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SELF-LIMITING EXCURSION TESTS OF A
HIGHLY ENRICHED PLATE-TYPE D₂O-MODERATED REACTOR
PART 1. INITIAL TEST SERIES

MASTER

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U. S. ATOMIC ENERGY COMMISSION

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ABSTRACT

Self-limiting power excursion tests have been made to investigate the dynamic response of the Spert II BD-22/24 expanded D₂O core. This D₂O-moderated core is comprised of 3-in.-square x 24-in.-long, highly enriched, uranium-aluminum, plate-type fuel assemblies spaced on 6-in. centers. The power excursions were initiated by stepwise reactivity insertions with the system at ambient temperature and atmospheric pressure, with no forced coolant flow, and with approximately a 50-in. D₂O head over the core. The reactivity insertions ranged from about 40 cents to 3 dollars and resulted in initial asymptotic reactor periods of from about 10 sec to 50 msec. The basic self-shutdown mechanisms are apparently thermal in nature. For initial periods longer than 600 msec, the self-shutdown is due to non-boiling processes. For initial periods less than 600 msec, steam formation contributes to the self-shutdown and is apparently the predominant shutdown mechanism for power excursions with periods less than 100 msec. Limited mechanical damage to fuel plates is not obtained in tests with periods greater than 70 msec. Extrapolation of the fuel-plate surface-temperature data indicates that partial melting of the fuel plates might occur in tests with periods of 30 msec or less. The high fuel plate surface temperature required for melting is obtained as a result of the reduced heat transfer associated with partial film boiling and not as a result of excessive energy release.

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SELF-LIMITING EXCURSION TESTS OF A
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I. INTRODUCTION

The evaluation of the safety of reactor systems by means of both experimental and theoretical studies has long been a major effort in the nuclear industry. The purpose of these studies is to gain knowledge of those parameters which are important in reactor dynamics and to predict the consequence of possible accidents. One of the basic purposes of the Spert program is to determine experimentally and/or analytically those parameters which strongly influence the dynamic behavior of reactors by extreme power-excursion testing of various types and classes of reactors which vary significantly in their static characteristics. Thus far in the Spert program, experimental work has been conducted with light-water-moderated, -reflected, and -cooled reactor cores. These tests have primarily involved the use of highly enriched, plate-type fuel, either aluminum clad [1] or stainless steel clad [1,2,3], but recently included a low-enrichment UO₂ fuel-rod-type core [4]. These cores, which differed quite markedly in many static characteristics, have been dynamically tested over a wide range of initial parametric conditions.

An important characteristic common to all these cores was the prompt neutron lifetime which, although it ranged from $\approx 100 \mu\text{sec}$ to $\approx 20 \mu\text{sec}$, was typical of light-water-moderated systems. In order to determine the effects of a longer neutron lifetime on the dynamic behavior of a reactor and for reasons basic to the study of specific shutdown mechanisms, the Spert II reactor was designed to operate with heavy water as a moderator-coolant and reflector. The selection of heavy water as the moderator-coolant-reflector was made because it has nuclear properties that will generally result in a reactor with a much longer prompt neutron lifetime than a light-water system with only minor changes in the physical properties of the moderator-coolant. The effects of changing the prompt neutron lifetime should then be apparent from comparison of the kinetic behavior of the heavy-water system with that of a light-water system. This comparison may provide a measure of the importance of neutron lifetime on reactor safety. Marked changes in the dynamic behavior of a reactor are to be expected if the prompt neutron

lifetime is changed significantly since the rate of power rise is proportional to the prompt reactivity insertion and the prompt neutron lifetime for a super-prompt critical reactor.

The purpose of this report is to discuss the results and conclusions which can be derived from the data obtained during the first series of self-limiting power excursion tests with the BD-22/24 expanded D₂O core, a highly enriched and D₂O-moderated system, whose configuration is typical of many heavy-water research reactors such as the MITR [5], DIDO [6], and CP-5 [7]. Considerable information is available regarding the self-regulation of light-water-moderated, -reflected and -cooled reactors in a runaway condition [8], including two planned [9,10] and one unplanned [11] destructive incident. The tests discussed in this report, however, constitute the first self-limiting power excursions conducted on a heavy-water-moderated and -reflected core in which the reactivity insertions were large enough to render the reactor super-prompt critical.

The data presented here include only those tests initiated by stepwise insertions of reactivity with the system at ambient temperature ($\approx 22^{\circ}\text{C}$), at atmospheric pressure, with no forced coolant flow, and a water head of approximately 50 in. above the core. Since these data were obtained under a limited set of initial conditions, they will not provide answers to all the questions posed regarding the dynamic performance or safety of D₂O reactors. The data will, however, allow an evaluation to be made of some performance aspects of a long prompt neutron lifetime reactor and constitute a base to which data from additional tests at different initial conditions may be compared in order to evaluate the effects of parametric changes.

II. BRIEF DESCRIPTION OF THE FACILITY

1. PLANT DESCRIPTION

The Spert II facility consists of the reactor building and its control center which are separated from each other by a distance of approximately 1/2 mile. The reactor building encloses the reactor vessel, coolant flow system, heavy-water system, and the necessary auxiliary equipment for operating a reactor of this type. The control center building houses the controls necessary for remote operation of the reactor and plant from a distance of approximately 1/2 mile during nuclear operation.

The reactor vessel and coolant flow systems are designed for operation at pressures up to 375 psig, temperatures up to 400°F, and coolant flow rates to 20,000 gpm, either upward or downward through the core. A detailed description of the plant is given elsewhere [12].

2. REACTOR DESCRIPTION

The Spert II reactor shown in Figure 1 consists of the reactor core with the control rods and drives, a transient rod with its drive, the reactor core structure, and the reactor pressure vessel with a full-diameter flanged top head. Inlet and exit coolant nozzles, blind flanged for this series of tests, penetrate the reactor vessel bottom head.

2.1 Reactor Components

The reactor vessel is a stainless-steel-clad, carbon steel vessel, 10 ft ID x about 16 ft high. Flanged nozzles for the egress of instrument leads are provided in the top head and at several elevations around the vessel shell. The reactor core is supported by an internal structure fastened to the bottom of the vessel. An aluminum flow skirt, 42 in. ID with 1/2-in. wall thickness, surrounds the core to a height of approximately 12 in. above the top of fuel. The core support grid is divided into four symmetrical quadrants by a 3/4-in.-thick aluminum rod guide cross.

The grid has 88 lattice positions available for the loading of fuel assemblies. The coordinate numbering system for these lattice positions, shown in Figure 2, can be described by considering the core to be divided into four quadrants (N, E, S, and W) by the cross. Numbers are assigned to each lattice position by con-

FIG. 1 SPERT II REACTOR VESSEL ASSEMBLY.

FIG. 2 SPERT II CORE COORDINATE SYSTEM.

sidering each individual quadrant separately as the first quadrant of a normal cartesian coordinate system with the origin at the center of the core. A given position is described by quadrant and by X- and Y- coordinates within the quadrant, eg, W 4-2. Thus, like-numbered positions have quadrant symmetry. The reactor is controlled by means of eight control rods which operate in the positions shown in Figure 2. In the center of the core is a blade-type transient rod.

The fuel assembly is the Spert type BD assembly shown in Figure 3. Basically, the fuel assembly consists of a 3-in.-square x 27-5/8-in.-long aluminum fuel assembly can, an aluminum lower end box with lockings lugs, two grooved side plates, a lifting bail, and 22 fuel plates. The lifting bail and the lower end box axially constrain the fuel plates in the fuel assembly can.

The fuel is contained in flat plates, each consisting of seven grams of U-235 alloyed with aluminum melting stock to produce a "meat section" 0.020 in. thick, 2.45 in. wide, and 24 in. long. This fuel section is clad with 6061 aluminum alloy to produce a plate 0.060 in. thick, 2.704 in. wide, and 25-1/8 in. long. A fuel assembly, when fully loaded with 22 of these plates, will contain 154 g of U-235. The water channel thickness is a nominal 0.065 in.

2.2 Reactor Control System

The reactor control instrumentation basically consists of two B-10-lined ionization chambers for reactor power operation above 0.1 watt and two B-10-lined neutron pulse chambers for low-level neutron measurements ($\approx$ one watt). The outputs of the ionization chambers are displayed on two neutron level recorders (one linear power and one logarithmic power) mounted in the reactor console. The outputs of the pulse chambers are connected to scalers and to a count rate meter.

Due to the nature of the tests conducted, the control system contains no automatic safety circuits such as power level scram or period scram circuits. The control circuitry is designed to be fail-safe and numerous interlocks, both electrical and mechanical, are provided. Thus, the control rods are automatically scrammed on loss of control system and/or interruption of other interlocks. A manual scram, power failure, loss of control rod air pressure, or loss of contact between any rod and its drive unit will cause an automatic fast rundown of all four control rod drives. A 15-kva gasoline-powered electric standby generator is provided to supply power for safe shutdown of the reactor in the event of commercial power failure.

FIG. 3 SPERT TYPE BD FUEL ASSEMBLY WITH A 22-FUEL-PLATE LOADING.

3. TRANSIENT INSTRUMENTATION

The transient instrumentation is utilized to measure variables associated with the study of reactor behavior under dynamic conditions. The variables measured include neutron level, fuel plate surface temperature, pressure, moderator flow, etc. This instrumentation is not used for operational purposes or as a safety control for any given test, ie, scram circuitry. However, the data obtained therefrom are extrapolated to establish specific limits for the overall test series.

3.1 Reactor Power Measurements

The measurement of reactor power is based on leakage neutron-flux measurements. The neutron-measurement system, shown in Figure 4, consists of a B-10-lined ion chamber, milli-micro ammeter, current driver amplifier, and recording oscillograph, with appropriate cabling and high-voltage supplies. For the Spert II power excursion tests reported here there were five linear and two logarithmic power channels.

FIG. 4 BLOCK DIAGRAM OF A TYPICAL NEUTRON POWER MEASUREMENT SYSTEM.

The ion chambers were either General Electric-type (G-9) chambers or Spert-type (SBCC) chambers [13]. Both types of ion chambers were B-10-lined and filled with N_2 to one atmosphere of pressure, and were uncompensated for gamma radiation. The chambers were positioned at various distances from the core to provide transient power level measurements over a power range of 10 decades. Periodic ion-chamber calibrations were made at the MTR to determine the electric current output from the chambers as a function of

thermal neutron flux level. The electronic circuitry associated with each ion chamber was calibrated prior to the tests and weekly thereafter, and the current output of the ionization chamber as a function of the neutron power level of the reactor was established by means of gold-foil activation measurements at the beginning and end of the test series. The dynamic response capabilities of the system are sufficient to follow a 5-msec period to within 1%.

3.2 Fuel-Plate Surface-Temperature Measurements

During the course of the initial series of power excursion tests, approximately 60 chromel-aluminum-alumel thermocouple junctions were attached to fuel plates in order to determine transient fuel plate surface temperatures. The thermocouples were distributed throughout the core in order to measure not only the hot spot temperatures but also to determine the temperature distribution throughout the core.

A block diagram of a typical temperature measurement system is shown in Figure 5. The active thermocouple junction is a 5-mil alumel wire and a 5-mil chromel wire, each spot-welded to the aluminum fuel plate surface. The absolute temperature at any junction cannot be established to better than $\pm 4^{\circ}\text{F}$ as supplied by the manufacturer's calibrations. Measurements also were made of the bulk-water temperatures during power excursions. These bulk-water temperature measurements established the time of expulsion of hot water from the assembly and provide a correlation of moderator temperature with transient flow. They also provide a measure of the temperature rise of the moderator caused by gamma and neutron heating. These measurements utilized ungrounded, chromel-alumel thermocouple junctions that were positioned in the moderator.

FIG. 5 BLOCK DIAGRAM OF A TYPICAL TEMPERATURE MEASUREMENT SYSTEM.

3.3 Pressure Measurements

Transient pressure measurements were made using unbonded strain-gage-type pressure transducers located at various positions in and around the core. The transducers were installed in the core so as to make free field measurement of pressure. A block diagram of a typical pressure measurement system is shown in Figure 6.

FIG. 6 BLOCK DIAGRAM OF A TYPICAL PRESSURE MEASUREMENT SYSTEM.

Environmental effects encountered during the course of the test series required the removal of some types of transducers or modifications in the transducer mounts. The major environmental problems in making transient and pressure measurements were those associated with radiation and temperature sensitivity. In particular, these factors significantly affected the data obtained from several pressure transducers located in the center of the core. The most reliable pressure data were obtained from transducers located near the outer edge of the core, and it is these data which are included in this report.

The main limitations on the accuracy of the pressure measurements presented in this report are those associated with radiation sensitivity. Radiation sensitivity tests indicate errors of no more than ± 1 psig in data taken at time of maximum power and less error at other times when the radiation field is less intense.

3.4 Miscellaneous Measurements

Attempts were made to measure the transient flow of water expelled from a fuel assembly and to measure the transient steam in a fuel assembly can. These measurements were developmental in nature and the results obtained are of qualitative value only.

III. STATIC CHARACTERISTICS

The Spert II BD-22/24 expanded D₂O core consists of twenty-four 3-in.-square x 24-in.-long highly enriched, plate-type fuel assemblies spaced on 6-in. centers as shown in plan form in Figure 7. This type of loading provides an expanded configuration that is typical of several heavy-water research reactor designs.

A detailed discussion of the experiments performed to determine the static characteristics of this core is given in IDO-16750[14]. A summary

FIG. 7 OPERATIONAL CORE LOADING CONFIGURATION OF THE BD-22/24 EXPANDED D₂O CORE.

of important core properties is presented in Table I. The operational fuel loading of 3.7 kg U-235 for the core provides a cold-clean excess reactivity of approximately \$8. Because of the long prompt neutron life-time for this core, ($\approx 750 \mu\text{sec}$), at least this amount of excess reactivity must be available to perform the required short-period tests at various system conditions. Isothermal moderator heating provided a measure of the density coefficients of reactivity for uniform changes in density of the moderator-coolant throughout the core and reflector region. Since the bulk of the density change occurs within the fuel assemblies during a power excursion (inter-element heating is negligible for the majority of the tests performed), the localized reactivity coefficient within the fuel assemblies is of more importance for self-shutdown analysis of power excursions. The local void coefficient was determined to be negative within all the fuel assemblies in the core, with an average value of $-0.02\phi/\text{cm}^3$. The thermal neutron flux is relatively uniform throughout the core and reflector; however, because of a flux depression within the fuel assemblies, hot spots could occur near the assembly edges during power excursions.

The reduced prompt neutron lifetime, ℓ/β_{eff} , was determined by analysis of 15 of the step-transient tests performed in this initial test series [15]. The step-transient data provided a value of

$$\ell/\beta_{\text{eff}} = 0.10 \pm 0.01 \text{ sec}$$

The value of ℓ/β_{eff} determined by analysis of the statistic of variance-to-mean ratio of the subcritical neutron population was $\approx 10\%$ higher than the value obtained from the transient tests. The data obtained during the subcritical experiment were not sufficient to permit inclusion of the delayed neutron effects in its analysis and neglect of their effect caused a systematic error in the analysis [16,17,18]. Consequently, the transient data provide a better estimate of the value of the reduced prompt neutron lifetime for this core.

TABLE I

STATIC CHARACTERISTICS OF THE SPERT II BD-22/24 EXPANDED D₂O CORE

Fuel Assembly Type	Spert BD
Number of Plates Per Fuel Assembly	22
Number of Fuel Assemblies	24
Core Configuration	"Expanded" (see Figure 7)
Mass of U-235	
Initial critical loading	2.8 kg
Operational loading	3.7 kg
Average Thermal Flux Per Watt	$\approx 6 \times 10^6$ nv/watt
Experimental Maximum-to-Average Core Flux Ratio	1.5
Isothermal Temperature Coefficient of Reactivity at a Constant System Pressure of 260 psig	-3.3¢/°C @ 20°C -4.4¢/°C @ 50°C -6.2¢/°C @ 100°C -8.0¢/°C @ 150°C
Temperature Defect (@ 260 psig)	
22°C to 96°C	\$3.5
22°C to 152°C	\$7.4
Available Excess Reactivity	
22°C, atmospheric pressure	\$8.10
96°C, 260 psig	\$4.65
152°C, 260 psig	\$0.60
Average Uniform Density Coefficient of Reactivity (per percent decrease in moderator density)	-94¢/%
Average Local Density Coefficient of Reactivity (per percent decrease of moderator density within the fuel assemblies)	-7¢/%
Average Void Coefficient of Reactivity Within the Fuel Assemblies	-0.02¢/cm ³
Maximum Measured Local Void Coefficient of Reactivity Within the Fuel Assemblies	-0.03¢/cm ³
Reduced Prompt Neutron Lifetime, ℓ/β_{eff}	0.10 $\pm$ 0.01 sec

IV. DYNAMIC BEHAVIOR OF THE BD-22/24 EXPANDED D₂O CORE

1. INTRODUCTION

The tests described in this section were initiated with the reactor system at ambient temperature ($\approx 22^\circ\text{C}$), at atmospheric pressure, with no forced coolant flow through the core, and a water head above the core of approximately 50 in. Each transient test was initiated by essentially stepwise increases in reactivity, with the reactor at an initial power level of about five watts. The range of initial asymptotic periods investigated was from 10.2 sec to 50 msec, which corresponds to reactivity insertions from approximately 40 cents to 3 dollars. The test with an initial period of 50 msec was the shortest-period test that could be performed without obtaining significant mechanical fuel plate damage. The first evidence of core damage in the form of permanent deformation of fuel plates was observed for tests with initial periods of approximately 70 msec.

During the test series transient measurements were made of reactor power, fuel plate surface temperature, and pressure. In addition, transient flow and strain measurements were made in a limited number of tests as a part of a continuing program of transducer development.

The summary plots of the data and derived quantities appear below in detail. The discussion is presented on a basis of reciprocal period and/or reactivity insertion in the interest of clarity in understanding the effects or implications of the behavior of the reactor. The details of individual tests and other supplemental information are presented in the appendices.

2. SELF-LIMITING BEHAVIOR

In the analysis of power excursions, the instantaneous power level and its time derivative are both important quantities. The exponential rate of rise of the power for a nuclear system is determined by the inhour equation,

$$\rho = \frac{\beta_{\text{eff}} \alpha_0}{\beta_{\text{eff}} K_{\text{eff}}} + \sum_i \frac{\beta_i / \beta_{\text{eff}}}{1 + \lambda_i / \alpha_0}$$

where the symbols have their usual meaning. When heavy water is used, the (γ, n) reaction in deuterium gives rise to additional neutrons in the system.

These are treated as delayed neutron groups and may be introduced into the inhour equation as an addition to those associated with the fissile material in the core. The effect of the delayed photoneutron groups is small and not significant for the larger reactivity additions which are normally of concern in reactor safety evaluations. A more significant change obtained in the inhour equation when heavy water is used instead of light water is the relatively large value for the reduced neutron lifetime, ℓ/β_{eff} . This change in the magnitude of ℓ/β_{eff} requires that a heavy-water reactor have a much larger reactivity addition than a light-water reactor in order to yield the same initial period. The relationship between the reactivity in dollars and the resulting asymptotic reactor period, τ_0 , for the BD-22/24 expanded D₂O core, using a value of 0.10 sec for ℓ/β_{eff} and including the additional delayed photoneutron groups is shown in Figure 8.

FIG. 8 REACTIVITY REQUIRED FOR A GIVEN PERIOD WITH THE BD-22/24 EXPANDED D₂O CORE.

The maximum reactor power, $\phi(t_m)$, achieved during the initial burst of a self-limiting power excursion is a function of the initial period or reactivity insertion as well as the nature and magnitude of the intrinsic mechanisms operating to quench the power rise. The power at time of peak, $\phi(t_m)$, as a

function of the reciprocal of the initial asymptotic period, α_0 , is shown in Figure 9. The power data are reasonably well represented on the logarithmic plot by a line of slope 1 for $\alpha_0 < 1.4 \text{ sec}^{-1}$; in the range $1.4 \text{ sec}^{-1} < \alpha_0 \lesssim 4 \text{ sec}^{-1}$, the power data are best represented by a line of slope 2; and for $\alpha_0 > 4 \text{ sec}^{-1}$, the data are seen to fall below the line with slope 2. It has been shown analytically [19] that if a single shutdown mechanism, which is linearly dependent on energy, were responsible for the quenching of the power excursion, the peak power should have an α_0^2 dependence in the region above prompt critical. This change in the behavior of the power data to a less than α_0^2 dependence for $\alpha_0 > 4$

FIG. 9 MAXIMUM REACTOR POWER VS RECIPROCAL PERIOD.

sec⁻¹ is apparently reflecting a change in shutdown mechanisms. In particular, as will be seen more clearly later, this behavior would appear to be due to a change from non-boiling shutdown to the case where steam formation is an important contributor to the quenching of the power burst. Although the magnitude of the power, $\phi(t_m)$, is clearly dependent on the shutdown mechanisms, it is obvious that relative changes in the importance of the mechanisms do not grossly change the general features of $\phi(t_m)$ as a function of α_0 .

The feature of the power behavior that is more sensitive to changes in the processes available to shut down the reactor than $\phi(t_m)$ is the time history of the power or burst behavior. Figure 10 is a plot of reactor power vs the dimensionless parameter, time in reactor periods, t/τ_0 , for five power excursions with periods of 4.1 sec, and 687, 215, 142, and 62 msec. It is clear from this figure that significant differences are observed in the reactor power behavior even though $\phi(t_m)$ as a function of α_0 did not reveal them with this degree of clarity. In particular, the data indicate that as the initial period becomes shorter, the power burst tends to become sharper and a large post-peak power dip is observed. Also, there is a trend for the power after the burst to change from one which is monotonically increasing toward an equilibrium level to a damped oscillatory power behavior, then to a steady oscillatory power behavior, and finally returning to a damped oscillatory behavior as the period is decreased. All of these changes in behavior can be linked to the temperature behavior and the presence of various types of boiling which will be discussed later.

The change in the power burst shape as the shutdown processes change also may be observed in a time history of the power burst up to the time of maximum power. Figure 11 shows the power history for a 4.1-sec, a 320-msec, and a 62-msec-period power excursion. Each of these is plotted for convenience of comparison as normalized power vs the dimensionless parameter, time in reactor periods. As the initial period becomes shorter, the power burst clearly changes shape from a broad burst to a much sharper one. The burst parameter, $\frac{\phi(t_m) \tau_0}{E(t_m)}$, where $E(t_m)$ is the energy release to time of maximum power, provides an indication of this change in the burst shape. This burst shape parameter is shown in Figure 12 as a function of α_0 . For the longer-period power excursions, the burst shape is nearly constant until near prompt critical where temperature information indicates boiling is starting at or before $\phi(t_m)$. For power excursions with reactivity insertions larger than \$1, corresponding

FIG. 10 REACTOR POWER BEHAVIOR AS A FUNCTION OF TIME IN REACTOR PERIODS.

FIG. 11 POWER BURST SHAPE CHANGES AS A FUNCTION OF TIME IN REACTOR PERIODS.

FIG. 12 POWER BURST SHAPE PARAMETER VS RECIPROCAL PERIOD.

to $\alpha_0 > 1.4 \text{ sec}^{-1}$, the burst becomes increasingly sharper, as indicated by the larger-burst-shape parameter value. The burst shape is not changing as rapidly for the larger values of α_0 , which indicates that the burst has nearly reached its maximum sharpness and that steam formation may now be the predominant mechanism available for shutdown of these subcooled power excursions for periods of 50 msec or less in this core.

Although the maximum power achieved in a power excursion and the time behavior of the power are useful measures of the inherent characteristics of reactor self-shutdown, the energy release is perhaps a more important variable

in evaluating the dynamic behavior of a reactor. The energy release is important for two reasons. First, the energy release is closely associated with self-shutdown characteristics of the reactor through the reactivity coefficients and the mode in which the energy released effects shutdown in the core, ie, temperature rise of water or metal, steam formation, radiolytic gas, etc. Secondly, the energy release is the variable that is normally associated with the severity of an accident.

The energy release to time of maximum power, $E(t_m)$, is an important variable in determining the nature of the shutdown processes since $E(t_m)$ is the maximum amount of energy available to provide the temperature, void, or other reactivity effect required to turn the power burst. $E(t_m)$ data and the reactivity compensation at time of maximum power, $\rho_c(t_m)$, (both of which are calculated from the power data) are shown in Figure 13 as a function of reciprocal period. For the smaller initial reciprocal periods, $\alpha_0 \lesssim 2 \text{ sec}^{-1}$, the energy and reactivity compensation are nearly proportional. For reciprocal periods, $\alpha_0 \gtrsim 2 \text{ sec}^{-1}$, $E(t_m)$ is increasing with α_0 because more reactivity compensation is required to quench the power excursion. The energy release, $E(t_m)$, as a function of α_0 does not, however, increase as fast as the reactivity compensation required to quench the power excursion. If the proportionality between energy and reactivity compensation that exists for $\alpha_0 \lesssim 2$ were to remain constant, the energy release would be represented by the dashed line shown in Figure 13. The change in the amount of reactivity compensation obtained per unit energy release can be clearly seen in Figure 14, which shows the reactivity compensation, $\rho_c(t_m)$, as a function of the energy, $E(t_m)$. For power excursions with $\alpha_0 > 5 \text{ sec}^{-1}$, which have energy releases to time of peak power of less than 10 Mw-sec and require reactivity compensations of less than 50 cents, the data indicate that the energy coefficient of reactivity, C_E , is approximately -4¢/Mw-sec . When the initial reciprocal period is greater than 5 sec^{-1} and $E(t_m)$ exceeds 10 to 11 Mw-sec, C_E increases in magnitude. For example, a 5-Mw-sec increment in energy between 5 and 10 Mw-sec will provide approximately 20 cents of reactivity compensation while the same energy increment between 15 and 20 Mw-sec will give approximately one dollar in reactivity compensation.

The fact that it requires less energy per unit reactivity compensation for the shorter-period tests where $E(t_m)$ exceeds $\approx 10 \text{ Mw-sec}$ is indicative of an apparent change in the nature of the shutdown mechanisms. The data pre-

FIG. 13 ENERGY TO TIME OF MAXIMUM POWER AND REACTIVITY COMPENSATION AT TIME OF MAXIMUM POWER VS RECIPROCAL PERIOD.

sented in Figures 13 and 14 would indicate that on the basis of energy release, the reactor would safely withstand much-shorter-period power excursions with only moderate increases in the magnitude of the energy release.

In analyzing the severity of an accident, the usual procedure is to initially postulate a situation that will result in a given reactivity insertion to the reactor and then predict the results of the accident. Because this postulation of an accident involves the reactivity addition as the independent parameter, it is

FIG. 14 REACTIVITY COMPENSATION AT TIME OF MAXIMUM POWER VS ENERGY RELEASE TO TIME OF MAXIMUM POWER.

of interest to determine the dependence of $E(t_m)$ as a function of the initial reactivity addition, ρ_0 , to the system. Figure 15 illustrates $E(t_m)$ as a function of ρ_0 for the BD-22/24 expanded D_2O core and shows that although $E(t_m)$ nearly doubles between initial reactivity additions of one and two dollars, further reactivity additions beyond two dollars will not result in additional energy release at this same rate. Extrapolation of the energy data indicates that a much larger reactivity addition to the system would result in only slight increases in energy release, $E(t_m)$. These data then would indicate, that on the basis of energy release, an accident caused by larger reactivity additions to the reactor would not necessarily be more severe than those tests which have been completed with this core.

The variables discussed in the previous paragraphs, such as $E(t_m)$ or $\phi(t_m)$, have been determined from extensive measurements and are average

FIG. 15 ENERGY RELEASE TO TIME OF MAXIMUM POWER VS REACTIVITY INSERTION.

quantities that are typical of the reactor as a whole. More often, however, the behavior of point locations or local areas in the core are of more importance in evaluating both what is taking place in the core and the safety of the system. The intensive or local measurements are also more sensitive to changes in behavior than the extensive or average measurements and consequently will provide, in some cases, a better indicator of the safe limits of operation.

An important local measurement made during the course of the power excursion tests on the D_2O core is that of the fuel plate surface temperature, which provides an indication of the local power or energy generation. The magnitude and behavior of the temperature at the surface of the fuel plates also indicate when steam formation is taking place and the nature of the heat transfer process.

The temperatures of the fuel plate surfaces at $\phi(t_m)$ are of importance in analyzing the shutdown processes. Figure 16 shows the maximum measured surface temperature of the plates at time of maximum power, $\theta(t_m)$, as a function of the reciprocal of the initial asymptotic period. For α_0 less than 1.4 sec^{-1} ,

FIG. 16 MAXIMUM MEASURED FUEL PLATE SURFACE TEMPERATURE AT TIME OF MAXIMUM POWER VS RECIPROCAL PERIOD.

$\theta(t_m)$ does not exceed the saturation temperature and this precludes steam formation as a participant in supplying the necessary reactivity compensation to quench the power rise. For α_0 greater than 1.4 sec^{-1} , the fuel plate surface in the area of maximum power generation exceeds the saturation temperature and indicates that boiling and steam voids may be contributing to the shutdown.

The temperature, $\theta(t_m)$, between α_0 of 1.4 sec^{-1} and 10 sec^{-1} is observed to be increasing relatively slowly as a function of α_0 until for α_0 greater than 10 sec^{-1} , $\theta(t_m)$ becomes a much more rapidly increasing function of α_0 . This behavior is intimately associated with the type of boiling heat transfer that is taking place on the surface of the fuel plate. Between an α_0 of 1.4 sec^{-1} and 10 sec^{-1} , the heat flux, temperature, and initial period are such that nucleate boiling can be maintained. This form of energy transfer will allow large amounts of energy transfer from the fuel plate to the moderator-coolant with a relatively small temperature differential between the fuel plate and water. Consequently,

the fuel plate temperature remains relatively low, energy is transferred to the water rapidly, and large reactivity compensations are provided with smaller incremental energy releases than those which are required for non-boiling shutdown. This form of boiling heat transfer is most advantageous for the safe quenching of a power excursion as it provides large reactivity compensations with little additional energy release and temperature rise.

For α_0 greater than 10 sec^{-1} , however, the temperature data indicate that conditions are such that the type of boiling heat transfer can change from nucleate boiling to partial film boiling. In this case, a steam blanket would be formed which essentially insulates the fuel plate so that energy cannot be transferred rapidly to the moderator, resulting in a rapid increase in fuel plate temperature. The appearance of partial film blanketing would thus account for the observed rapid rise in $\theta(t_m)$ vs α_0 greater than 10 sec^{-1} . This transition in temperature behavior does not grossly affect the shutdown capability of the reactor as a whole since the film boiling is restricted to a small central portion of the core in the period range investigated. In the much larger portion of the core nucleate boiling heat transfer continues to take place, providing the major part of the boiling shutdown contribution.

The local appearance of partial film boiling with its associated temperature behavior, although not important for shutdown in these cases, does have important safety considerations since it may result in fuel plate meltdown. In terms of fuel plate meltdown considerations, it is not $\theta(t_m)$ that is important, but rather the maximum fuel plate surface temperature, $\theta(\max)$. In Figure 17 $\theta(\max)$ vs α_0 data for the expanded D_2O core are shown. The dependence on α_0 of these data is similar to that of $\theta(t_m)$ vs α_0 , except that in general, the temperatures are higher. In the range of $1 < \alpha_0 < 10 \text{ sec}^{-1}$, $\theta(\max)$ is not much greater than $\theta(t_m)$ because nucleate boiling predominates. For $\alpha_0 > 10 \text{ sec}^{-1}$, the introduction of partial film blanketing causes the hot spot temperatures to rise rapidly, resulting in $\theta(\max)$ exceeding $\theta(t_m)$ by more than 100°C for $\alpha_0 = 20 \text{ sec}^{-1}$. The data in Figure 17 indicate that power excursions with $\alpha_0 \approx 30 \text{ sec}^{-1}$ would result in fuel plate temperatures high enough to cause melting even though extrapolation of the energy data in Figure 13 would indicate only a small increase in the energy release. The data on $\theta_{\max}$ vs ρ_0 , shown in Figure 18, are of interest for accident analysis. The data in Figure 18 have the same general features as the $\theta(\max)$ vs α_0 data. This indicates that although the system did not reach meltdown for reactivity addition of about \$3, $\theta(\max)$ is a rapidly increasing function of ρ_0 above \$2.

FIG. 17 MAXIMUM MEASURED FUEL-PLATE SURFACE TEMPERATURE VS RECIPROCAL PERIOD.

FIG. 18 MAXIMUM MEASURED FUEL PLATE SURFACE TEMPERATURE VS REACTIVITY INSERTION.

Extrapolation of these data indicates that a reactivity addition of about \$3.5 would result in probable fuel melting.

Figures 19, 20, and 21 are examples of the time behavior of fuel plate surface temperatures for power excursions with initial periods of 496, 88, and 62 msec. Collectively, they exhibit the behavior that is observed as a function of the initial period, the temperatures achieved, and of the changing types of heat transfer occurring during the course of the transient. Figure 19 illustrates the fuel plate surface temperature measured near the area of maximum power generation in the core during a 496-msec-period power excursion. In this example, the temperature increases essentially exponentially up to and slightly beyond the saturation temperature; it then increases at a slower rate until, or slightly after, the time of maximum power. The deviation of the temperature rise from an exponential function and its constancy approximately 20°C above the saturation temperature, even though the power is increasing, is indicative of a heat transfer mechanism (in the constant temperature region)

FIG. 19 FUEL PLATE SURFACE TEMPERATURE AS A FUNCTION OF TIME DURING A 496-MSEC-PERIOD POWER EXCURSION.

FIG. 20 FUEL PLATE SURFACE TEMPERATURE AS A FUNCTION OF TIME DURING AN 88-MSEC-PERIOD POWER EXCURSION.

FIG. 21 FUEL PLATE SURFACE TEMPERATURES AT THREE SYMMETRIC CORE LOCATIONS AS A FUNCTION OF TIME DURING A 62-MSEC-PERIOD POWER EXCURSION.

that permits the transfer of large amounts of energy with only small temperature differentials. This type of behavior can be associated with nucleate boiling, as is well confirmed from steady-state heat transfer studies and also from both in-pile [20,21] and out-of-pile [22,23,24] transient studies.

As the initial period is decreased, the temperature and heat flux increase until nucleate boiling cannot be maintained, resulting in the onset of partial film blanketing, as observed in the temperature data shown in Figure 20. These data again show the exponential increase of temperature during the heat conduction phase and the break from the exponential after the temperature has exceeded the saturation temperature and nucleate boiling occurs. About the time of maximum power, the temperature exhibits a rapid increase, even though the power is near maximum or is decreasing. The rapid increase in temperature, even though the power is decreasing, is an indication of the reduction of energy transfer rate from the fuel plate to the water due to the onset of partial film boiling heat transfer. The onset of this form of heat transfer accounts for the rapid increase in $\theta_{\text{(max)}}$ as a function of α_0 even though the energy release is only increasing by small amounts.

The data shown in Figure 21 were obtained from thermocouples attached to the surface of fuel plates in three symmetrically located fuel assemblies. It would be expected that, since these thermocouples are at locations where the power density is the same, the outputs of the thermocouples should be nearly the same. It is observed that all of the thermocouples indicate the same temperatures during the well-behaved conduction heat transfer portion of the power excursion and that the initiation of nucleate boiling occurs at about the same time in each case. However, after boiling has been initiated, the temperature behavior observed at each location is indicative of the randomness of the boiling process. The time history of each thermocouple's output after boiling has been initiated is an individual process that is not only influenced by the magnitude of the temperature and heat flux, but also by the surface conditions, nucleation centers, etc, so that specific boiling and temperature behavior during boiling is stochastic in nature.

The data obtained from two of the thermocouples shown in Figure 21 were analyzed, using an IBM 650 program [25] to determine heat transfer rates and the energy partition between the fuel plate and water. These data were chosen because they exhibited the characteristics of three types of heat transfer, ie, conduction, nucleate boiling, and partial film boiling. In addition, the two thermocouple locations were in symmetrical positions in the core so that some information on reproducibility could be obtained. The local power, surface temperature, calculated center temperature, and calculated heat transfer rates at each thermocouple position are shown in Figures 22 and 23. Figures 24 and 25 illustrate the total energy generation, energy in the fuel plate, and energy in the water as functions of time for the same two thermocouples locations. In interpreting these data, it is noted that they are applicable only to a particular point in the core, and are only indirectly related to an average or extensive measurement such as the total reactor power. It must also be realized that these are only two examples for one set of conditions and will only offer an indication of what may be happening. A much more extensive study would be required to provide the complete information.

Figure 22 shows the time history of the power, temperature, and heat transfer rate for the fuel plate surface thermocouple in the E1-2 fuel assembly. It is apparent from this figure that the heat transfer rate begins to be a sharply increasing function and the temperature starts to deviate from an exponential rise

FIG. 22 POWER, FUEL PLATE SURFACE TEMPERATURE, CALCULATED CENTER TEMPERATURE, AND CALCULATED HEAT TRANSFER RATE FOR A POINT AT POSITION E1-2 DURING A 62-MSEC-PERIOD POWER EXCURSION.

at a time when the power is still on an exponential and well before maximum power. This type of behavior is indicative of the initiation of nucleate boiling as has been observed by Miller [20,21] in in-pile photographic studies of transient boiling. The onset of nucleate boiling occurs at a temperature of about 235°F, indicating that a saturation temperature overshoot or superheat of about 25°F is required to initiate boiling under the test conditions.

The initiation of nucleate boiling provides the rapid transfer of energy from the fuel to the water and tends to prevent the fuel from reaching much higher temperatures. Nucleate boiling transfers large amounts of energy to the water, resulting in the formation of steam bubbles and water heating and in the large reactivity effect required to quench the power burst. It would appear, however, from these data that nucleate boiling was not maintained throughout

FIG. 23 POWER, FUEL PLATE SURFACE TEMPERATURE, CALCULATED CENTER TEMPERATURE, AND CALCULATED HEAT TRANSFER RATE FOR A POINT AT POSITION N1-2 DURING A 62-MSEC-PERIOD POWER EXCURSION.

the portion of the transient in which the fuel plate surface was above saturation temperature. The data indicate that slightly before time of maximum power the surface temperature reached about 315°F, and partial film blanketing occurred. In this test, the temperature rise is halted because the power is starting to decrease and film boiling is only partial.

Figure 23 presents data similar to that shown in Figure 22 for the same type thermocouple in the N1-2 fuel assembly. The power generation should be nearly identical and hence it might be expected that the temperature behavior would be the same. By comparing Figures 22 and 23, it can be seen that the

FIG. 24 ENERGY PARTITION BETWEEN FUEL PLATE AND WATER FOR A 62-MSEC-PERIOD POWER EXCURSION (POSITION E1-2).

FIG. 25 ENERGY PARTITION BETWEEN FUEL PLATE AND WATER FOR A 62-MSEC-PERIOD POWER EXCURSION (POSITION N1-2).

general behavior is nearly the same; however, many of the details of the two are different, indicative of the statistical nature of boiling.

Figures 24 and 25 show the total energy generated and the calculated partition of this energy between the water and the fuel plate as a function of time for the two thermocouple locations discussed above. Examination of these figures indicate that during the conduction phase of heat transfer the majority of energy generated is retained in the fuel plate. Once nucleate boiling is initiated, a large fraction of the energy generated is transferred to the water with the energy in the plate remaining nearly constant. The initiation of partial film blanketing is noticeable in Figure 24, as evidenced by the rapid increase in the energy retained in the plate near the time of maximum power. The same general behavior is also noted in Figure 25, although to a less noticeable degree because the transition to film blanketing was apparently not as abrupt in this case.

The rapid transfer of energy into the water during a power excursion via boiling heat transfer and the generation of sizeable steam volumes in the core during short periods of time causes transient pressures to be generated during a power excursion. Figure 26 presents pressure data measured at the edge of the core at time of maximum power, $P(t_m)$, and the maximum pressure measured, $P(\max)$, both as functions of the reciprocal period. $P(t_m)$ does not reach magnitudes that might cause damage, being only about 10 psig for the 50-msec power excursion. For excursions with $\alpha_0 < 10 \text{ sec}^{-1}$, the maximum pressure, $P(\max)$, normally occurs slightly after the peak of the power burst and is associated with the maximum steam generation rate. For $\alpha_0 > 10 \text{ sec}^{-1}$, $P(\max)$ appears to occur not as a result of steam generation but rather because of the collapse of the steam voids, the return of the water to the core causing a sharp pressure pulse by a water-hammer effect. This change in the cause of $P(\max)$ is indicated in the data shown in Figure 26, where for $\alpha_0 > 10 \text{ sec}^{-1}$ $P(\max)$ increases much more rapidly as a function of α_0 than it did for $\alpha_0 < 10 \text{ sec}^{-1}$.

An example of the time history of the pressure during a 62-msec power excursion is shown in Figure 27. In this case, it is noted that the initial pressure pulse is created by steam formation and that the maximum in this pulse occurs slightly after time of maximum power. For power excursions with longer initial periods, where the steam volume would not be as large, this initial pulse will be the only significant pressure pulse observed. However, for this or shorter-period power excursions, enough water has been accelerated

FIG. 26 PRESSURE AT TIME OF MAXIMUM POWER AND MAXIMUM PRESSURE MEASURED AT THE EDGE OF THE CORE VS RECIPROCAL PERIOD.

from the core to make the collapse of the bulk of the voids, the rapid return of the water to the core, and the continuation of boiling result in more complex and severe pressure effects. The magnitude of the pressure that is associated with the water-hammer effect, as seen in the sharp pressure pulse, is not unduly large although the time rate of rise is quite sharp. The largest magnitude pressure pulse shown here increases from 0 to 25 psig in approximately 6 msec, or at a rate of nearly 4000 psi/sec. This time rate of change of pressure may, by causing significant pressure differentials throughout the core, result in significant core damage.

FIG. 27 PRESSURE AT THE EDGE OF THE CORE AS A FUNCTION OF TIME FOR A 62-MSEC-PERIOD POWER EXCURSION.

The permanent deformation of fuel plates has been observed for power excursions with initial periods less than 70 msec. The deformation was manifest in the bowing of the outer fuel plates in the central fuel assemblies outward toward the fuel box. A photograph of one such fuel assembly is shown in Figure 28. This deformation appears to be a consequence of the pressure and/or the thermal stresses caused by the temperature differentials associated with the onset of nucleate and film boiling. In some cases the fuel plates were bowed sufficiently so that the central portion of the outside plate was apparently in contact with, or in close proximity to, the can so that the water channel width was severely reduced. Limited local damage of this type necessitated the replacement of numerous fuel plates after each of the shorter-period tests to preclude the occurrence of hot spots and anomalous temperatures during subsequent tests.

FIG. 28 TOP-VIEW PHOTOGRAPH OF SPERT II FUEL ASSEMBLY SHOWING FUEL PLATE BOW-
ING,

V. SUMMARY AND CONCLUSIONS

The initial series of self-limiting power excursions with a D_2O -moderated core in Spert II has indicated that the basic mechanisms for shutdown are thermal in nature. The specific nature of the shutdown can be further sub-divided into either non-boiling shutdown such as fuel plate expansion and moderator heating, or boiling shutdown which is caused by steam void formation.

For reactivity additions of about \$1 or less, corresponding to initial periods of about 600 msec or longer, the necessary reactivity compensation required to turn the initial power burst is provided entirely by non-boiling mechanisms. This reactivity compensation is provided at the rate of about 4¢/Mw-sec of energy released during the power excursion for all non-boiling shutdown for excursions with periods of ≥ 600 msec. The power burst shape in this region is broad and symmetrical as would be predicted by a feedback model in which the reactivity feedback is linearly dependent on the energy release. Power excursions with initial periods of from about 600 to about 100 msec, corresponding to reactivity additions of from about \$1 to \$2 apparently were quenched by a combination of non-boiling and boiling shutdown mechanisms. This region is a transition region between that where non-boiling shutdown predominates and that where boiling shutdown predominates. The indication that steam formation is contributing to the self-limitation of the power increase is indicated in several ways. First, it is observed that the fuel plate surface temperatures exceed the saturation temperature at time of maximum power with the presence of boiling indicated by the time behavior of the temperature; second, the power burst shape becomes sharper when steam contributes to the shutdown; and finally, larger increments of reactivity compensation are obtained with smaller increments of energy release, as would be expected if nucleate boiling were taking place and steam voids were being formed.

The self-limitation of power excursions with initial periods less than about 100 msec, corresponding to reactivity additions greater than about \$2, apparently occurs predominantly by a boiling shutdown mechanism. This is indicated by the fact that the power burst shape is approaching a nearly constant form and that almost all of the fuel plate surfaces are above saturation temperature before time of maximum power. The reactor power and energy release at peak power both reflect the influence of the large contribution of

boiling to the shutdown, which provides large reactivity compensation with only nominal additional energy release. This is a direct consequence of the existence of nucleate boiling in a large portion of the core. The data indicate, however, that although a large or major portion of the core is experiencing nucleate boiling, the central or maximum power generation area is beginning to experience partial film boiling. The onset of this type of boiling-heat transfer in any portion of the core will rapidly negate any inferences of safety that might be derived from extrapolation of the energy or power data. The onset of partial film boiling, with its steam blanketing of fuel plate surfaces, will cause rapid increases in temperature with only small increases in the total energy release and could result in rapid localized meltdown. The sizeable amounts of steam generated to quench the power excursion also result in an added danger to the system because of possible transient pressure generation. In these tests, although the volume of steam voids may approximate 6 to 8 liters, nearly 15 to 25% of the moderator volume within the assemblies, the rate of generation is relatively slow and the pressures observed during the time of steam formation are not large. The more damaging pressures occur after time of maximum power when the steam voids collapse, allowing water to return to the core very rapidly. This rapid return of water to the core results in pressures of about 30 psig with rise rates of about 4000 psig/sec.

Data obtained during these tests indicate that, for power excursions with initial periods less than about 500 msec, the shutdown processes are occurring primarily within the fuel assembly cans. No indication of any appreciable expulsion of water or steam from the can is observed until at or after the time of maximum power. There is also no indication of heating at the exterior surface of the can, or in the water channels between, until after time of maximum power.

The onset of limited local core damage such as fuel plate deformation for this core has been determined to occur at initial periods of about 70 msec (or reactivity additions of about \$2.4) at these given initial conditions. Extrapolation of the data indicates that localized melting of fuel plates might be expected for step insertions of about \$3.5 reactivity, corresponding to initial periods of about 30 msec.

Comparison of the results of these tests and the results of similar tests of light-water-moderated cores shows that the heavy-water-moderated core experiences limited fuel plate damage for tests with periods five to ten times

longer than the corresponding cases in light-water cores, and for reactivity additions approximately twice as large as for light-water reactors. This is a consequence of the differences in prompt neutron lifetime of the reactor and the magnitude of the shutdown coefficients. Although the self-shutdown mechanisms for the heavy-water-moderated core may be expected to be basically the same as for highly enriched, plate-type, light-water systems, a larger dependence on steam formation for shutdown of subcooled power excursions is observed for the D_2O core. While nucleate boiling is a highly efficient mechanism in terms of total energy release required for steam formation and for quenching of power excursions, the onset of film boiling can lead to dangerous burnout or meltdown conditions quickly. Extrapolation of the data to the shorter-period region indicates that while only a nominal increase in energy release would result, changes in heat transfer mechanisms may be expected to result in a higher probability of obtaining a partial meltdown of the core.

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APPENDIX A
SUMMARY OF EXPERIMENTAL DATA

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SUMMARY OF EXPERIMENTAL DATA

A summary of experimental data obtained during the self-limiting transient tests on the Spert Expanded Core is given in Table A-1. The symbols used in Table A-1 have the following meaning:

<u>Symbol</u>	<u>Unit</u>	
No.	- - -	Oscillograph record or transient number
τ_0	sec	Initial asymptotic reactor period
α_0	sec ⁻¹	Reciprocal of the initial asymptotic reactor period
$\phi(t_m)$	Mw	Maximum reactor power (initial power burst)
$E(t_m)$	Mw-sec	Energy release to time of maximum power
ρ_0	\$	Reactivity insertion
$\rho_c(t_m)$	\$	Amount of reactivity compensation at time of maximum power
$\theta(t_m)$	°C	Maximum recorded fuel plate surface temperature at time of maximum power
$\theta(\max)$	°C	Maximum recorded fuel plate surface temperature
$P(t_m)$	psig	Pressure at side of core at time of maximum power
$P(\max)$	psig	Maximum pressure at side of core

TABLE A-1

SUMMARY OF EXPERIMENTAL DATA

No.	τ_c (sec)	α_o (sec ⁻¹)	ϕ_{tm} (Mw)	E_{tm} (Mw-sec)	ρ_o (\$)	$\rho_c(tm)$ (\$)	θ_{tm} (°C)	θ_{max} (°C)	P_{tm} (psig)	P_{max} (psig)
00702	10.2	0.098	0.341	—	—	—	69	69	—	—
00671	5.87	0.17	0.497	7.8	0.51	0.32	65	79	—	—
00112	4.1	0.24	0.790	7.8	0.58	0.27	64	89	—	—
00240	2.3	0.43	1.40	7.0	0.70	0.28	76	83	—	—
00292	1.1	0.91	2.72	7.4	0.86	0.27	89	109	—	—
00782	0.703	1.42	5.02	7.7	0.97	0.28	107	113	—	—
00313	0.687	1.46	5.44	7.5	—	—	101	117	—	—
00787	0.496	2.02	9.88	9.1	1.07	0.32	105	119	—	—
00392	0.364	2.75	13.6	10.1	1.17	0.36	123	123	—	—
01130	0.320	3.1	19.1	10.7	1.22	0.38	120	123	0.7	1.2
00459	0.237	4.22	33.9	12.7	1.35	0.46	131	136	—	1.6
00924	0.215	4.65	36.2	12.2	1.40	0.51	135	151	1.5	2.5
00910	0.208	4.8	36.6	11.9	1.42	0.51	130	147	1.0	1.9
01125	0.174	5.8	51.4	13.4	1.53	0.60	128	156	1.5	2.4
01104	0.143	7.00	70.2	14.7	1.66	0.80	137	158	2.0	3.0
00465	0.142	7.04	71.3	14.2	1.70	0.79	147	152	—	—
00931	0.114	8.8	96.9	14.9	1.85	0.92	149	160	3.0	5.5
00575	0.088	11.4	137	16.5	2.12	1.13	172	189	—	—
01025	0.071	14.0	173	16.6	2.39	1.41	206	231	5.0	17.0
00582	0.062	16.1	228	18.5	2.60	1.70	196	243	5.5	23.5
00596	0.050	20.0	306	19.7	2.95	1.97	236	337	9.5	33.0

APPENDIX B
TEST DATA FIGURES

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TEST DATA FIGURES

In the figures (29 through 59), selected variables from this initial series of tests are displayed as functions of time. Data presentations of this nature illustrate the overall reactor response under the given test conditions. On these figures the time of maximum reactor power and the saturation temperature of the moderator-coolant have been indicated for reference. The principal means of identification of each test is the test record number shown on each figure.

The reactor period has been used to establish the order of presentation of data figures and all data figures are presented in order of decreasing period. Most of the test data figures cover a time interval commencing with an observable rise in the power level and terminating when the reactor scrammed. On the shorter-period tests, the rate of change of the measured quantities is extremely rapid near the time of maximum reactor power. In order to facilitate a detailed observation of the measured variables in this region, portions of the shorter-period transients have been expanded in time. These figures follow the corresponding data figure illustrating the complete test interval.

FIG. 29 TEST 00702, 10.2-SEC PERIOD.

FIG. 30 TEST 00671, 5.9-SEC PERIOD.

FIG. 31 TEST 00112, 4.1-SEC PERIOD.

FIG. 32 TEST 00240, 2.3-SEC PERIOD.

FIG. 33 TEST 00292, 1.1-SEC PERIOD.

FIG. 34 TEST 00782, 703-MSEC PERIOD.

FIG. 35 TEST 00313, 687-MSEC PERIOD.

FIG. 36 TEST 00787, 496-MSEC PERIOD.

FIG. 37 TEST 00392, 364-MSEC PERIOD.

FIG. 38 TEST 01130, 320-MSEC PERIOD.

FIG. 39 TEST 00459, 237-MSEC PERIOD.

FIG. 40 TEST 00459, 237-MSEC PERIOD (EXPANDED).

FIG. 41 TEST 00924, 215-MSEC PERIOD.

FIG. 42 TEST 00924, 215-MSEC PERIOD (EXPANDED).

FIG. 43 TEST 00910, 208-MSEC PERIOD.

FIG. 44 TEST 00910, 208-MSEC PERIOD (EXPANDED).

FIG. 45 TEST 01125, 174-MSEC PERIOD.

FIG. 46 TEST 01125, 174-MSEC PERIOD (EXPANDED).

FIG. 47 TEST 00465, 142-MSEC PERIOD.

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FIG. 48 TEST 01104, 143-MSEC PERIOD.

FIG. 49 TEST 01104, 143-MSEC PERIOD (EXPANDED).

FIG. 50 TEST 00931, 114-MSEC PERIOD.

FIG. 51 TEST 00931, 114-MSEC PERIOD (EXPANDED).

FIG. 52 TEST 00575, 88-MSEC PERIOD.

FIG. 53 TEST 00575, 88-MSEC PERIOD, (EXPANDED).

FIG. 54 TEST 01025, 71-MSEC PERIOD.

FIG. 55 TEST 01025, 71-MSEC PERIOD (EXPANDED).

FIG. 56 TEST 00582, 62-MSEC PERIOD.

FIG. 57 TEST 00582, 62-MSEC PERIOD (EXPANDED).

FIG. 58 TEST 00596, 50-MSEC PERIOD.

FIG. 59 TEST 00596, 50-MSEC PERIOD (EXPANDED).

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