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SPERT PROGRAM REVIEW

W. E. Nyer and S. G. Forbes

September 27, 1957

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SPERT PROGRAM REVIEW

By

W. E. Nyer and S. G. Forbes

ABSTRACT

The needs of the reactor safety program for experimental data are discussed and the importance of reactor kinetic behavior in runaway accidents is described. The way in which the Spert kinetic studies are related to these needs is presented along with a review of the experimental work and recommendations for the continuation of the Spert program.

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SPERT PROGRAM REVIEW

By

W. E. Nyer and S. G. Forbes

I. INTRODUCTION

A. Objective of the Report

The purpose of this report is to review the Spert work in its relationship to the problems of reactor safety and to present plans and proposals for the future Spert program.

A general presentation is required to understand individual phases of the work which might otherwise appear somewhat isolated. The presentation is hampered by the lack of a well-developed terminology in the reactor safety field. To bridge this gap in communication, several arbitrary definitions and classifications are introduced as a framework for discussion.

The presentation will begin with a discussion of the problems and needs of the reactor safety program. This is followed by a review of experimental and theoretical work on reactor kinetics. Finally, the proposed continuation of the program is presented.

B. Reactor Hazards and Objective of Safety Program

The important factors and the full nature of the problems in reactor safety will be taken to be clearly revealed by the instructions for the preparation of hazards reports(1) and the Geneva paper of McCullough, Mills and Teller(2). For the purpose of this report the following simplified views will be adopted:

The discussion of reactor safety will be limited to the problem of reactor runaway and its immediate consequences. The term "accident" will be used to mean "runaway accident" unless otherwise stated. Accidents will be viewed as having three phases: Initiation, Reactor Response, and Consequences, with three distinct classes of damage existing under Consequences: the reactor and plant, the plant operators, and the general public. Damage occurs through the agency of excessive pressures, excessive temperatures, excessive radiation levels, or movement of fission products. Most likely, accidents confined to the plant equipment will be typified by mechanical damage. When the damage volume also encompasses plant personnel, all factors become important but the spread of fission products assumes the major role. In

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the larger volume occupied by the general public the chief agent of damage is undoubtedly the spread of fission products. Only two distinctions will be recognized: mechanical damage (to the reactor plant) and fission product damage (to all volumes).

C. The Phases of an Accident

The nature of the various problems related to runaway accidents will be examined next.

1. Initiation

A runaway accident can be initiated in numerous ways, but each of these can be characterized as a reactivity disturbance under a particular set of initial conditions of the reactor system. The disturbance must be further specified by its magnitude and speed of occurrence, which brings into consideration the various means of reactivity introduction: malfunctioning of the reactor, accidental damage, sabotage, operator error, and even inherent instability. The means of introduction of the disturbance and its time-scale, magnitude and probability of occurrence are so intimately related to the design details of individual reactor plants that an extended general discussion is not greatly rewarding. However, it can be noted that the non-nuclear components of the system, particularly the control system, exercise an important role in this phase.

2. Response Characteristic of the Reactor System

Subsequent to the reactivity disturbance, events are determined by the response characteristics of the reactor system, which are a combination of the largely non-nuclear characteristics of appendages like the control system, and inherent characteristics, which are strongly nuclear. In the most extreme accident cases the inherent characteristics will fully determine the results. In less extreme cases where the characteristics of the appendages may produce a sensible effect on the result, a simplification results if the internal and external parts of the reactor loop are ignored so that this phase will be considered determined solely by the inherent reactor characteristics.

3. Consequences

In the final phase additional problems arise such as the containment of explosions and factors affecting the dispersal of fission products.

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D. Importance of Reactor Response Characteristics

In each of the above phases it is possible to take effective action to minimize the consequences of an accident. Actions in the first two phases are in the nature of accident prevention; whereas, action in the third phase is essentially that of providing adequate containment.

Historically, the primary safety effort has been devoted to the problems of initiation where the line of attack was more readily apparent. However, to reduce the probability of initiation to a negligible level it may be necessary to surround the reactor system with so many inhibitory devices that its functional purpose is compromised. Conversely, in providing for normal operating procedures the screen of safety devices may be breached. A further weakness of the inhibitory approach is that failure of such a device may in itself initiate an accident.

At the other extreme, the containment approach is a matter of coping with a bad situation after it has already occurred. Since reactor accidents can conceivably be catastrophic, it is obviously better to prevent them than to cope with them. Even when containment successfully prevents hazard to the general public, it is unsatisfactory from the point of view of plant protection. Clearly, the severity of an accident is not solely determined by the initial reactivity disturbance, but also by the responses of the reactor to that disturbance. Ideally, the reactor characteristics should be such as to prevent the occurrence of the more serious consequences. Therefore, maximum operational flexibility commensurate with safely limited consequences will be obtained by optimizing the reactor response characteristics. It is the investigation of the factors affecting the reactor response characteristics that constitutes the reactor kinetics program.

II. REACTOR STUDIES

A. General Needs of Reactor Studies

In Section I it was stated that reactor response is of major importance in reactor safety and that the reactor kinetics program is concerned with the study of factors influencing reactor response. This section will discuss the general needs of a kinetics program and the relation of the Spert work to these needs.

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1. Reactor Classes to be Investigated

Practical considerations orient the program along lines of investigating reactor systems of particular immediate importance to the reactor field such as heterogeneous water reactors, fast reactors, and homogeneous reactors. The emphasis of the following discussion on heterogeneous water systems should be taken as an indication of the state of development of the program rather than as an indication of its limitations. There are, however, reasons for deferring the examination of other systems until some backlog of experience has been obtained. For the present, attention will be confined to the heterogeneous water class.

2. Types of Reactor Information Needed

On the one hand, information about reactor kinetic behavior in general is needed. This is potentially of greatest use, but requires an extended effort of considerable magnitude. It represents the long-term objective of the program.

On the other hand, the present state of growth of the reactor industry and the nature of such growth require more specialized information which will have more immediate application to the reactor class investigated than to the general program. Even more specialized information is sometimes needed for specific reactors and will have only limited application to the class.

These three kinds of information will be designated as Basic, Class-Prototype, and Reactor-Prototype, in increasing order of specialization. These last two types of information, which represent short-term goals, can be obtained and applied more quickly, but a program based entirely on these would be of little long-range value because the necessary unifying factors would be lacking.

In the initial stages of the program when experimental data are absent, any series of experiments will contribute to all three areas of information. As the fund of information is accumulated it becomes more difficult for prototype tests to contribute to the basic understanding of kinetic behavior and the types of testing become more distinct.

3. Related Studies

Outside the area of reactor response are related safety studies, such as work on fuses, control systems, and core meltdown which in some stage of their development require tests under simulated accident conditions or which for simple convenience need the use of a Spert-type facility to provide neutron bursts.

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4. Types of Information to be Obtained by Spert Program

Initially the Spert program was deliberately oriented toward the basic objectives for three reasons: first, the long-term benefits are most important; second, the common tendency to overemphasize short-term objectives would be appropriately balanced by a well-established basic program; third, the approximations and kinds of testing required to yield useful short-term information could best be met with a background of consideration and initial work on a basic program. This last point is of particular importance in discussing the extensions of the program to other reactor classes.

To a somewhat lesser extent, the program has also been directed along shorter-term lines and has in fact been multipurpose as is shown by the discussion below.

Spert I. This reactor has been used primarily to obtain basic information, but to a large extent is also the class-prototype for swimming pool reactors. A portion of the Spert I effort has also been devoted to purely service measurements in which the reactor was used as a pulsed source. In the future it is expected to serve these same functions.

Spert II. The primary intention for the use of Spert II is to obtain basic information; however, it is expected that needs will develop for the use of this system to obtain prototype information as well as its use as a pulsed source.

Spert III. The emphasis on information to be obtained in this system is shared among the basic and prototype groups. This reactor will also be used as a driving source for sub-assembly tests.

Prototype testing can be expected to increase in relative importance as the program matures until it becomes a significant part of the effort because the reactor safety program can only be kept abreast of the rapidly developing reactor technology through selective prototype testing. The demands for short-term tests originate from the following sources:

a. The Atomic Energy Commission's need for answers to very specific questions in the reactor safety program.

b. The Atomic Energy Commission's need for information for the development of policies and rules governing regulation of domestic and foreign production, sales, or use of reactors.

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c. Requests by foreign nations for information from the Atomic Energy Commission in connection with the United States international activities.

d. Requests from commercial organizations for information from the Atomic Energy Commission.

The general program is well under way and some of the outstanding needs of specific reactor systems have become sufficiently clarified and intensified that the program should now more actively encompass activity directed toward these needs. To this end certain steps and additional facilities are proposed in Section III as a continuation of the present activity. It should be emphasized that this is a matter of bringing prototype testing into balance with the other directions of the program and not a matter of permitting it to dominate or divert the part of the program devoted to basic studies.

B. Reactor Safety Studies

1. General Considerations

The means by which the reactivity disturbance which initiates reactor runaway might occur warrant special consideration. Obviously no physical process can actually achieve an instantaneous change in reactivity. In some cases the reactor response will produce reactivity compensation before the reactivity addition has been completed and thus, loosely speaking, the accident will be over before all the available reactivity has been inserted. Whether this limitation on the effective magnitude of the disturbance will be of practical value in preventing serious consequences depends on the degree of self-regulation of the reactor and on the relative time scale for introduction of the disturbance as compared with the speed of the reactor response. For moderate rates of reactivity injection it becomes quite difficult to inject excess reactivities above prompt critical where there is a rather abrupt decrease in the reactor period, particularly in a reactor with a short prompt neutron lifetime. However, extremely rapid power rises may outrun shutdown effects and increase the hazards.

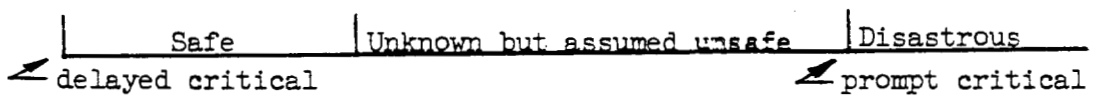
One means of limiting the initiating reactivity disturbance is to limit the rate at which reactivity can be introduced into the system to as small a value as practicable. Aside from problems of fuel handling during loading and special experimental situations, the reactor control rods are the most obvious sources of possible reactivity disturbances. Limitation of rod withdrawal rates is therefore clearly desirable from a safety standpoint. The permissible withdrawal rate will depend in part on the power level at which the reactor is operating because reactor response time must be expressed in terms of the time required to reach the power level at which self-regulation becomes effective rather than by reactor period alone.

Limitation of the withdrawal rates is not a sufficient guarantee of safety because other inherent sources of reactivity disturbances exist such as sudden temperature or pressure changes. One consideration which appears to be of particular importance is that of reactor stability. Instability has been frequently treated as a problem in plant operation while being regarded as an unlikely source of large and sudden reactivity changes. This restricted view has not been borne out by experiment and the inter-relation and importance of all these factors must be investigated further.

2. Indices of Safety

For a large class of problems, reactor safety and reactor safety studies can be conveniently discussed in terms of the way in which the reactor ultimately responds to a step-input of reactivity. The measure of the reactivity input is taken to be the asymtotic period, τ, or its reciprocal, α. This can be translated to the customary units of Δk if suitable secondary determinations of prompt neutron lifetime and delayed neutron fraction can be made, but τ is preferred because it is a directly observable experimental quantity.

Three types of behavior can be distinguished: safe, unsafe, and disastrous, corresponding to no damage, limited mechanical damage and widespread fission product release, respectively. These response-types form zones of behavior as a function of τ with hazards increasing as τ decreases, as illustrated in the sketch below.



In practice, reactor safety has been achieved essentially by considering the safe, or operating, zone to be that for which the excess reactivity was almost zero. The region of disaster was generally believed to begin in the neighborhood of prompt critical and the region between these was unknown, but assumed unsafe. It should be emphasized that generally the imperfect state of knowledge permits neither the determination of the boundary of the danger zones nor the extent of the danger in the zones.

A second behavior characteristic that can be used as an index of safety is reactor stability. The many factors which contribute to reactor instability are only partially understood at present but in general increasing the operating power level will also increase the tendency toward unstable behavior. Since instability is not dependent on large initial reactivity disturbances the stability

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index of safety may be regarded as being power dependent. While instability can be classed as a particular case of the initiation phase, it is distinguished from other sources of reactivity disturbances by its spontaneous nature.

3. Types of Measurements

Four types of experiments have been made in the kinetics program:

1. Step transients
2. Ramp transients
3. Stability tests
4. Static measurements

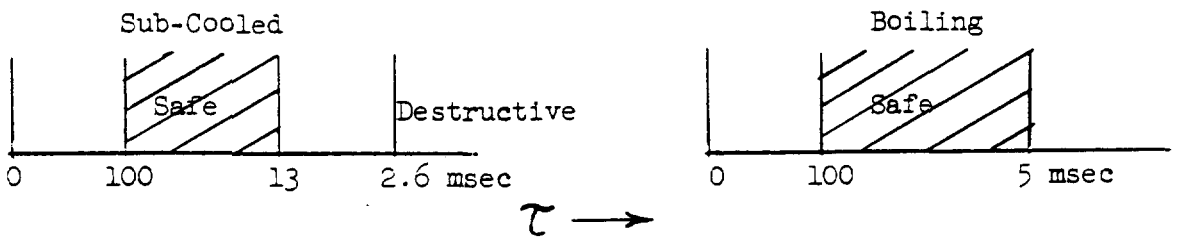
Ramp and step studies are concerned mainly with accident behavior with the ramp being the more likely representative of conditions surrounding an actual accident, but with the step being simpler to interpret in the present view and also more representative of extreme accidents. Stability is an ever-present problem which is rather specially related to the behavior about an equilibrium power under normal operating conditions; however, its role as an initiator of major reactivity disturbances needs investigation. The static measurements are of major importance first as part of the calibrations for the experiments, and second because these basic reactor quantities are required in the direct interpretation of the experiments.

The theoretical and experimental work on these problems will now be surveyed.

4. Borax Test Series

a. Scope of Experiments

The first extensive experimental kinetic studies were the Borax experiments(3)(4). Quite briefly, the non-destructive Borax I and II transient work covered the period region between 100 msec and 5 msec at boiling temperatures, and between 100 msec and 13 msec sub-cooled to ambient temperature. Burst behavior was explored as a function of temperature for a selected number of periods. In addition, one sub-cooled destructive test was run at a period of 2.6 msec. Schematically, the regions explored were these:



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That is, this was an examination of burst behavior first as a function of alpha and second as a function of temperature.

Stability tests were conducted in the safe and unsafe zones under non-pressurized and pressurized conditions.

b. General Borax Results

Even though the experiments as such did not permit any generalizations of reactor behavior, there were the following very important consequences:

(1) The experiments demonstrated that useful information could be obtained from tests of this type.

(2) The existence of experimental data, where heretofore there had been none, formed a focal point and stimulus for theoretical work.

(3) Questions concerning the uncertainties became more pointed. With the evidence that a particular type reactor could withstand sizeable reactivity injections, it became meaningful to ask such questions as, "Exactly how much reactivity injection can any reactor stand?" That is, to whatever extent prompt critical had previously represented the transition between harmful and disastrous conditions, it did so no longer.

c. Specific Results for Borax-Type Reactor

More quantitative conclusions are possible for the specific type of reactor represented by Borax:

(1) For step-inputs of reactivity, the permissible limits on available reactivity for this system could be set with greater assurance and the location of the three zones of operation could be better established. The safe operating zone could be expanded from a narrow area about zero reactivity to the start of the uncertain region which could be put well above prompt critical, with the beginning of the disastrous region somewhere below 4 per cent.

(2) The destructive test represented a considerable step forward in the evaluation of accident consequences.

(3) The major advances in stability were made largely with the pressurized systems which demonstrated a high degree of safety and ease of control at a usably high power for a boiling reactor system.

(4) Some measure was obtained of the conditions under which stability decreased to the point of becoming unsafe.

(5) Implicit in (4) above was the fact that stability could be at least partially controlled in design and operation.

d. Associated Theoretical Work

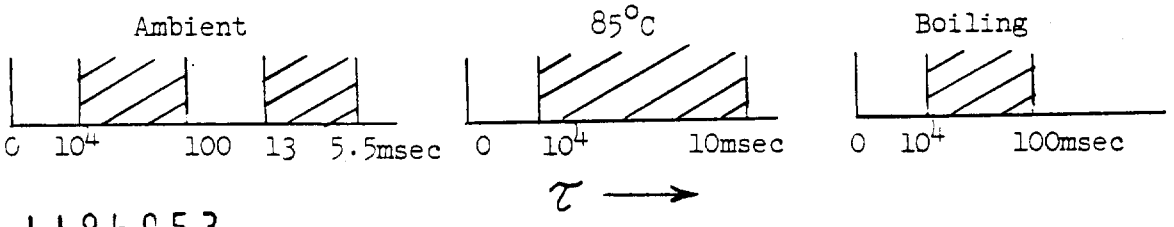
Although not a part of the Borax activity, pertinent theoretical efforts to explain burst behavior were stimulated at ORNL, NDA and NRL by Borax results(5)(6)(7). These attacks were all concerned primarily with steam-void growth in the moderator and differed only in the models adopted for describing the mechanism of energy transfer from fuel to moderator.

5. Spert Experimental-Theoretical Activities

a. Scope of Experiments

The Spert project, which was initiated at least partly in response to the new questions raised by the Borax work, had the original directive to continue and extend the Borax tests. A part of this effort was to be specifically directed toward repeating the Borax work and at the same time to extend the work into the region between the destructive and non-destructive tests.

In like manner to the treatment of Borax work, the Spert unpressurized work can be summarized as explorations of various zones of transient and stability behavior, and static measurements. The step transient tests covered the regions from 10 sec to 5.5 msec sub-cooled to ambient; from 10 sec to 5 msec boiling; and from 10 sec to 10 msec sub-cooled to the intermediate temperature at 85°C, as shown schematically below. The new areas explored are indicated by the cross hatching



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In addition to the study of burst behavior as a function of alpha and of temperature, a start was made on the study of the influence of water-head. Ramp transient explorations were conducted for a range of starting powers of a factor of 10^9, i.e., from 10^{-4} watts to 10^5 watts; and for a range of reactivity injection rates from 0.01% Δk/sec to 0.36% Δk/sec. Stability studies were made of the conditions for onset of "divergent" power oscillations as functions of reactivity in the system, temperature and water-head. As a part of the static measurements series a detailed investigation of void coefficients was carried out.

b. Scope of Theoretical Work

As part of the Spert program, theoretical studies have been made dealing with excursions, stability and statics. The excursion work followed three major directions distinguished by the relative emphasis given to the types of shutdown mechanisms postulated, the details of energy transfer to the given mechanism, and the method of coupling shutdown effects to burst behavior.

The individual groups of calculations are described below with the first six items covering transients:*

(1) ORNL, NDA, NRL Boiling Models. One effort was the application of the NRL, NDA, and ORNL models to Spert work with a relatively minor adjustment in the theory of the reactivity to be compensated. In place of the initially inserted prompt excess reactivity or total excess reactivity, the k compensated at the time of peak power was used. This was obtained from computer analysis of Spert data. This permitted extension of the models to include the full range of Spert experimental work.

(2) Diffusion Calculations of Temperature. Diffusion model calculations of temperature distribution, such as those performed by NRL, were used to obtain the energy transferred to the moderator as a function of time. Refinements consisted of the use of the k-compensated information and the experimentally obtained temperature coefficient and plate temperature.

(3) Bubble Formation. Because the importance of the details of the initiation and growth of bubbles

*As a temporary measure at the outset of the program, the Ramo-Wooldridge Corporation was engaged as a sub-contractor to Phillips Petroleum Company to carry out theoretical investigations. A final report is being prepared at the present time. It covers the majority of the work on items 2, 3, 5 and 7.

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could not easily be evaluated, a preliminary survey was made of the problems in this field in order to determine whether this line should be extended.

(4) General Model with Coupling Between k (t) and Power History. The Fuchs model[8] for burst behavior, which explicitly includes the coupling between self-induced shutdown effects and power behavior, was applied to the reactor problem by generalization of the shutdown coefficient and the inclusion of inertial effects to predict transient pressure behavior. The principal virtue of this simplified approach lies in the fact that analytical expressions for the importance of the individual reactor parameters can be readily derived and used as guides in predicting the results of more detailed mathematical models for which analytical solutions are generally not realizable.

(5) Models with Coupling. Another group of calculations was made in which coupling was also employed with specific functional forms for heat transfer from fuel to moderator. This work was partly analytical and was intended to include analog machine study.

(6) Model with Coupling and Characteristic Time Constants. The above models on one hand suffered from the neglect of feed-back and delayed neutron effects, and on the other hand attempted too detailed an analysis of the energy transfer process. In principle, the first defects are easy to remedy. The problems of complexity were sidestepped by accounting for the shutdown effects in terms of a formation rate and disappearance rate, i.e., in terms of characteristic times for energy transfer and energy escape. Early models are "one-shot" models since they describe only the primary burst of a step-transient. The inclusion of a reactivity escape mechanism is the only means whereby any model can describe the secondary power bursts which follow the primary burst.

Items (1) and (2) were simply applications of existing boiling model approaches with a minor modification to broaden the region of applicability. The remaining efforts were extensions because of the inclusion of coupling and delayed neutrons, and the lack of restriction to boiling shutdown.

(7) Oscillations. Studies of the divergent oscillations observed in Spert I have resulted in linear and nonlinear mathematical models which describe the

observed behavior qualitatively, but a close relationship between the parameters in the models and physical phenomena remains to be established.

(8) Transfer Function. Transfer function work, which is now in its earliest stages, has two objectives--the study of linear and non-linear stability, and the measurement of reactor constants.

(9) Statics. Calculations of the steady-state properties of Spert reactors have been required for purposes of reactor design and experiment design.

c. General Spert Results

The Spert experiments[9] have produced some intensification and clarification of the general results obtained by Borax in that the useful information has assumed a more quantitative aspect. Certain other consequences of general application to all reactor systems have also resulted.

(1) The reactivity-compensation required to turn a burst is sensibly less than the injected reactivity for an important class of tests. The ratio is influenced very strongly by the shape of the burst. The immediate practical importance of this result is that for many incidents, that is those involving reactivity injections up to the region of control by prompt neutrons alone, the emergency control of a reactor does not require a fast-acting, strong control. A fast-acting, small control will produce the needed delay in the burst to permit the slower insertion of the strong control. This advantage is limited to reactivity additions not greatly in excess of prompt critical. This simplified view does not adequately describe reactors which can be autocatalytic for part of a burst.

(2) Of equal importance and generality of application is the result that the severity of the ramp-produced burst is strongly dependent on ramp-rate and weakly dependent on initial power over a very wide range. The results imply first that in many cases the source strength is unimportant and indeed that only a very weak source is necessary, and second that reactivity injection rate must be limited.

(3) It is possible for the small initial reactivity disturbances characteristic of stability problems

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to lead to the large reactivity changes associated with runaway. The very rapid and large self-modulation of reactivity demonstrated by the Spert oscillations shows that it is not an adequate claim to reactor safety to show that reactivity cannot be introduced into a reactor at the rate required to produce an accident--it must also be shown that the reactor itself cannot do this. Stability thus becomes an important factor in runaway initiation.

(4) Static measurements of reactor properties are required in detail for interpretation of theories on transient behavior and for safety evaluation. The fundamental constants such as the effective delayed neutron fraction β and prompt lifetime l are not adequately known for these purposes.

d. Specific Results for Spert I-Type Reactor

The more quantitative results, which are strictly applicable of course only to the specific reactor type, are the following:

(1) The extension of the sub-cooled work from the 13 msec region covered by Borax to the 5.5 msec region covered the major part of the unexplored zone between 13 and 2.6 msec.

(2) A very slight decrease in period would result in fuel plate melting. Hence, the division between the unsafe and disastrous zones has been located with considerably greater precision. For example the location has been narrowed from the period-range of 13 to > 2.6 msec to the period-range from 5.5 to > 2.6 msec with the prediction that it begins at a period of 4 msec. Extrapolation of the curves of the measured quantities to the 2.6 msec region becomes correspondingly more accurate.

(3) On the basis of such extrapolations it seems highly unlikely that there were any discontinuities in the Borax curves of the energy released, the temperatures developed, or the peak powers reached. It also seems likely that there was a discontinuity in the pressures developed because the extreme pressure observed in the Borax destructive test would not be predicted from Spert data. This discontinuity in pressure is probably due to a simple steam explosion caused by melting of the fuel plates which greatly increased the rate of energy transfer.

(4) An increase of water-head or water-temperature lowered the reactivity-compensation level at which oscillations diverged; that is, decreased stability.

(5) The need for detailed consideration of local void effects was demonstrated by a series of static measurements performed explicitly as an aid to the AEC in evaluating the treatment of this subject in hazards reviews. It was shown that a superficial treatment was inadequate. Void worth calculations must be carried out in detail as a function of void size and location.

(6) Several consequences are of a more detailed nature. The extension of the sub-cooled tests to the long-period region revealed that boiling could not be the shutdown mechanism for periods greater than about 50 msec. The boiling models are thus inappropriate over a large region of interest and other effects must be incorporated into the theory.

It was also observed that the energy released and the reactivity compensated had the same general dependence on reactor period and that both passed through a minimum in the region of prompt critical. This is probably a general result arising from the neutron kinetic behavior, although certain characteristics of the shutdown process could contribute to this effect. The reactivity compensation required to halt the burst was found to be an order of magnitude smaller than had previously been assumed on the basis of the initial reactivity injection. A detailed examination of mechanism previously considered insignificant thus becomes necessary. In making such calculations accurate measurements of kinetic parameters become basic to the interpretation of long-period tests. For example, the value of the effective delayed neutron fraction probably lies between 0.70% and 0.80% but this spread represents an uncertainty of about a factor of two in the calculated values of required reactivity compensation. The magnitude of this uncertainty makes selection of the responsible mechanisms impossible. Special experiments will be required to determine the importance of individual mechanisms.

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III. CONTINUATION OF THE SPERT PROGRAM

A. General

The program will continue to extend the experimental-theoretical activities as follows:

1. Basic Studies

The long-range basic investigations of the importance of various parameters in reactor kinetics behavior will be continued both experimentally and theoretically. This program consists of a systematic study of the effects and importance of nuclear, engineering, and environmental variables whose importance has been generally acknowledged but not previously investigated, in order that they may be properly incorporated in the theoretical models of reactor kinetics. This experimental information is essential in the formulation and particularly in the evaluation of reactor kinetic theories. The extensive series of tests which has been performed on the Spert I "A" core constitutes a detailed set of reference measurements on a system having a particular combination of characteristics. From this base point the effect on kinetic behavior produced by varying individual parameters will be studied in detail. Thus the past two years of experimental work will reach its maximum utility as the program of parameter variations matures. Similarly, the testing and selection of realistic theoretical models will be placed on a much firmer foundation as this phase of the program develops.

Specifically, the program includes variations of void and temperature coefficient, prompt neutron lifetime, thermal characteristics of fuel plates, fuel plate geometry, moderator hydrodynamics, pressure, temperature and coolant flow. The investigations will be carried out along the established lines of static, step, ramp and stability studies together with other techniques to be developed and incorporated wherever these appear to be useful adjuncts to safety studies.

One such technique which has been under consideration but not previously employed at Spert is that of pile oscillator measurements. This technique is being actively pursued for immediate application to the Spert I reactor. Ultimately transfer function investigations of essentially linear oscillations will be carried out on all systems as a part of the study of reactor stability as an operating problem. The purposes are to check the predictions of Bethe[10] in regard to the suitability of transfer function techniques in determining reactor safety, to develop the technique for testing and analysis, and finally to be a preliminary for the study of non-linear oscillations--an accident problem. The conditions for linear stability do not necessarily determine non-linear stability and the relation between these must be established. The independent determination of

such important static parameters as prompt neutron lifetime and effective delayed neutron fraction by this means is also an attractive possibility.

2. Class Prototype Tests

Although the Spert I reactor is in many ways typical of the swimming pool class of reactors it is inadequate as a class prototype in the important investigation of reactor stability. In the Spert I studies it was found that both water head and the size of the water reservoir had an important bearing on stability and neither of these factors could be adjusted to be properly representative of the swimming pool class in the present facility. It should be noted that the condition represented by the deep pool is not equivalent to the application of static pressure because pressure generated by moderator expulsion depends on the magnitude of the inertial load and its acceleration and is, therefore, time-dependent; whereas an externally imposed static pressure remains constant during moderator expulsion. The urgent need for safety information on this rapidly growing class of research reactors makes it advisable to conduct a complete class prototype test in the immediate future. It appears entirely feasible to construct a facility which will fulfill this need and which will possess adequate flexibility for other class-prototype or small reactor-prototype testing as the need arises.

3. Destructive Tests

Planned destructive tests were not considered as a part of the early stages of the program because of their interference with the more basic studies. Although such tests are principally concerned with the consequence phase of an accident and hence fall outside the response phase which is the field of kinetics, many aspects of destructive behavior are so intimately related to the reactor behavior that representative data can only be obtained in a reactor experiment. At present the data are meager on this phase of an accident and the possibility that such tests will become necessary cannot be excluded from the program.

4. Services

Services for special tests such as neutron bursts for detector evaluation, fuse testing, control system testing and meltdown studies will be provided.

5. New Reactor Types

Examination of other reactor types for possible inclusion in the Spert program will continue. New kinetic reactor studies should to some extent follow rather than lead demonstration of the probable usefulness of new reactor types.

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The program set forth represents a continuation and broadening of the program as conceived at the outset of the Spert project. It is based upon the past developments within the program, and discussions with and informal recommendations from persons active in the reactor field. The program of basic studies is to continue without interference from other activities. Prototype measurements, destructive tests, service functions and investigations of new reactor types are to be kept in balance with the current needs of reactor technology. The manner in which the present program extends the earlier proposal can be seen by comparing it with Figure 1 and the following paragraphs, which are condensed from the proposal submitted to IDO in October 1955[11].

Measurements were proposed on the effects of variation of:

fuel-plate spacing
prompt neutron lifetime
thermal characteristics of fuel plates
inertial load
ramp-rate
pressures
temperatures
flow
initial power
flux- or space-dependent factors such as
void coefficients

This involved six core types:

A - aluminum plate
B - aluminum plate adjustable spacing
C - stainless steel
D - zirconium
E - aluminum low concentration
F - mixed combinations of A and E

in the reactor systems Spert I, II, III. Provision was made for special tests such as the "loss of flow" series. The plan as outlined has been followed except for the addition of a limited amount of service tests at the request of the AEC.

The facilities considered necessary to carry the program forward are outlined in the following section.

B. Facilities

Each of the three reactor facilities presently included in the program was designed to provide control over certain environmental conditions believed to be important in reactor behavior. These variables were based principally on the needs for the basic studies

and to a lesser extent (primarily in Spert III) on the need for class prototype information. The investigation of environmental parameters, nuclear characteristics, hydrodynamical and thermal characteristics is provided for by selection of specific cores to be used in these reactor facilities. In addition to these three plant installations it is proposed to add a fourth facility to provide the high water head and large reservoir necessary to carry out adequate prototype tests of swimming pool reactors. The importance of these factors has been demonstrated but not adequately explored in Spert I.

The new facility will provide a flexible site for prototype tests of small reactors to meet the needs for such experiments without interference with the basic program and could also be used for basic or class prototype tests if occasion so demanded.

For completeness the characteristics of these four installations are summarized below:

Spert I is a minimum facility providing for a limited adjustment of water temperature and hydrostatic head but without provision for cooling or flow control. Control of water quality is limited by infiltration of airborne debris and contamination by rusting of the carbon steel vessel. The reactor construction is simple and readily modified. The core is easily accessible for static measurements, core modifications and instrument installation, within the limits of individual core interior dimensions.

Spert II is an intermediate pressure facility providing considerable flexibility in pressure, temperature and flow. The system is designed primarily to permit selection of various moderator and reflector materials, principally light and heavy water. Water quality control is good, concomitant with the requirements for maintaining heavy water purity. Both up and down controlled flow is provided for. The facility is relatively complex and much less flexible than Spert I in regard to design modifications, static measurements and instrument installation.

Spert III provides for a wide range of temperature, pressure and flow plus heat removal capacity for power operation. Because of the complexities introduced by high-pressure and high-temperature operation, it is by far the least flexible with regard to either plant modifications or experimental work. However, the plant will provide class-prototype data for pressurized water reactors. Because of the difficulties in even simple opening and closing of the pressure vessel the experimental progress will be necessarily slow. For this reason experiments on a particular core for this reactor will be carried out wherever feasible in the Spert II facility over the range of environmental conditions available therein. Control of water quality will be good. Water flow control over a wide range is provided for but flow is restricted to the up direction for practical reasons.

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Spert IV would provide shelter and facilities for construction and testing of small reactor prototypes. Although no detailed proposal has been prepared, it is conceived that the facility would consist of a large pit suitably housed and provided with coffer dams to permit division into either water filled and/or dry sections. A traveling bridge would be provided for supporting swimming pool core and control devices. Other small reactors would be accommodated by draining the pool and using the pit as a suitable shield for protection in the event of destructive tests.

Cores and Tests. To carry out the investigation of various core characteristics several cores of varying design are needed. Again for completeness, characteristics are listed for all cores including past test cores, presently committed cores and suggested future core types.

A Core: This is a reinforced MTR type with enriched uranium-aluminum fuel plates. It provides the reference set of measurements for full range of environmental conditions in Spert I, II and the full range of the environmental factor of pressure in Spert III.

B Core: The B core is an enriched uranium-aluminum core having removable plates so that several core configurations can be produced giving essentially different parameters.

In Spert I, II this provides for the variations in void coefficient and hydrodynamical characteristics to the extent determined simply by plate spacing over the full range of environmental conditions available in the two systems, for light water conditions. Possible use of this core in Spert II is intended as part of the heavy-water investigations in Spert II. The heavy-water, light-water series is intended to study the effect of prompt neutron lifetime.

C Core: This is an enriched uranium-stainless steel plate type core capable of withstanding the complete range of environmental conditions available in Spert III.

D Core: The D core is similar in purpose to the stainless steel C core except that zirconium will be used and the effect of the changed heat conduction properties of the cladding will be observed.

The A, B, C, D series provides three variations of plate characteristics and also a means for separating their effect from other parameter changes so that a fairly direct comparison of the effect of the heat transfer characteristics can be obtained.

E Core: The tests with this low concentration core, in combination with A, B, will permit a comparison of the behavior of more typical D_2O systems with typical light-water systems.

Low Enrichment Core: At the high end of the available range of enrichments, an enrichment change can be replaced in most considerations by a simple change in fuel concentration. Over the full range of enrichments this is not possible because so many factors are thereby affected. For example, the form in which fuel is used is in its gross characteristics determined by the enrichment factor. Similarly so are the reactor volume, details of moderator shutdown effects, and specifically nuclear shutdown due to the Doppler effect. Hence, these cores are also required for two reasons: firstly, because of the technology, and secondly, as a part of the study of the physics of reactor kinetic behavior.

F Core: This combination of A and E cores will permit study of space dependent effects within the core. However, this approach can be fruitfully pursued only after a basic understanding of the behavior of unmixed systems has been achieved. Its inclusion therefore is contingent on the developments in the basic program.

APPR Core: The APPR core is a stainless steel clad enriched uranium core which will be used in the Spert I reactor for early evaluation of the thermal properties of stainless assemblies under the limited range of conditions available.

BSR Core: As a possible alternative to the aluminum core for swimming pool reactors, ORNL is considering a stainless steel core which has operating safety advantages in the higher melting point material and in the fast-acting control system which can be used with the small core.

Under severe accident conditions when the possibility of complete failure of the control system must be taken into account, the advantage of high melting point may be offset by the shorter prompt neutron lifetime. In addition, no measurements are available on the void coefficients but the size of the core suggests that this may be small. Hence, in order to compare the two cores, static and dynamic measurements are required; the first to determine void coefficients, and the second to determine whether the gains offset the disadvantages. The kinetic measurements should include tests of the ability of the fast-acting control system to cut off short-period bursts and thereby check the predictions concerning the need for small reactivity compensation.

Tests on this core provide information for the basic program, pool-reactors and improvement of control systems. The latter point is an extension of the field of endeavor to include problem areas in the initiation phase.

Plutonium: A plutonium fueled core will provide a significant change in the delayed neutron fraction and therefore will permit some separation of behavior depending principally on the neutron kinetics from effects produced by shutdown mechanisms. The probable future importance of plutonium in a nuclear power economy makes information on cores of this type desirable.

Borax IV Type: This is needed to extend the Borax IV measurements as far as possible from the points of view of operating suitability of this type of core and understanding of shutdown mechanisms. The Borax IV core has a long thermal time constant for conduction from the fuel region to the metal water interface which suggests possibilities for separating prompt shutdown effects from the delayed thermal effects. This will be explored by applying a plastic coating to the Type A assemblies in order to delay heat conduction from the fuel to the moderator. The recent Borax IV results have considerably reinforced the possibility suggested earlier by Spert work that prompt effects are causing shutdown rather than the relatively delayed heat-transfer through the fuel plates. If this were to prove true, it would be enormously important to reactor design trends because it is conceivable that thick fuel plates are safer than thin ones.

Program Schematic. The way in which the cores will be used in the various reactor facilities is presented schematically in Figure 2. The approximate scheduling of these tests is indicated by the vertical time scale, contingent on starting dates for each facility. Those parts of the program to which the project is presently committed are enclosed in solid lines. Dashed lines are used for proposed extensions of the program and double dashed lines indicate the more tentative portions of a future program. The elapsed time assigned to each portion of the program cannot be determined with certainty. The discovery of unexpected phenomena in these reactors will very likely add considerably to the experimental program as has indeed been the case with the Spert I A core tests. Accommodation of special requests, such as those by duPont, will cause displacements in the indicated program. Therefore, the scheduling and sequence become less firm as the program progresses chronologically.

Test Outline: An outline of the test sequences to be followed on the various cores is presented in Figure 3 for those parts of the program that have been given detailed consideration. The more tentative portions of the program would follow similar sequences with appropriate modifications. Again these sequences are subject to

changes or additions which may become desirable in light of experimental results. The outline lists only the types of tests to be carried out rather than the detailed experimental procedures which are selected on a current basis as the program progresses.

Classification of Experiments: The relation of the experimental work to the needs of the safety program, that is, basic studies, class prototype tests, reactor prototype tests and services, is shown in Figure 4. It will be noted that although the distribution of plant equipment appears to be more or less uniform over the various portions of the program the division of effort is still largely concentrated on the basic investigation.

Summary of Proposed Facility Additions: For convenience the facilities and cores necessary to carry out the proposed extensions which constitute additions to present commitments are listed below:

Spert IV facility
Zirconium core
Low enrichment core
APPR core
BSR-type core
Plutonium core
Borax IV type core

These additions should be provided at intervals which are consistent with the approximate time schedule indicated in Figure 2, although not necessarily in the indicated sequence.

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SPERT PROGRAM SCHEMATIC
(OCTOBER, 1955)

IDO 16415

CORE / REACTOR	A. ALUMINUM PLATE TYPE	B. ALUMINUM ADJUSTABLE SPACING	C. STAINLESS STEEL	D. ZIRCONIUM	E. ALUMINUM LOW CONCENTRATION	F. COMBINATIONS OF A & E
I UNPRESSURIZED NO FLOW	I A α, I, ϕ_o, $\frac{dk}{dt}$	I B α, b, I, ϕ_o, $\frac{dk}{dt}$	I C α, I, ϕ_o, $\frac{dk}{dt}$	I D α, I, ϕ_o, $\frac{dk}{dt}$	I E α, I, ϕ_o, $\frac{dk}{dt}$	I F α, I, ϕ_o, b, $\frac{dk}{dt}$
II. PRESSURIZED LARGE TANK NO FLOW (FLOW ADDED-1956)	II A α, I, ϕ_o, P_o, $\frac{dk}{dt}$	II B α, I, ϕ_o, P_o, T_o	II C α, ϕ_o, P_o, T_o	II D α, ϕ_o, P_o, T_o	II E α, ϕ_o, P_o, T_o	II F α, ϕ_o, P_o, b
III. HIGH PRESSURE FLOW	III A TESTS AT HIGH PRESSURE (TEMP. LIMITED)	III B TESTS AT HIGH PRESSURE (TEMP. LIMITED)	III C FLOW α, ϕ_o, P_o, T_o, $\frac{dk}{dt}$	III D FLOW α, ϕ_o, P_o, T_o, $\frac{dk}{dt}$		

FACTORS TO BE VARIED SHOWN IN BOXES WHERE

- α - RECIPROCAL PERIOD
- I - INERTIAL LOAD
- ϕ_o - INITIAL POWER
- $\frac{dk}{dt}$ - RATE OF ADDITION OF REACTIVITY
- b - VOID COEFFICIENT
- P_o - SYSTEM PRESSURE
- T_o - SYSTEM TEMPERATURE

IN ALL TESTS MEASUREMENTS TO BE MADE OF POWER, ENERGY RELEASE, TRANSIENT PRESSURE, TEMPERATURE OF FUEL AND TEMPERATURE OF MODERATOR.

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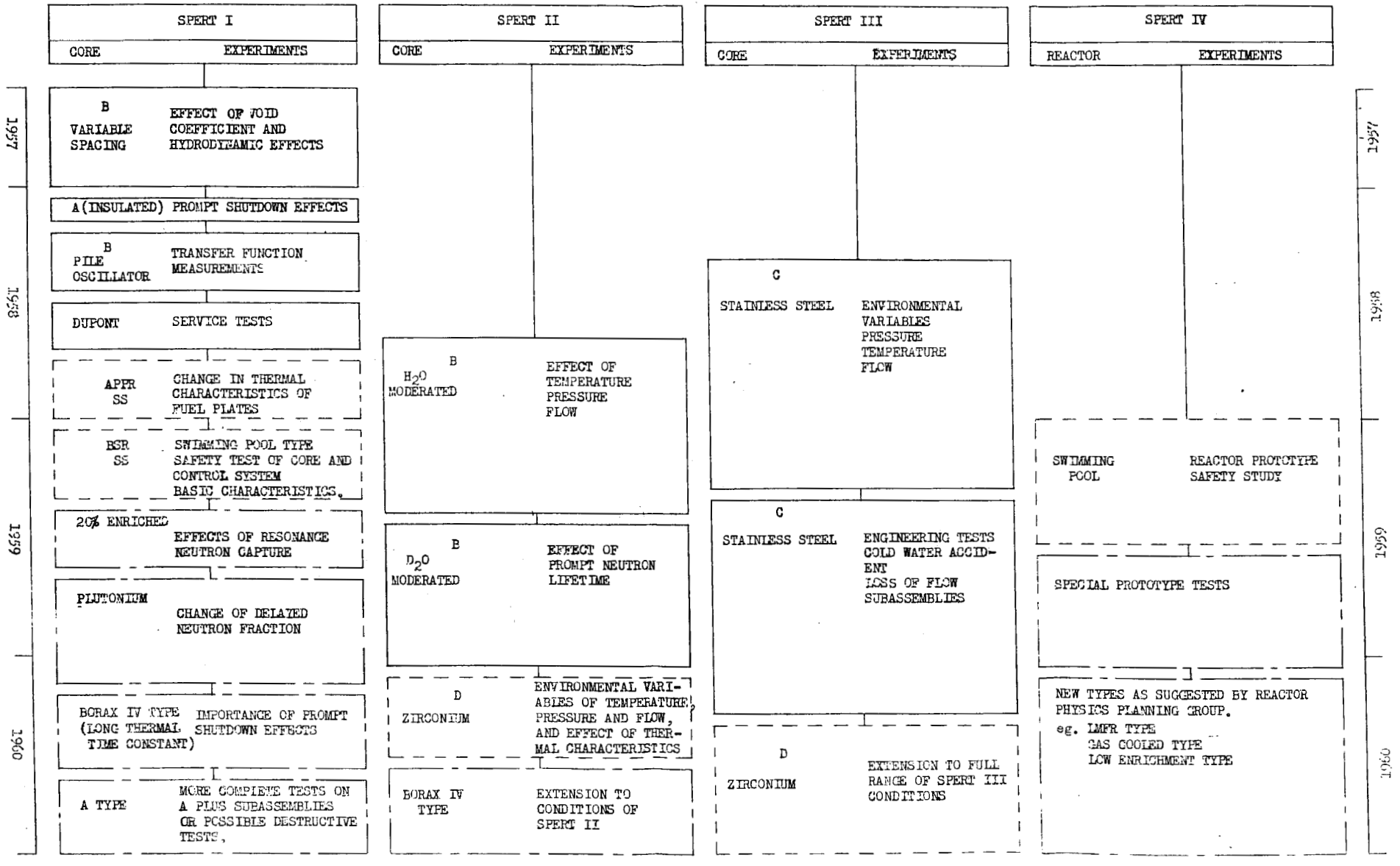
Figure 1

SPERT EXPERIMENTAL PROGRAM



SPERT EXPERIMENTAL PROGRAM OUTLINE

SPERT-I

CORE B

THREE CORE CONFIGURATIONS WITH METAL/WATER RATIOS STRADDLING MINIMUM CRITICAL MASS, VARIATIONS OF VOID COEFFICIENTS.

STATIC TESTS FOR EACH CORE CONFIGURATION	CRITICAL MASS ROD CALIBRATION TEMPERATURE COEFFICIENT VOID COEFFICIENT POWER CALIBRATION POWER COEFFICIENT FLUX MAP
TRANSIENT TESTS FOR ALL CONFIGURATIONS WITHIN STRUCTURAL CAPABILITIES OF CORE	20°C STEP SERIES 20°C RAMP SERIES BOILING STEP SERIES BOILING RAMP SERIES STEP SERIES AT SELECTED PERIODS WITH VARIABLE TEMPERATURE
OSCILLATION MEASUREMENTS FOR SELECTED CORE CONFIGURATIONS	PILE OSCILLATOR TESTS, TRANSFER FUNCTION MEASUREMENTS AS A FUNCTION OF OPERATING CONDITIONS STABILITY FOR SMALL OSCILLATIONS STABILITY FOR LARGE OSCILLATIONS BASIC CORE CONSTANTS

CORE A INSULATED

STEP TRANSIENTS WITH CONDUCTION HEAT TRANSFER SUPPRESSED TO EXAMINE IMPORTANCE OF PROMPT SHUTDOWN EFFECTS.

APPT CORE (S.S.)

CORE WITH DIFFERENT THERMAL CHARACTERISTICS AND HIGHER TEMPERATURE CAPABILITIES. TEST SERIES SAME AS "B"

BSR (S.S.)

PROTOTYPE SWIMMING POOL REACTOR - SAFETY TESTS
STEP TRANSIENTS
RAMP TRANSIENTS
STABILITY TESTS
CONTROL SYSTEM TESTS
TRANSFER FUNCTION

SPERT-II

PRE NEUTRON TESTS

SYSTEM CHECKOUT	PRESSURE TEMPERATURE FLOW LEAK CHECK FOR D_2O USAGE REACTOR CONTROL SYSTEM PLANT INSTRUMENTATION AND CONTROL

E CORE

STATIC TESTS (VARIABLE TEMPERATURE) H_2O AND D_2O MOD.	CRITICAL MASS FLOW TESTS ON CORE ROD CALIBRATION REACTIVITY COEFFICIENTS FLUX MAP (LOW POWER)
KINETIC TEST H_2O PRESSURE, TEMP. AND FLOW AS INDEPENDENT VAR.	POWER CALIBRATION AND COEFFICIENT STEP SERIES RAMP SERIES OSCILLATOR STUDIES STABILITY TESTS
KINETIC TESTS D_2O MOD. PRESSURE TEMPERATURE AND FLOW AS INDEPENDENT VARIABLES	SAME AS H_2O

D CORE (ZIRCONIUM - H_2O MODERATOR)

STATIC MEASUREMENTS AND LOW PRESSURE AND TEMPERATURE KINETIC TESTS - SERIES TO BE EXTENDED IN SPERT - III

STATIC TEST VARIABLE TEMP. H_2O	CRITICAL MASS ROD CALIBRATION REACTIVITY COEFFICIENTS POWER CALIBRATION AND FLUX MAP POWER COEFFICIENT vs FLOW
KINETIC TESTS PRESSURE, TEMP. AND FLOW AS VARIABLES	STEP TRANSIENTS RAMP TESTS OSCILLATOR TESTS STABILITY TESTS

SPERT-III

PRE NEUTRON TESTS

SYSTEM CHECKOUT	PRESSURE TEMPERATURE FLOW PLANT INSTRUMENTATION AND CONTROL REACTOR CONTROL SYSTEM

C CORE (STAINLESS IN H_2O)

STATIC TESTS	CRITICAL MASS AT 20°C APPROXIMATE ROD CALIBRATION FLOW TESTS ON CORE CRITICAL MASS vs TEMPERATURE 20°C TO 350°C
(EACH AT 20°C, 200°C, 300°C, 350°C)	ROD CALIBRATION VOID COEFFICIENT TEMPERATURE COEFFICIENT POWER CALIBRATION AND FLUX MAP
KINETIC TESTS	STEP TESTS RAMP TESTS STABILITY TESTS OSCILLATOR TESTS (ENVIRONMENTAL VARIABLES 20°C TO 350°C, 0 TO 2500 psig, 0 TO 20,000 gpm WITH AND WITHOUT EXPANSION SPACE.)
POWER TESTS	STEPS RAMPS LOSS OF FLOW COLD WATER BLIND STARTUP OSCILLATOR
SPECIAL TESTS	SUB-ASSEMBLIES STEAM RODS

ADDITIONAL TESTS AS INDICATED BY TEST RESULTS

Figure 3

CLASSIFICATION OF EXPERIMENTS

BASIC STUDIES

REACTOR AND CORE	PARAMETERS INVESTIGATED
I - A	BASE POINT FOR PARAMETER INVESTIGATIONS STATIC MEASUREMENTS RECIPROCAL PERIOD INITIAL TEMPERATURE (20°C TO BOILING) INITIAL POWER (LIMITED) HYDRODYNAMIC HEAD (LIMITED) RAMPS (RATE AND INITIAL POWER) STABILITY (vs TEMPERATURE AND HEAD) SURFACE TENSION (EXPLORATORY) DISSOLVED GAS (EXPLORATORY)
I - B	VARIATION OF VOID AND TEMP. COEFFICIENTS HYDRODYNAMIC EFFECTS OF CHANNEL SPACING OSCILLATOR TECHNIQUE FOR STATIC AND DYNAMIC CHARACTERISTICS.
I - A INSULATED	PROMPT SHUTDOWN EFFECTS
20% ENRICHED	NUCLEAR CHARACTERISTICS (RESONANCE CAPTURE CORE SIZE AND SMALL VOID COEFFICIENT)
PU	NUCLEAR CHARACTERISTICS (DELAYED NEUTRON FRACTION)
I - A TYPE	FILL IN AS NEEDED FOR COMPLETION OF BASE POINT DATA CONSEQUENCE PHASE OF RUNAWAY (DESTRUCTION OF SUB-ASSEMBLIES)
BORAX IV TYPE	IMPORTANCE OF FUEL PLATE THERMAL TIME CONSTANTS
II - B	PROMPT NEUTRON LIFE TIME PRESSURE - TEMPERATURE - FLOW
II & III - D	FULL RANGE OF ENVIRONMENTAL VARIABLES PRESSURE, TEMPERATURE, FLOW - Zr PLATES
III - C	FULL RANGE OF ENVIRONMENTAL VARIABLES FOR SS PLATES

CLASS PROTOTYPE

REACTOR CORE	REACTOR CLASS REPRESENTED
I - A	SWIMMING POOL (APPROXIMATE)
20%	GENEVA - RES. REACTORS
III - C	PRESSURIZED WATER REACTORS SS CORES, OPERATING PROBLEMS, ETC.
II & III - D	PRESSURIZED WATER REACTORS Zr CORES

REACTOR PROTYPE

BSR	SWIMMING POOL
	SMALL RESEARCH REACTORS
	SPECIAL TYPES

SERVICE

FUSE TESTS

SUB ASSEMBLIES

BURSTS FOR EQUIPMENT TESTS