date, the first part of the problem has been considered to be the more difficult but it may prove equally difficult to correctly predict the effective dynamic reactivity coefficients and to relate them to the static coefficients.

The major factors of importance in this view of the kinetic problem appear to be the following:

(1) the coupling equation, which may be taken as a starting point or as a consequence of,

(2) the machanisms by which reactivity changes are generated, and,

(3) the sign and magnitude of the reactivity coefficients associated with the various mechanisms.

Each of these factors--coupling, mechanisms and coefficients--can be expected to be of importance in all reactor systems. However, the important mechanisms and the sign and magnitude of the reactivity coefficients will vary greatly from reactor to reactor. Nevertheless, the mechanisms and ranges of the values of the coefficients will probably be characteristic, in a broad sense, of each class of reactor just as each class of reactor has, again in a broad sense, characteristic critical masses and prompt neutron lifetimes.

One level of "understanding" reactor kinetics is represented by an empirical correlation between experiment and theory, based on assumed forms of the coupling equations, but the degree to which the understanding can be extrapolated to new situations is limited. Clearly, the degree to which understanding is more than an empirical correlation is determined by the degree to which an understanding of the mechanisms exists and by the ability to calculate the coefficients. That is, the highest level of understanding reactor kinetics rests on the level of understanding of reactor statics.

These features of the major safety problems were well known, but perhaps had not been explicitly stated, at the time the Borax tests were initiated in 1953.

2. Step-Induced Bursts

Early theoretical attempts(13) to describe the Borax excursion behavior were based on the choice of a specific mechanism of shutdown. There was, at this time, widespread conviction that this mechanism was steam formation(14). In addition to this limitation to a single specific mechanism, these theoretical attempts ignored feedback effects by omitting a coupling equation.

The coupling was replaced by a cutoff time for the burst which was, in turn, based on the accumulation of sufficient steam voids to compensate for the injected reactivity. Peak power could be brought into agreement with the experimental data by proper choice of fitting constants. The ease with which this fit was obtained for models that represented very different pictures of the underlying mechanisms raised the question as to the appropriateness of these approaches because all of the theories could not be correct. Yet, there was no means of choosing between them on the basis of the data that existed at that time. At the very least, it appeared that the results were not extremely sensitive to the assumptions. In addition, the predictions of the other quantities such as temperature and pressure were not satisfactory.

A similar approach based on somewhat simpler assumptions regarding the physical process responsible for reactor self-shutdown was used at Spert. This model, called the "Conduction Boiling Model", was found to give a good correlation of the peak power, energy release and fuel plate temperature for short-period transients in six cores (the Borax core and five Spert cores) under both boiling and subcooled conditions, with only one arbitrary constant which was adjusted to give the best fit for a given series of transient experiments. Although these correlations were better than those mentioned above, the model was still empirical in nature and, hence, suffered all the shortcomings noted above.

It was felt at the outset of the Spert theoretical program that a more profitable line of attack on kinetics problems would be an analytical investigation of the reactor behavior which would result from various forms of the coupling equations, without specifying in detail the specific reactor properties which might lead to a given coupling equation. This approach is essentially one of investigating the general properties of the kinetics equations under a wide variety of assumed conditions. Once these general properties are understood, it is relatively easy to describe fairly accurately the dynamic behavior to be expected from a particular reactor by consideration of the coupling equation which is appropriate to that particular system. The two main advantages of this approach are that the results can be applied to a wide range of reactor types and that the dependence on various reactor parameters is obtained explicitly. The primary disadvantage is that the detail of the physical processes responsible for these dynamic properties is not contained in the analysis and, hence, other techniques must be employed to investigate these processes in order to select the appropriate form of the coupling equation for a given reactor. Nevertheless, it was felt that the development of the general understandings of reactor kinetics should proceed without waiting for a complete and detailed investigation of the internal processes in any particular system.

The first analytical work undertaken along the lines described above was an investigation of the Fuchs Model. This model had been found to describe burst behavior for a fast assembly very well. Furthermore, it had the virtue of great simplicity and feedback was specifically included.

The Fuchs equations consist of the prompt approximation to equation (1) in which the summation term is neglected (hence, equation (2) is not applicable) and a coupling equation in which the reactivity changes resulting from internal factors are assumed to be negative and proportional to the energy released by the reactor. The coupling equation permits two types of shutdown to be represented; one in which the shutdown effects due to energy release begin to appear immediately, and the threshold case in which a fixed amount of energy release is required before shutdown effects appear. The solutions to the Fuchs equation for step transients lead to the following predictions.

For the case without a threshold:

(1) the power excursion is a round-topped burst which is symmetrical about the peak power;

(2) the peak power increases as the square of α, the reciprocal of the initial asymptotic period;

(3) the energy release increases linearly with α;

(4) the peak power and energy for a given α should vary linearly with the prompt neutron lifetime divided by the void coefficient;

(5) numerical solutions of similar equations including delayed neutrons exhibited a break in the peak power vs α curve at prompt critical.

For the threshold case:

(6) raising the threshold level makes the power burst increasingly more asymmetrical (the power decrease after the power peak is more rapid than the initial increase);

(7) the dependence of peak power, energy and temperature rise on α is decreased.

Comparison of these predictions with the experimental results leads to the following observations.

(1) The Fuchs equations predict the observed behavior of peak power, temperature rise and energy release as functions of α quite well in the short period region.

(2) Inclusion of delayed neutrons leads to results in qualitative agreement with experimental results of the long-period transients.

(3) The predicted power burst shape for both the threshold and nonthreshold cases is a very poor representation of the observed burst shapes for short-period transients.

(4) The observed peak power and energy release at a given α vary more nearly as the square root of the prompt neutron lifetime divided by the void coefficient rather than the linear dependence predicted by the Fuchs Model.

(5) The values of the shutdown coefficient which enter the model are very close to observed experimental values.

From these observations it is concluded that the Fuchs Model is a good guide to transient behavior characteristics but fails to describe the system in detail. The latter is not surprising in view of the simplicity of the model compared to the relative complexity of the shutdown processes in a heterogeneous reactor.

Some of the experimental results suggested that a somewhat more general form for the coupling equation might be more successful in matching the experimental results. Accordingly, a model referred to as the "Empirical Model" was developed in which the negative reactivity effect was assumed to be proportional to the n^{th} power of the energy release. Provision was also made for the inclusion of a fixed time delay between the energy release and the appearance of reactivity effects. The exponent n was an arbitrary constant in the analysis and, hence, it could be used as a disposable parameter in matching the experimental results. For the case of zero time delay and n equal to unity, the Empirical Model degenerates to the Fuchs Model. Analytical solutions to the Empirical Model for step transients were obtained for two limiting cases which were zero time delay and a long time delay compared to the reactor period. The results of this analysis were the following.

(1) The long delay model with n equal to about two was found to give a very good match to the experimental short period power burst.

(2) The agreement between the predicted and observed peak power and energy release as functions of α found for the Fuchs Model could be preserved in the long delay Empirical Model.

(3) The observed square root dependence of peak power and energy release on the prompt neutron lifetime divided by the void coefficient was predicted by the long delay model with n equal to two.

(4) A wide variety of burst shapes ranging from very broad to very sharp could be represented by the model by adjustment of the constant, n.

Thus, a simple modification of the coupling equation was found to correct the major deficiencies of the Fuchs Model, while retaining the virtue of simplicity. The model can be applied to any reactor system for which the power burst can be reasonably well matched by selection of the parameter, n. Once the best fit value of n is found, the functional relationships between other variables are obtained immediately from the model. Basically, the model provides a set of relationships between various dynamic properties of a reactor so that if one property is known to exist, others may be predicted.

3. Ramp-Induced Bursts

The Fuchs prompt approximation can be modified for ramp transients by replacing the constant reactivity in the step case by an external reactivity change which increases linearly with time. An approximate solution for the power behavior can be obtained for those cases in which the initial power is relatively low. The results from this solution are in good agreement with the experimental data for those ramp bursts which reach prompt critical before self-shutdown occurs, but for slower bursts the agreement is less satisfactory because of the neglect of delayed neutrons in the prompt approximation. The inclusion of delayed neutrons, even by approximate methods, removes this defect at the expense of some loss of simplicity. However, in many cases the more severe transients are the primary concern in safety considerations and the simpler prompt approximation can be used to advantage.

The analytical results of primary interest are as follows.

(1) The maximum reciprocal period, α_m, varies as the square root of the ramp rate multiplied by a logarithmic factor. This factor involves the initial power and the shutdown coefficient, but is so weakly dependent on these parameters that it may be treated as a constant over a very wide range. The major term is therefore the ramp rate, which is in agreement with the experimental data.

(2) The peak power is proportional to the ramp rate divided by the shutdown coefficient multiplied by a logarithmic factor involving the initial power. This factor is again slowly varying and the dominant term is the ramp rate, as was found experimentally.

(3) The dependence of the peak power on α_m is the same as that for a step transient except for a correction factor, which is the ratio of two slowly-varying logarithmic factors and which has a value of about 1 to 2. This is in agreement with the experimental observation that a ramp burst can be treated as a step burst of equivalent α.

(4) The inclusion of delayed neutrons in the analysis leads to good agreement between theoretical and experimental values for α_m for all the ramp transients performed. This, in conjunction with Item (3) above, essentially completes the analytical correlation between ramp and step transients. The only other quantity needed to complete the analysis is the effective dynamic value of the shutdown coefficient.

(5) The power at which α_m occurs is, over a large range, independent of the starting power. This agrees with observation.

(6) The modification of the shutdown equation that was made in the Empirical Model to include nonlinear shutdown effects in step transients also produces improved agreement between calculated and observed burst shapes for ramps. The predictions stated above remain essentially unchanged, except for a weakened dependence of peak power and energy on the shutdown coefficient.

These results place the understanding of ramp accidents on an equal footing with the more readily analyzed step accidents. Because of the established relationships between ramps and steps, the general considerations previously discussed for the step analysis apply to ramps as well. It is important to note that for accident considerations in which the inherent self-limitation of the reactor is the dominant factor, the maximum tolerable accident will often be above prompt critical and the use of the prompt approximation is justified.

4. Systems with Positive Coefficients

Many reactor systems will exhibit internal changes which cause the reactivity to increase during the initial part of a runaway. Eventually the reactivity changes must become negative with increasing energy release if for no other reason than that of violent disassembly of the core. More practical situations arise in which a reactor has, for example, a temperature coefficient which is positive at room temperature but becomes negative at some higher temperature. The solutions to the kinetic equations for coupling equations of this form have been investigated for certain specific cases, and the following general observations can be made.

a. If the maximum internal reactivity increase accruing from the positive coefficient is added to the external reactivity step which initiates the accident, the resulting burst can be found by the usual step analysis, using this total excess as the initiating step. Since the internal reactivity contribution is a fixed property of the reactor, this represents a lower limit to the size accident that can occur. If this minimum accident is worse than the maximum tolerable accident, the reactor must be regarded as inherently unsafe. If the reactor is safely self-limiting for the minimum accident, then safe excursions initiated externally may be possible. If the energy release from the minimum accident is very much smaller than the safe limit, the existence of the positive coefficient may be ignored, and the reactor may be treated as one having a negative (but not necessarily constant) reactivity coefficient.

b. A reactor with a variable reactivity coefficient which is initially positive may be safer than one with a constant negative coefficient when consideration is given to the amount of flexible reactivity which must be provided in the control rods in order to overcome reactivity losses at the operating power. A system with the reactivity coefficient changing from positive to negative will possess a stable non-zero operating power with no investment of flexible reactivity in the control rods, and the maximum internal potential excess reactivity will, in general, be much less than would be required in control rods in order to operate a constant negative coefficient reactor at the same power with the same degree of self-stabilization. Thus, the existence of an initially positive coefficient is not, per se, a disadvantage and may indeed be desirable in some cases.

5. Reactivity Behavior for Power Bursts

In the foregoing analyses the form of the coupling equation was arbitrarily chosen and the resulting power burst found by solution of the differential equations (1), (2) and (3). An alternative approach is to select an analytical form for the power burst and solve the resulting equations for the reactivity compensation at the time of peak power, $k_c(t_m)$, which is the amount of internal reactivity change required to stop the rise of power and, hence, produce a self-limiting burst. For transients above prompt critical the compensated reactivity at the time of peak is simply the prompt excess; i.e., the reactor must be brought down to prompt critical to halt the power rise. For very slow transients in which delayed neutrons are essentially in static equilibrium with the prompt neutrons, the compensated reactivity at time of peak is the total excess reactivity; that is, the reactor must be brought to delayed critical to halt the power rise. For intermediate transients the situation is complicated by the time-varying contributions from delayed neutrons which are not in static equilibrium with the prompt neutrons. Since the plot of $k_c(t_m)$ vs α, as derived from the experimental data, showed considerable structure in the transition region below prompt critical, the question arose as to how much of this behavior was due to the properties of delayed neutrons and to what extent the nature of the self-shutdown process might be revealed by data of this sort.

To facilitate the analysis of this problem the power burst was represented by an approximate form consisting of the difference between two exponential terms. By adjustment of an arbitrary constant, this two-term approximation could be made to produce a wide variety of burst shapes ranging from a very narrow burst in which the power rose exponentially all the way to the peak, to a broad round-topped burst similar to that given by the Fuchs equation. The narrower bursts are characteristic of those given by the Empirical Model for nonlinear shutdown effects. Thus, by the use of the two-term approximation the general dependence of the $k_c(t_m,\alpha)$ function on the type (i.e., degree of nonlinearity) of shutdown process involved can be established. The approximate form for the power burst is used in preference to other forms because solutions containing the explicit dependences on the various factors in the equations can be obtained analytically.

The analysis carried out along the lines indicated above revealed the following general points.

(1) The shape of the $k_c(t_m,\alpha)$ function is dependent on the delayed neutron half-lives and abundances, the prompt neutron lifetime, and the degree of nonlinearity of the shutdown process.

(2) The agreement between the Spert I experimental data and the analytical results obtained by using the Spert I constants is very good.

(3) The reactivity change needed to halt the power rise for transients having intermediate periods can be several orders of magnitude less than the initial reactivity increase, if the change can be made in a small fraction of a period. Continued reactivity reduction will be necessary to prevent further power increase but the required reduction rates are not excessive.

(4) Because of the intimate relationship between reactivity change, energy release, peak power, and

peak temperature, most of the Spert I results can be predicted and understood from consideration of the behavior of the $k_c(t_m,\alpha)$ function.

(5) These results can be readily extended to other reactors to good advantage in predicting general burst behavior. Although the Spert results are shown to be typical of many reactor types, significant differences are to be expected in some cases, depending on the various factors mentioned in (1) above.

(6) The results have important implications in control system design, the design of pulsed reactors and general safety considerations. For example, it can be shown that some reactors may be safely self-limiting for small reactivity steps, be unsafe for somewhat larger steps, but have another region of safe behavior for steps in the vicinity of prompt critical. That is, there may be four separate zones characterized by their safety; a small-step safe zone, an intermediate unsafe zone, a "prompt"-step safe zone, and a large-step unsafe zone. The Godiva reactor is of this type. In the Spert and Borax reactors these upper and lower safe zones overlap and there is no intervening unsafe zone.

6. Shutdown Mechanisms

The theoretical work discussed up to this point has been concerned primarily with general properties of reactor systems and the burst behavior associated with these properties. Some consideration was given to the effects of nonlinear shutdown processes, and it was shown that the existence of such processes could be inferred from various aspects of the experimental data. However, for greatest utilization of these theoretical results in the evaluation of particular reactor designs, it is necessary to obtain a detailed understanding of the exact nature of the various self-shutdown processes.

As a starting point in the investigation of the shutdown processes in Spert I, the reactivity changes due to moderator heating by conduction from the fuel plates and due to expansion of the fuel plates themselves, were calculated from the observed power behavior at different values of α. The use of the two-term burst approximation makes it possible to solve the thermal diffusion equation analytically, and the distribution of heat energy can be calculated with considerable accuracy. The resulting reactivity changes at the time of peak power were compared with the $k_c(t_m)$ data to see to what extent these observed changes could be accounted for by straightforward calculations. These calculations included such factors as nonlinear expansion of the moderator, direct moderator heating by neutrons and gamma rays, and the effects of nonuniform distribution of density changes in the core, with the following results.

(1) Plate and moderator expansion account for all of the observed reactivity changes in the intermediate period region.

(2) For long-period transients the calculated effects are smaller than the observed reactivity changes. This difference could be due to the appearance of radiolytic gas in slow transients.

(3) For short-period transients the calculated effects are again lower than the observed reactivity changes. The reactivity contribution from steam, as calculated from the conduction boiling model, can be adjusted to close the gap between calculated and observed effects for all short-period transients. This means that, even though the conduction boiling model contains an arbitrary constant, the model predicts the correct form for the dependence of steam formation on the reactor period.

(4) The reactivity contribution from moderator heating is the dominant factor in long-period transients but declines to insignificance for short periods.

(5) The reactivity contribution from plate expansion is small for long-period transients, increases to become the dominant factor at prompt critical, and decreases in importance relative to steam for short-period transients. At a period of ten milliseconds the shutdown effect from plate expansion is still about one-third of the total effect. If all other shutdown effects were absent, the plate expansion alone would be sufficient to safely self-limit the reactor for periods as short as twenty milliseconds.

(6) In spite of the shift in the predominant mechanism from moderator expansion to fuel plate expansion, and then to steam formation as the period is shortened, the reactivity change per unit energy released remains essentially constant for all periods. This fact is undoubtedly responsible for the success of the simple Fuchs Model in predicting the observed behavior.

D. Summary of Technical Work

The experimental work on the first Spert I aluminum core included step, ramp and stability experiments in sufficient detail to permit a general evaluation of the properties of the system for various types of accidents. The initial objectives of repeating and extending the Borax experiments and the comparison of ramp and step accidents were essentially fulfilled by these tests. The extensive data on this core also served as a reference point in determining the influence of various parameters on dynamic behavior. Experiments on a total of five cores having widely differing void coefficients constituted the first phase of the general program of determining the effects of individual reactor parameters.

The analytical work on ramps has provided an accurate and generally applicable method for relating ramp accidents to step accidents so that the two situations can be treated as a single problem. Thus, there is no reason for not placing the treatment of accident initiation on a realistic basis by incorporating the reactivity addition rate explicitly into hazards analyses. The analysis of step transients has provided simple mathematical models which not only correlate most of the features of the experimental data on the five cores tested, but which also predict the behavior characteristics of reactors of many other types. Since these models reveal the influence of various reactor parameters explicitly, the general burst properties of a system can be stated at the outset if these parameters are known. Conversely, the nature of the internal processes and the corresponding dynamic properties of a reactor can be inferred with considerable accuracy from the observed behavior of a few transient bursts. This means that the applicability of many of the experimental and analytical results should extend beyond the class of heterogeneous water moderated reactors.

When the implications of the analytical results have been more fully digested, it should be possible to make a significant contribution to the ultimate objective of formulating general criteria for the evaluation of reactor safety. For greatest utility, these general criteria should be expressible as approximate relationships which are simple enough to be committed to memory, even if more elaborate calculations may be required for maximum rigor. Progress has already been made in this direction.

IV. FUTURE PROGRAM

A. Introduction

Although the development of mathematical models will be a continuing part of the program, the discussions in Section III show that there is even now a sufficient background of analytical understanding of reactor kinetic behavior to suggest a change in emphasis in the experimental work. The need for investigations of general behavior characteristics on a broad scale is reduced and the experimental work can profitably be directed toward confirmation of those predictions of theory which are outside the scope of past experimental experience. In this regard it now appears that the influence of certain factors, such as the delayed neutron fraction, are sufficiently well understood to permit reliable predictions to be made, and the experiments originally planned to investigate these factors may be postponed or eliminated. This development and the completion of new facilities make it possible to undertake two extensions of the program at an earlier date than had formerly been anticipated. It becomes appropriate as a part of the study of factors of importance in general kinetic behavior to extend the range of investigations into the unexplored region of very short periods. At the same time a considerable effort should be devoted to the investigations of the details of all shutdown mechanisms likely to be of importance in reactor safety, so that the program will not continue to be oriented so strongly around the water-moderated systems, but will have greater applicability to other important reactor types as well.

These suggested activities in combination with the substantial portions of the present program which need not be altered form the basis of the experimental program proposed in this report.

B. Outline of Future Program

Briefly, it is proposed that the immediate program should continue with the studies of the instability phenomenon, the detailed investigation of self-shutdown processes, and the checking of theoretical predictions of dynamic behavior. As quickly as possible, increasing emphasis should be placed on extension of the tests into the destructive region, measurements of effective shutdown coefficients for representative reactor types, the development of acceptance testing techniques for the measurements of dynamic properties of reactors in situ, and the extensive application of the results.

The suggested experimental approach to these general objectives is outlined below in greater detail and the proposed sequence of tests for each of the four facilities is shown in Table II along with the core type and major features of the various tests. Table III lists some of the specific experiments in each test series.

It is convenient in discussing the tests to group them by reactor:

<u>SPERT I</u> - The completion of the Spert-II, -III, and -IV facilities will provide more satisfactory means for conducting many of the

experiments heretofore carried out in Spert I. Consequently, the Spert I work can be concentrated on those experiments for which it is best suited but which, in the past often had to be deferred because of the heavy test schedule. Examples of these are preliminary explorations, mechanism studies and destructive tests. As a part of the necessary preliminary experiments for the destructive tests, subassembly experiments will be performed which will provide useful meltdown information. The mechanism studies can, for the most part, be carried out in capsule or subassembly tests. Hence, any suitable core can be used and the classification of tests by core type, which was used in IDO-16415, is no longer pertinent.

The first group of tests shown in Table II are the capsule and pile oscillator experiments with the APPR core. These experiments and their objectives are described in Section III. They will be followed by ramp tests and a limited survey of the stability properties of the core. These, along with the earlier step and oscillator tests will provide complete dynamic studies by excursion methods and by oscillator methods. The analytical work of comparing excursion and oscillator experiments should provide information which will be useful in assessing the value of oscillator tests for predicting reactor behavior under accident conditions.

The full test program with the ORNL BSR-II core, involves experiments at ORNL by the ORNL staff and tests at Spert by Phillips' personnel. The latter tests have several objectives. From the point of view of the Spert program, the most important objective is the checking of certain predictions of the theory, described in Section III, that have important consequences in the philosophy of design of control systems. The tests will also provide information for the basic program. To a large extent, this program is a proof-test of a core- and control-system designed to be an improved swimming-pool prototype.

Following the BSR-II tests, it is proposed to use the APPR core in another test series culminating in a destructive test. The objectives of such tests, which were discussed only briefly before, are manifold. In this connection it is important to state several generalizations. First, the period at which the fuel plate just melts will in general not vary enormously, primarily because the energy content per unit volume required to melt materials used in reactor construction is relatively constant. Second, as pointed out in a recent presentation[15], these "melting periods" are in a range where marginal gains in the ability to withstand shorter periods yield significant gains in safety. This is because the majority of accidents producing such periods will, in fact, be ramps, and the ramp rates required to produce such periods become unattainably fast, or nearly so. Finally, on the basis of the one destructive test that has been conducted, it is generally believed that in heterogeneous water-cooled reactors core-meltdown on a large scale is, one way or another, responsible for core destruction. Hence, the periods immediately below the 5 msec limit previously explored are of particular importance in the safety of these reactors, and essentially no experimental data exist on behavior in this region.

It should also be observed that much of the expected damage in this period range may be as much a consequence of the low boiling point and high vapor pressures of the water coolant as it is a consequence of the melting point of the materials of construction. For some other reactor designs fuel melting may not present a problem. It is, therefore, desirable to determine which features of behavior in this region are determined mainly by the behavior of the shutdown mechanisms, and which features are consequences of the meltdown. This will be an objective of the destructive program.

The physically attainable rates of reactivity insertion and the nuclear limitation on the reproduction factor effectively limit the attainable periods and in consequence may limit the magnitude of possible energy release, which would have an important effect on the whole problem of reactor safety. Hence, another objective of the series of tests would be to determine the energy release as a function of period. Furthermore, the ultimate problems of reactor safety are concerned with the magnitude of the consequences of incidents, and another objective would be to determine the scope of the problem that really exists when a destructive runaway occurs. Finally, these last problems involve all the broad problem areas of reactor safety, and these tests will provide means of bringing the different activities together under realistic conditions. Tests pertinent to the Chemical Reaction Studies and Containment Studies can be combined with the reactor tests. For example, the information obtained on the time-scale for energy-release during destructive tests will be valuable to Containment Studies.

The low enrichment ceramic core suggested for study after the APPR destructive test is intended mainly to provide a system with a long thermal time constant as an aid in the general understanding of shutdown mechanisms. The tests should be carried to destruction as a part of the program. A possible core for this use would be the critical assembly for the N.S. Savannah, if a sufficient portion of it could be made available. This single core would replace the 20% enriched and Borax IV cores previously planned as separate tests.

The proposed sequence of tests should be regarded as flexible because scheduling considerations and the detailed aims of individual tests may require a change in the order. As mentioned earlier, it no longer appears necessary to include a Pu or U-233 core to study the effect of changing the delayed neutron fraction.

Several general features of the Spert I tests should be mentioned. The predictions of the theory should be checked in more detail by experimental work. This may involve experiments on cores quite different from those previously included in the program. For example, systems with positive coefficients should be examined. Consideration should be given to the use of reactors other than those at Spert for tests of less hazardous nature in order to obtain data on a wider variety of reactor types.

Specific shutdown mechanisms should be investigated in detail, and methods for predicting and measuring dynamic coefficients should be developed in order to provide a sound basis for application of the general theory to many reactor types. It is particularly important in this connection to emphasize that the past work on mechanisms has concentrated on those important in current designs of water-moderated systems. The feasibility of using other mechanisms should be investigated.

SPERT II - The objectives for the Spert II work are unchanged from those given in IDO-16415. It is intended primarily to examine the importance of the prompt neutron lifetime in kinetic behavior over a wide range of environmental conditions, and secondarily, to study the special features associated with heavy water systems. The early part of the program is unchanged from that given in IDO-16415 except for the additions of plans to study positive coefficient cores. The Zr cores proposed for later in the program will not be needed. At present it would appear that the only unique contribution from the use of a Zr core would be to provide data on the mechanical properties of Zr fuel in severe transients. This information can probably be obtained equally well in subassembly tests at greatly reduced cost.

SPERT III - The objectives of Spert III are also unchanged. This system is intended to provide an environment similar to that expected in high power reactors and to provide means for certain engineering tests. No program changes are envisioned other than the elimination of the Zr core for the same reasons given for Spert II.

SPERT IV - As stated in IDO-16415, the initial objectives of Spert IV are to study the instability phenomenon, and this will proceed as planned. Some step and ramp tests will be included but these will be of secondary importance and mainly for orientation purposes.

The experimental sequence and approximate scheduling for the next few years are given in Table I. Additions or deletions from this sequence may be expected in light of past experience, but the general course of the work should not deviate greatly from that indicated. The program selected for the next two years is intended to provide for the most rapid development of new information in reactor kinetics.

TABLE I

SPERT EXPERIMENTAL PROGRAM
(From IDO-16415)

1957 1958 1959 1960

SPERT I

Core	Experiments
B Variable Spacing	Effect of Void Coefficient and Hydrodynamic Effects
A (Insulated)	Prompt Shutdown Effects
B Pile Oscillator	Transfer Function Measurements
duPont	Service Tests
APPR SS	Change in Thermal Characteristics of Fuel Plates
BSR SS	Swimming-Pool Type Safety Test of Core and Control System Basic Characteristics
20% Enriched	Effects of Resonance Neutron Capture
Plutonium or U^{233}	Change of Delayed Neutron Fraction
Ceramic Core Borax IV Type (Long Thermal Time Constant)	Importance of Prompt Shut-Down effects
A Type	More Complete Tests on A Plus Subassemblies or Possible Destructive Tests

SPERT II

Core	Experiments
B H_2O Moderated	Effect of Temperature Pressure Flow
B D_2O Moderated	Effect of Prompt Neutron Lifetime
D Zirconium	Environmental Variables of Temperature, Pressure and Flow, and Effect of Thermal Characteristics
BORAX IV Type	Extension to Conditions of Spert II

SPERT III

Core	Experiments
C Stainless Steel	Environmental Variables Pressure Temperature Flow
C Stainless Steel	Engineering Tests Cold Water Accident Loss of Flow Subassemblies
D Zirconium	Extension to full Range of Spert III Conditions

SPERT IV

Reactor	Experiments
Swimming Pool	Reactor Prototype Safety Study
Special Prototype Tests	
New Types as Suggested by Reactor Physics Planning Group e.g., LMFR Type, Gas-Cooled Type, Low Enrichment Type	

TABLE II

PROPOSED PROGRAM SCHEDULE

CALENDAR YEARS	SPERT I	SPERT II	SPERT III	SPERT IV	ANALYTICAL WORK
1959	APPR Core Pile Oscillator Experiment Capsule Experiment Stability, Ramp, BSR Core Control System Tests Ramps, Steps and Stability	Construction Plant Checkout and Acceptance B Core H_2O Hot Criticals Plant Shakedown	Critical December 19, 1958 SS Core Hot criticals, full operational loading Slab criticals Plant Shakedown Hydraulic Tests Static Measurements Temperature, Pressure and void coefficients over full range of operating condition. Preliminary transient tests		Major Tasks (not a schedule) Burst Analyses Condensation and dissemination of Analytical work Calculation of static and dynamic reactivity coefficients Survey of dynamic characteristics of different reactor types Prediction of behavior for specific systems Analysis of specific mechanisms Development of general safety criteria Development of approximate rules of thumb
1960	APPR Core Sub-Assembly Meltdowns and Destructive Test Clean Up Ceramic Core Steps, Ramps, Stability Destructive Test Clean Up	Hydraulic Testing B Core D_2O Hot Criticals B Core H_2O Transient Tests Various Core Configurations including positive coefficients	Transient Testing	Construction Aluminum Core	
1961	Mechanisms Studies Measurement of Dynamic Coefficients for Different Reactor Types Exploration of new Shutdown Mechanisms	B Core D_2O Transient Tests	Engineering Experiments	Pool Stability Studies with several core types, with and without forced circulation	

TABLE III

SPERT EXPERIMENTAL PROGRAM OUTLINE

SPERT I

APPR CORE

- Completion of Oscillator Studies
  - High-Power Studies
- Kinetic Tests
  - Completion of Step Series
  - Ramp Tests
  - Stability Survey Studies

BSR CORE

- Static Tests
  - Critical Mass
  - Reactivity Coefficients
  - Control Rod Calibration
- Control System Tests
  - Reactor Shutdown by Control Rods
- Kinetic Tests
  - Step Tests
  - Ramp Tests
  - Reactor Shutdown by Control Rods During Transients
  - Stability Survey Studies

APPR CORE

- Sub-Assembly Meltdowns And Destructive Tests

CERAMIC CORE

- Static Tests
  - Critical Mass
  - Reactivity Coefficients
  - Control Rod Calibration
- Kinetic Tests
  - Step Tests
  - Ramp Rests
  - Stability Survey Studies
  - Destructive Test

SPERT II

PRE-NEUTRON TESTS

- System Checkout
  - Pressure
  - Temperature
  - Flow
  - Leak Check for D_2O
  - Reactor Control System
  - Plant Control System

B CORE

- Static Tests - H_2O
  - Critical Masses
  - Operational Loadings
  - Rod Calibrations
- Hydraulic Tests
- Static Tests - D_2O
  - Critical Masses
  - Operational Loadings
  - Reactivity Coefficients
  - Flux Maps
  - Rod Calibrations
- Static Tests - H_2O
  - Reactivity Coefficients
  - Flux Maps
- Kinetic Tests - H_2O
  - Step Series
  - Ramp Series
  - Stability Studies
  - Temperature, Pressure, Flow As Variables
- Kinetic Tests - D_2O
  - Step Series
  - Ramp Series
  - Stability Studies
  - Temperature, Pressure, Flow as Variables

SPERT III

STAINLESS STEEL CORE

- Hydraulic Tests
  - Flow Tests
  - Core Pressure Drops
  - Hydraulic Behavior with Pressure and Temperature
  - System Heat Balance
- Static Tests
  - Reactivity Coefficients, Rod Calibrations over full range of temperature and pressure
- Transient Tests
  - Step Tests
  - Ramp Tests
  - Stability Studies
- Engineering Experiments
  - Loss of Flow
  - Flow Restriction
  - Sub-Pressure Tests
  - Cold Water Accident
- Special Tests
  - Sub-Assembly Tests

SPERT IV

PRE-NEUTRON TESTS

- System Checkout
  - Flow
  - Reactor Control System

ALUMINUM CORE

- Static Tests
  - Critical Masses
  - Reactivity Coefficients
  - Rod Calibrations
- Kinetic Tests
  - Possibly Step Tests for orientation
  - Possibly Ramp Tests for orientation
  - Pool Stability Studies with and without forced circulation

V. REFERENCES

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VI. APPENDIX A

I. REACTOR TRANSIENT TEST FACILITY AND PROGRAM

A. Justification

The feasibility of the several reactors now being developed for operation in populated areas depends, among other factors, on designing each reactor system so that the release of a significant quantity of radioactive materials into the environs is improbable beyond reasonable doubt. A thorough understanding of the transient behavior of each type of reactor is essential as the basis for the design and evaluation of the reactor system. The recent Borax tests have provided invaluable information but have also brought to attention the urgent need for additional and continuing investigations for determining the dynamic behavior of reactors over wide ranges of conditions. Since such studies must include experiments which present unusual hazards, an adequately remote facility designed for such experiments is required.

B. Objectives

1. Conduct experimental and theoretical studies as necessary to obtain a thorough understanding of the behavior of reactors under transient conditions with particular regard to phenomena which might result in an explosive release of energy such as extreme pressure surges due to excessive rates of energy release, or possibly chemical reactions. Experiments with plate fuel elements in an open tank with and without forced circulation are of most immediate interest.

2. Determine experimentally the maximum reactivity as a function of time rate that can be safely introduced into a few selected types of reactor cores. Appropriate theoretical studies should be a part of this program for planning of the experiments and analysis and interpretation of the results. The following reactor types are currently of interest in order of priority:

   (1) A reactor of the general type being planned for the University of Michigan. Such a reactor will probably have stainless steel clad fuel elements, a fixed beryllium oxide reflector and forced circulation.

   (2) A homogeneous reactor of the type being considered for university research. North American Aviation has a project for the planning of such an experiment and the fabrication of the reactor core and control system.

3. Conduct transient experiments to determine dynamic behavior of other types of reactors such as:

   (1) A sodium-cooled graphite-moderated reactor;

(2) An intermediate reactor (SIR);

(3) A fast breeder reactor.

C. Facility

The physical plant might consist of an underground instrument shelter which would serve as a central facility for three to four reactor test pits. The pits would be tanks ten to twenty feet in diameter and twenty to thirty feet deep either buried below grade or above grade with earth barricade. A removable shelter with space heating will allow core assembly and control rod testing in these pits during inclement weather.

The test pits would be located several hundred feet from the central instrument shelter.

Control decision and instrumentation would be telemetered to a control station approximately two miles from the pits.

Every attempt should be made to obtain complete temperature and pressure records during tests as well as to obtain photographic records of physical perturbations.

Flexibility of instrumentation to adapt to cores of widely varying design will be specified.

VII. APPENDIX B

SUPPLEMENTARY REFERENCES

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