MASTER

DYNAMIC PROPERTIES OF HETEROGENEOUS
WATER REACTORS

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PHILLIPS
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DYNAMIC PROPERTIES OF HETEROGENEOUS WATER REACTORS

by

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ERRATA SHEET

IDO-16701

1. Please change the order of authors on the front cover to read: "S. G. Forbes, W. E. Nyer".

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2. IDO-16701, "Dynamic Properties of Heterogeneous Water Reactors" by S. G. Forbes and W. E. Nyer, was erroneously referenced as IDO-16659 on page 23 of Nuclear Safety, Volume 2, Number 3. To our knowledge, this is the only time the number IDO-16659 (now cancelled) has been used.

Thomas J. Wadsworth
Thomas J. Wadsworth

THIS REPORT WAS PRESENTED TO THE SIXTH
INTERNATIONAL CONGRESS ON NUCLEAR ENERGY
IN ROME IN 1959.

Part I is a reprint of the paper published by the
Congress, and Part II contains the oral presentation
by Dr. S. G. Forbes. The report is issued at this
time to provide a record of the oral presentation.

DYNAMIC PROPERTIES OF HETEROGENEOUS WATER REACTORS

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I. PAPER SUBMITTED

Sixth International Congress on Nuclear Energy

1. INTRODUCTION

The purpose of the Spert (Special Power Excursion Reactor Tests) project is the improvement of reactor safety through the experimental investigation and understanding of reactor kinetics. The importance of reactor kinetics in the achievement of controlled nuclear power was recognized as soon as the possibility of a nuclear chain reaction became evident. In the May, 1939 issue of Nature, F. Adler and H. Von Halban, Jr. pointed out the danger of a nuclear runaway and suggested a method for making a reactor self-stabilizing. In the ensuing 20 years nuclear fission has developed into an important energy source, and the basic controllability of reactors has been well established. Still, in many cases the ease of operation and safety of a reactor are principally determined by its inherent kinetic properties. Reactor kinetics remains, therefore, as a central problem in reactor safety which affects the future development of practical nuclear power. For this reason the Spert program of reactor kinetic experiments has been undertaken by the U. S. Atomic Energy Commission. A review of these studies will be presented in this paper with particular emphasis on the applicability of the results to reactors of various types.

Some of the factors which influence reactor kinetic behavior are (1) the delayed neutron fraction, (2) prompt neutron lifetime, (3) void and temperature coefficients, and (4) operating conditions. The Spert reactors have been designed to provide a wide range of these variables in order to determine their individual influence and importance.

Three heterogeneous water reactors have been constructed for the experimental program. Spert I, which has been in operation since July, 1955, consists of an unpressurized open vessel 4 ft in diameter and 15 ft high. The vessel is filled with light water which serves as moderator, reflector, and coolant for the core under investigation. The Spert II reactor is a closed system which can operate up to 375 psig and 400°F, with flow of 20,000 gpm through the core. The reactor is designed to operate with either light or heavy water and will be completed in the summer of 1959. Spert III, which has been in operation since December, 1958, is a high pressure, high temperature, light water system designed to simulate the conditions encountered in a power reactor of either the pressurized water or boiling water type. Maximum design operating conditions are 2500 psig, 650°F, 20,000 gpm flow, and 60 Mw power for short intervals. Since these experiments are intended to determine the extreme conditions under which a reactor will be safely self-limiting, it is necessary to conduct all operations from a control point one-half mile away.

The reactor experiments can be divided into four major classifications: (1) static experiments to determine such core characteristics as void and temperature coefficients; (2) step tests in which the system is suddenly made supercritical; (3) ramp tests in which reactivity is added to the reactor at a constant rate; and (4) stability tests which may involve either spontaneous or externally

excited oscillations. These tests are performed under various conditions of temperature, pressure, and coolant flow on cores of differing design. Five highly enriched plate-type cores have been used in the Spert I experiments, four of aluminum and one of stainless steel. These cores differed principally in their void coefficients and prompt neutron lifetimes.

2. GENERAL FEATURES OF STEP TESTS

Following a reactivity step the reactor power rises exponentially until the energy released by fission begins to cause physical changes in the reactor, such as thermal expansion of the fuel or moderator, which reduce the reactivity of the system. The power passes through a maximum and approaches a lower equilibrium value. Almost all of the fission energy is retained in the fuel plates in the form of heat. In the most severe transient the fuel plate temperature reached 600°C, which is nearly the melting point of aluminum. The initial period of this test was 5 msec and the peak power exceeded 25 Mw.

For long period transients resulting from small reactivity steps the peak power increases approximately linearly with the inverse period. Above prompt critical the power increases more nearly as the square of the inverse period. Since the power increases monotonically with increase in the reactivity step, it might be expected that the energy released would also increase uniformly with increasing reactivity, but this is not correct. For reactivity steps up to about 50 cents the energy release increases, but larger additions of reactivity up to one dollar actually result in a decrease in burst energy. Beyond prompt critical the energy release again increases uniformly with reactivity addition. From the point of view of energy release, which is frequently the major factor in accident severity, a self-limiting prompt critical accident may be safer than one resulting in a period of about 5 sec.

These results can be understood in a qualitative way by consideration of the neutron kinetics of a chain reacting system. These kinetic properties are determined by the abundances and decay constants of the delayed neutron precursors which are properties of the fissile material, and by the prompt neutron lifetime, which is determined by the core design. Because the delayed neutrons tend to lag behind the power, it is possible to halt a power burst without reducing the excess reactivity of the system to zero; i.e., the reactivity change required to stop a runaway can be very much less than the initiating reactivity disturbance. It has been shown both analytically and experimentally that for reactors in which the self-induced reactivity changes are simple functions of the energy released, the reactivity change required to stop a runaway passes through a maximum value of about 20 cents for initial reactivities of 50 cents and passes through a minimum for reactivity steps of about one dollar, i.e., prompt critical. For initial reactivities somewhat above prompt critical, the reactivity change needed to halt the runaway approaches the prompt excess reactivity. The minimum at prompt critical is most pronounced for reactors which have short prompt neutron lifetimes such as fast reactors

For reactors having prompt neutron lifetimes of 500 microsec or longer, such as heavy water or graphite reactors, this minimum degenerates to an inflection point. For the Spert reactors the value at the minimum is about 10 cents, or a factor of 10 below the injected reactivity. Since these reactivity changes are approximately proportional to the energy release, the functional

dependence of the energy release on the initial reactivity step should be essentially the same as that just described, which is in complete agreement with the experimental results cited above. This means that a light water or fast reactor which will shut itself down safely on a transient having a 5 sec initial period will be safely self-limiting somewhat beyond prompt critical, provided the self-shutdown process does not change radically. Although these results depend to some extent on the nature of the physical processes which produce self-limiting behavior, they are principally properties of delayed and prompt neutron kinetics, and similar results are therefore to be expected for reactors of all types.

One advantage that can be gained from the properties of the neutron system arises from the fact that the reactivity change necessary to stop a runaway becomes smaller if it is made very rapidly. From a reactor control point of view this means that a fast-acting, weak control rod can be used to arrest a power burst. Although the excess reactivity must eventually be reduced to zero, this can be accomplished more slowly.

3. TYPICAL RESULTS OF RAMP EXPERIMENTS

The most likely nuclear accidents are those in which the reactivity increases with time, and it therefore becomes important to determine the reactivity addition rates and initial power for which the reactor will be safely self-limiting. Due to the continuous injection of reactivity, the rate of power rise does not have a characteristic initial value, but increases from zero at the start to some maximum value and then, in response to the reactivity changes internal to the reactor, decreases to zero again at the time of peak power. The controlled variables in tests of this type are the initial power and reactivity injection rate. The minimum period is an observed quantity along with power, energy, etc.

The initial power is a relatively unimportant parameter in a ramp accident since a change of a factor of 10^5 in the initial power produces only about a factor of 10 change in the peak power. However, the peak power is approximately proportional to the ramp rate. If a ramp burst is characterized by its minimum period the burst behavior is equivalent to a step burst of this same initial period. Analytical methods have been developed for predicting the minimum period for a given initial power and ramp rate. The highest ramp rate used in these experiments, about 50 cents/sec, although higher than those normally encountered in control rod drive systems, is still an order of magnitude below the maximum safe limit even for very low initial power levels. Thus it appears that, with regard to permissible rates of reactivity addition, the startup procedures for many reactors are unnecessarily conservative.

4. SELF-SHUTDOWN MECHANISMS

From the experimental observation that the reactivity change per unit of energy released is essentially constant for all values of the reactor period, it might be inferred that a single physical process is responsible for the self-limiting behavior of the reactor. A more detailed analysis reveals, however, that at least three processes are important in these experiments.

For long period transients, the primary shutdown process is heating of the moderator by thermal conduction. As prompt critical is approached, the power rise is too rapid to permit significant heat transfer to the water, and thermal expansion of the fuel plates becomes the predominant effect. For reactor periods shorter than about 50 msec the temperature of the plates exceeds the boiling point of water and the formation of steam appears to be the major factor. Because of the lack of detailed information on the nature of transient boiling of water, this process is not well understood, but an empirical description of the boiling process has been found to correlate the experimental results sufficiently well to permit reasonable predictions of reactor transient behavior for short periods.

The reactivity changes resulting from steam formation are not related in a simple manner to the energy released by fission in the fuel plates for two reasons: (1) the boiling process has a threshold, the boiling point, below which no steam is formed; and (2) the transfer of heat from the plates and the growth of steam voids is a very non-linear function of the plate temperature. The result of these effects is that during a burst the reactivity change increases approximately as the square of the energy release. Analysis of the effect of such a process on reactor self-shutdown reveals that the peak power and energy release for a given initial period vary approximately as the reciprocal of the square root of the void coefficient of reactivity. This prediction has been confirmed by experiments on cores having widely different void coefficients. This weakened dependence of burst behavior on the void coefficient will appear in any reactor in which the reactivity changes increase more rapidly than the energy release, and the reactor designer's control of dynamic behavior is lessened in reactors having this characteristic.

5. REACTOR STABILITY

In some of the Spert tests, power oscillations appeared which became sufficiently violent to approach reactor destruction. The possibility of reactor instability is well known and the conditions for the occurrence of oscillations in feedback systems have been well developed in circuit analysis. However, a nuclear reactor is a non-linear device, and in the case of heterogeneous water reactors the feedback loop is also a non-linear system of essentially unknown properties when transient boiling is involved. The observed instability was always associated with the existence of steam voids in the core, and instability introduced by cyclic collapse and growth of steam bubbles was expected. The extreme rapidity and completeness of the void collapse which occurred was not anticipated, however. The Spert stability experiments have been limited in extent and confined to a particular set of experimental conditions, so the detailed results cannot be applied to other reactors as yet. The instability problem does not appear to be difficult to cope with if it is given proper consideration in reactor design, but it must be recognized that under some conditions the potential danger from unstable reactor behavior can be as great as any of those normally considered in reactor evaluations. While it is true that the power level at which such oscillations may appear may be predicted by means of pile oscillator tests of the externally excited type, the non-linear nature of the reactor system makes it possible to predict the eventual course of self-induced oscillations. Indeed, it can be generally stated that the conditions encountered in a serious power excursion are much too severe to be investigated by any experiment of a continuous type, and transient experiments must be resorted to for the fullest understanding of reactor accidents.

The proper design of a reactor for maximum safety for ramp, step, and instability accidents may lead to conflicting requirements. The problems can be illustrated by considering, for example, the influence of the temperature coefficient of reactor safety. For a step initiated runaway, the energy release can be reduced by having a large negative temperature coefficient, even though the self-shutdown effects exhibit time delays between energy released and reactivity change. However, if the reactor is to be operated at high temperature, it is necessary to provide sufficient excess reactivity to overcome the reactivity loss in going from room temperature to operating temperature. Thus, a large negative temperature coefficient requires a large available excess reactivity, and the size of the maximum possible reactivity accident is correspondingly increased. Excess reactivity may also be required for fuel burnup or xenon override, but to some extent a large negative temperature coefficient may actually increase the reactor hazard.

From the point of view of stability, a large negative temperature coefficient increases the feedback and stabilizes the system, provided that there is no phase shift in the feedback. However, if time delays exist, the resulting phase will decrease the stability of the system and a large negative temperature coefficient will increase the tendency toward unstable behavior. Since many of the self-limiting reactivity effects do exhibit time lags, it is frequently desirable to make these reactivity coefficients as near zero as possible to prevent unstable behavior.

From these considerations it can be seen that a large negative reactivity coefficient is not necessarily desirable and may, in fact, be very dangerous.

6. EXTENSION OF THE PROGRAM WITH SPERT II AND SPERT III

Up to the present time all of the experimental work has been conducted in the Spert I reactor which provides only limited control of the important variables in reactor kinetic studies. The Spert II and III reactors, which will begin experimental operation in the near future, were designed to extend the range of these investigations.

The Spert II reactor will be operated with both light and heavy water as a moderator in order to determine experimentally the importance of prompt neutron lifetime on reactor kinetic behavior. The provision for control of pressure, temperature, and flow will permit a much wider investigation of reactor instability and will also extend the range of variables for step and ramp tests beyond those available in Spert I.

The Spert III experiments will be of particular interest in the field of power reactors, and it is for this reason that Spert III has taken precedence over Spert II in the construction schedule. The design parameters of the Spert III reactor such as pressure, temperature, coolant flow, heat flux, and power density were all selected to have as wide a range of flexibility as present technology would permit. The reactor, therefore, incorporates many special features which represent a significant extension of reactor operating conditions. In addition to the mechanical problems encountered with a high pressure, high temperature system, there are rather severe nuclear problems as well. These are brought about principally by the very rapid expansion of water as the temperature approaches the critical

point. Heating the reactor from room temperature to maximum operating temperature reduces the density of the moderator in the core by nearly a factor of two. The rate of density change at this temperature is about 1 %/°C and the resulting negative temperature coefficient is about 25 cents /°C. This presents an interesting operating problem since a drop in the inlet water temperature of only 4°C will make the reactor prompt critical.

In addition to tests of the type described for Spert I, the Spert III experiments will also include simulated accidents of the engineering type such as cold water accidents and loss of flow accidents. The reactor pressure can be controlled independently of the temperature, which permits experiments of either the pressurized water or boiling water type. Pressurization will allow the boiling process in self-shutdown to be suppressed in order to explore more fully the relative importance of boiling and other processes in reactor kinetic behavior. The results of the Spert III experiments should find wide applicability to the field of power reactors.

II. ORAL PRESENTATION

Sixth International Congress on Nuclear Energy

The Spert program has been principally oriented toward the problems of inherent reactor safety in cases of reactor runaway accidents; i.e., the type of reactor accidents which are usually treated in safety reports in an effort to evaluate the maximum hazards to the reactor plant and to the general public. The characteristic accident which is so treated is one in which relatively large reactivity additions occur with the reactor at a low initial power. In such a case the inherent properties of the reactor itself determine the course of the accident and its consequent hazards. One of the objectives of the Spert experimental work has been to determine how much reactivity addition, either instantaneous or as a linear function of time, can be safely tolerated by reactors of various types, and what factors can be varied in order to increase the size of this maximum tolerable reactivity disturbance.

The description of the reactors used in these experiments and the detailed data obtained have been presented in other reports and are outlined in the paper printed in your program. These will not be repeated here. Instead, I would like to concentrate on those aspects of the paper which deal with the general results of the program and which have a bearing on the problems facing a design engineer who wishes to apply these results to a particular case. For this purpose I would

like to consider a reactor whose basic design has been largely determined by operating considerations, and point out the problem of what gains can be made in inherent safety by the relatively small design modifications which remain at the disposal of the design engineer.

The theory upon which the following discussion is based has been substantiated by experimental results for a wide variety of cases and the specific characteristics of individual reactors are not expected to materially alter the general conclusions.

Figure 1 shows the typical behavior of the reactor power following either a sudden increase in reactivity, referred to as a reactivity step, or a reactivity addition at a constant rate, referred to as a ramp. The abscissa is time and the ordinate is the reactor power on a logarithmic scale. Following a step increase in reactivity, the reactor power rises exponentially with a constant period until the energy

Fig. 1 Reactor power vs increases in reactivity

released by fission begins to cause a decrease in reactivity. As these reactivity losses continue, the power passes through a maximum and subsequently decreases to a low value. The reactivity loss per unit of energy release will be referred to as the shutdown coefficient.

In the case of a ramp addition of reactivity, the reactor period decreases continuously until the rate of reactivity loss due to energy release is equal to the rate of reactivity insertion. At this point the reactor period has its minimum value and the power behavior from this point on is much the same as for a step transient having this same period. Simple analytical means have been found for obtaining this minimum period from the ramp rate and prompt neutron lifetime and it is therefore possible to determine the maximum permissible ramp rate in terms of the maximum permissible step insertion of reactivity.

The hazards associated with large power excursions arise primarily from the energy released and this quantity will be taken as the index of accident severity. For a system with a prompt negative shutdown coefficient the energy release will vary with the size of the reactivity step in the general manner indicated in figure 2. Here the size of the reactivity step is shown in dollar units on the abscissa and the energy release is shown in relative units on the ordinate. In the vicinity of prompt critical (one dollar on the scale) the energy release is dependent on the prompt neutron lifetime, as indicated by the three curves. The top curve represents a long lifetime reactor such as a heavy water system. The middle curve represents a short lifetime thermal neutron system, and the bottom

Fig. 2 Energy release vs reactivity step.

curve represents a fast reactor. For reactivity steps somewhat above prompt critical, the energy release is independent of the prompt neutron lifetime.

The maximum tolerable energy release will be taken to be that which will raise the temperature of the core to the melting point since this represents a hazard to the reactor itself even though such an accident might not be serious enough to constitute a hazard to the general public. Such a limit on the energy release is represented by the horizontal dashed line labeled MAXIMUM ENERGY on figure 2. The intersection of this line with the energy release line then determines a maximum permissible reactivity step. This intersection point can be moved to the right, i.e., in the direction of increasing safety, in either of two ways. The maximum energy line can be moved up by making design changes which increase the maximum permissible energy release. Alternatively, the energy release line can be moved down by increasing the size of the shutdown coefficient. The reactor designer will usually have only limited control of these factors and consequently the improvement of the permissible step will not be large. However, these gains may actually result in a significant improvement in safety when ramp insertions are considered because the permissible ramp rate increases as the square of the permissible step. This is illustrated in figure 3. Here ramp rate is shown as a function of the equivalent step with neutron lifetime as a parameter. At the top of the figure is a simple relationship between a ramp and the equivalent step, which is that the ramp rate in dollars per second is approximately equal to the square of the prompt step in dollars divided by 30 times the neutron lifetime, ℓ , divided by the delayed neutron fraction, β . The

Fig. 3 Ramp vs equivalent step

factor of 30 in the denominator is approximate and will vary somewhat for particular situations, but for a very wide variety of cases this relationship will be correct within a factor of two. Aside from small variations of the constant, this is a general relationship which applies equally well to fast or thermal reactors.

Note that marginal gains in the permissible step represent more substantial gains in the permissible ramp because of the squared relationship. Note also that, for a given permissible step, a reactor with a short neutron lifetime will have a higher permissible ramp than a reactor with a longer neutron lifetime. This somewhat unexpected result does not imply that design changes which reduce the neutron lifetime will necessarily result in a gain in permissible ramp, because in some cases these changes will also reduce the permissible step and the advantage may be lost.

Because of this relationship between ramps and steps it is extremely important to consider even small gains that can be made in the maximum tolerable reactivity step because the corresponding permissible ramp rates increase so rapidly. If sufficient gains can be made in this direction, the maximum possible accident for a reactor will be determined not by the amount of excess reactivity that may be available, but by the maximum possible addition rate. The degree to which design changes influence the maximum permissible reactivity additions will vary for reactors of different types. I would like to discuss the degree of influence for three different kinds of reactors characterized by the nature of their shutdown coefficients. These three types of shutdown coefficients may be identified as:

- (1) prompt, negative, constant
- (2) delayed, negative, constant
- (3) prompt, negative, energy dependent

An example of a reactor having a prompt, negative, constant shutdown coefficient is one in which the negative temperature coefficient is associated with the temperature of the fuel itself so that there is no delay between the energy release by fission and the appearance of negative reactivity effects. This is the case already considered in the foregoing discussion. Examples of reactors of this type are TRIGA and GODIVA.

A more common type of reactor is one which has a delayed negative shutdown coefficient. An example is a reactor in which the negative temperature coefficient arises from heating some part of the core structure, and the time delay is a result of the transfer of heat from the fuel region to the core structure. In this case the effective shutdown coefficient decreases with the decreasing period. As a result, the effect of changes in design factors is much weaker than for the case with a prompt negative coefficient.

A third case of practical interest is a reactor which has a prompt shutdown coefficient that increases with energy release. An example would be a reactor whose temperature coefficient increased with increasing temperature. In this case the reactivity loss will increase more rapidly than the energy release. As a result, significant gains in the permissible step can be achieved by minor design modifications. Due to the square law dependence between steps and ramps, the achievable gains in the permissible ramp are even more marked. In fact, the permissible ramp rates can be extended so far that they become mechanically difficult to achieve in practice. For example, the computed permissible ramp rate for the Spert I-A core is about 12 dollars/sec. The Spert I reactor contains

a transient rod especially designed to provide rapid reactivity injections for experiments of the step type. The maximum rate of reactivity addition provided by this fast rod is about 25 dollars/sec, which is only a factor of two greater than the permissible ramp rate for this reactor. Thus a 50% increase in the permissible step would result in a factor of two increase in the permissible ramp and the reactor would be inherently safe against reactivity additions even from this very fast rod.

To summarize the types of reactors which have been considered, the three types of shutdown coefficients are listed on figure 4 along with the relative influence of design changes. For a prompt, negative coefficient, design changes have a moderate influence on the permissible reactivity addition. For the delayed negative coefficient this influence is weak. For a reactivity coefficient which increases with energy release the influence of design changes is relatively strong. The shutdown coefficient of the Spert reactors has been found to be of the latter energy dependent type. Thus, the permissible steps and ramps in the Spert reactors, and others similar to it, can be substantially improved by proper adjustment of the design factors.

It is apparent, that the degree of the reactor designer's control over maximum permissible reactivity disturbances will depend strongly on the reactor type. It should also be pointed out that the magnitude of the shutdown coefficient is generally characteristic of the particular type of reactor. It should not be assumed, for example, that decreasing the prompt neutron lifetime by changing from a thermal reactor to a fast reactor will result in a gain in maximum permissible ramp rate because the shutdown coefficients for fast reactors are often very much smaller than for thermal reactors. The discussion of the effects of various factors should be considered as valid only for those variations which are made without changing the basic reactor type.

The two main conclusions of the foregoing discussion are: (1) small gains in the permissible reactivity step result in important gains in the permissible

SHUTDOWN COEFFICIENT	DEPENDENCE ON DESIGN MODIFICATIONS
prompt negative constant	moderate
delayed negative constant	weak
prompt negative energy dependent	strong

Fig. 4 Types of shutdown coefficients

addition rate; and (2) in many cases the permissible addition rate can be made to exceed rates that are likely to be encountered in practice. Thus, the runaway accident hazard will be determined by the reactivity addition rate rather than by the amount of reactivity available.

The applicability of these general concepts to specific cases requires some knowledge of the nature of the reactivity coefficient which is a function of the properties of a core and the conditions under which the reactor is operated. As an example of the influence of operating conditions on the shutdown coefficients, the temperature coefficient in Spert III at the maximum design temperature of 350°C is a factor of 10 greater than at room temperature.

Spert I has provided a means for examining many of the factors which influence the nature of the shutdown coefficient. The Spert II and Spert III reactors provide means for extending these studies, especially the range of operating conditions. The Spert III reactor is essentially a high pressure, high temperature, power reactor of either the pressurized water or boiling water type. The Spert III experiments include step transients, ramp transients, stability studies, and simulated plant accidents such as loss of flow and cold water injection. These experiments will be of particular interest to designers of power reactors.

III. REFERENCES

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