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REACTOR POWER EXCURSION TESTS IN THE SPERT IV FACILITY

MASTER

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**PHILLIPS
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ATOMIC ENERGY DIVISION

**NATIONAL REACTOR TESTING STATION
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ABSTRACT

A program of self-limiting reactor power excursion tests was performed in the Spert IV facility with a highly enriched, plate-type, water-moderated core. The tests were initiated by the stepwise addition of excess reactivity with the system at ambient temperature and atmospheric pressure. Three series of tests were performed: (a) A fiducial series of tests was performed with an 18-foot hydrostatic head above the core with no forced coolant flow in the range of asymptotic periods from approximately 1 sec to 7 msec. No damage, other than minor fuel plate bowing, was observed due to any test in this series; however, extrapolation of measured fuel plate surface temperatures indicated that fuel plate melting could be expected for excursions with periods shorter than 6 msec. (b) A brief series of tests was performed in the short-period region ($\tau < 20$ msec) with no forced coolant flow in which the hydrostatic head was reduced to two feet in order to determine the effect of changes in that variable on reactor power excursions. The only observed change in the response of the reactor for the lower-head tests was a slight reduction in measured fuel plate surface temperatures. (c) A series of tests was performed with an 18-foot head and various forced coolant rates up to 12 ft/sec through the core with reactor periods in the range from approximately 500 msec to 10 msec. Results from these tests showed that the addition of forced flow causes higher peak powers and broader power bursts for equal reactivity insertions than were obtained during tests without flow. The effects were most pronounced in the long-period region and decreased as the period was shortened. Also, the addition of forced flow resulted in the attainment of higher equilibrium powers which, for most tests, were approached in an oscillatory manner. This report presents a summary of the results and the conclusions which were derived from these series of tests.

SUMMARY

In 1963 a series of self-limiting reactor power excursion tests initiated by stepwise insertions of reactivity was performed in the Spert IV facility with a highly enriched, water-moderated, plate-type core. The primary purpose of these tests was to establish the response of the core to step insertions as an aid in planning tests in which the reactor is allowed to exceed the threshold of instability. Tests were performed with initial asymptotic periods in the range from approximately 1 sec to 7 msec. The tests were initiated with the reactor either critical or slightly subcritical at power levels of a few watts or less, with the system at atmospheric pressure and ambient temperature, and with forced flow rates of 0 to 12 ft/sec through the core. For most of the tests, there was an 18-foot hydrostatic head above the core, although a few short-period tests with no forced coolant circulation were performed with a 2-foot head to investigate the effect of that variable. No damage to the core other than minor fuel plate bowing occurred as a result of the tests, except in one case where a hole was melted in a fuel plate during a short-period test with forced flow.

A fiducial series of tests was performed with no forced coolant circulation and with an 18-foot hydrostatic head above the core. In the shortest period test which was performed in this series, a 7-msec-period test, the peak power reached was 875 MW, the maximum fuel plate surface temperature recorded was 338°C, the peak transient pressure was 21 psig, and the energy released during the burst was 12 MW-sec. During this series, it was determined that for reactor periods longer than 15 msec no fuel damage occurred. For 15-msec-period tests minor fuel plate bowing began to be observed. Extrapolations of available temperature data indicated that fuel plate clad melting could be expected for tests with periods shorter than 6 msec.

A few tests were performed with no forced coolant circulation and only a 2-foot head above the core to determine the effect of the hydrostatic head on the transient response of the core. These tests were limited to the short-period region ($\tau < 20$ msec) where boiling is one of the significant reactivity-compensating mechanisms for highly enriched, plate-type cores, and where it was expected that the change in static pressure and resultant change in moderator boiling temperature would have the most significant effect. The test results showed, however, that no significant effect on the reactor response was produced by the change in the water head, although the fuel plate surface temperatures reached in the 2-foot-head tests were slightly lower than for the 18-foot-head

tests. The shortest period test which was performed with a 2-foot head was an 8.8-msec-period test in which the power reached 570 MW, the maximum fuel plate surface temperature recorded was 184°C, the peak transient pressure was 18 psig, and the energy released during the burst was approximately 7 MW-sec.

In order to investigate the effect of forced coolant circulation on the response of the Spert IV core, a series of tests was performed with reactor periods in the range from approximately 500 msec to 10 msec with forced coolant velocities of 1.2, 2.4, 6, and 12 ft/sec upward through the core. All of these tests were performed with a hydrostatic head of 18 feet above the core.

For the tests in the long-period region ($\tau \approx 500$ msec), the addition of forced coolant circulation resulted in an elimination of an initial power peak for all flow rates investigated. In all of the tests with periods shorter than 100 msec, the reactor power exhibited an initial peak. It was observed for all of these tests that the higher flow rates resulted in higher peak powers and broadened burst shapes. This effect was most pronounced in the longer period tests and decreased as the period was shortened. As the burst shapes tended to be broadened for the higher-flow-rate tests, the energy released during those tests also was increased.

The post-peak power behavior of tests with flow exhibited an oscillatory nature. For the longer period tests, the oscillations took the form of random fluctuations of a few percent about the mean power level, while for the short-period tests, large-amplitude, damped oscillations with a definite frequency were observed. It was observed that both the magnitude and frequency of the oscillations increased as the flow rate was increased. Although many of the tests were terminated before true equilibrium power levels had been established, it was determined that the equilibrium level toward which the power tended varied approximately proportionally with the flow rate.

In general, the addition or increase in the rate of forced coolant circulation decreased the fuel plate surface temperatures which were reached during the tests. This effect was most pronounced in the long-period region and decreased for shorter periods.

There was no detectable effect on the transient pressures generated during the tests as a result of the addition of flow.

Since the outside moderator-coolant channels in the standard Spert IV fuel assembly have a thickness which is one-half that of the normal channels, these channels have lower flow rates than the standard channels, and departure from nucleate boiling occurs at substantially lower powers in these channels causing

abnormally high temperatures for adjacent fuel plates. This effect is not particularly significant during power excursions since relatively little energy is released during the excursion, but can become extremely important for tests with long run-out times or for steady high-power operation and may limit the stability tests which can be conducted with this core under forced flow conditions.

During a 10-msec-period test with 12 ft/sec coolant flow, a hole was melted in a fuel plate which was adjacent to one of these narrow coolant channels. In this test, the initial peak of 480 MW was followed by successive peaks of approximately 140, 100, 70, and 50 MW and subsequent small oscillations about a steady power of 35 MW for a few seconds. It was determined by performance of several repeated tests that the occurrence of melting in this channel was not a typical result, and it was concluded that in the initial test the channel must have been blocked or partially blocked either prior to the test or as a result of the initial power burst. The melting did, however, point out the sensitivity of fuel plate surface temperatures to the decreased flow in the narrow channels and demonstrated the consequences which can be expected due to channel blockage during tests with significant energy releases.

A value of 7.94 ± 0.26 msec for the reduced prompt neutron generation time, $\Lambda/\beta_{\text{eff}}$, was obtained from analysis of excursion tests which were initiated by reactivity insertions greater than one dollar. This value is in good agreement with the value of 8.07 ± 0.17 msec which was obtained by analysis of the inherent fluctuations in the neutron population at various subcritical reactivities [1].

REACTOR POWER EXCURSION TESTS IN THE SPERT IV FACILITY

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I. INTRODUCTION

A primary objective of the Spert Program of reactor safety research and development is that of increasing the understanding of the dynamic behavior of nuclear reactor systems by the performance and analysis of reactor kinetic experiments. In general, the types of experiments which have been conducted in the Spert reactor kinetics testing programs may be classed as (a) static experiments in which such core characteristics as temperature coefficients, neutron flux distributions, control rod worths, etc, are determined; (b) power excursion tests in which excess reactivity is added to the system, either in stepwise fashion or as a function of time, in order to produce a power burst; and (c) stability tests which involve either self-induced or externally-excited oscillations. Tests of these three types have been performed in the various Spert reactors with cores [2-6] of different mechanical designs, delayed neutron fractions, prompt neutron lifetimes, reactivity coefficients, etc, under various operating conditions in order to determine the relative importance of the many factors which may influence the kinetic behavior of a reactor system.

As part of the overall Spert program, the Spert IV facility was designed to permit the experimental investigation of the dynamic behavior of a wide variety of reactor core designs under conditions similar to those found in large-pool-type facilities. Basically, the facility is composed of two 20-foot-diameter x 25-foot-deep reactor tanks with provision for forced coolant circulation at rates up to 5000 gpm and steady-state heat removal capacity of 1 MW. Spert IV, as is the case with all Spert reactors, is controlled remotely from a control center which is approximately one-half mile from the reactor building.

Early in the history of the Spert Project, a series of tests [7] was conducted in Spert I in which severe oscillations of the reactor power were observed. The limitations of the Spert I facility, however, did not permit adequate control of test conditions necessary for a comprehensive reactor stability study. Therefore, one of the primary reasons for the design and construction of Spert IV was to permit a continuation of these reactor stability studies under more closely controlled environmental conditions. A highly enriched, plate-type, light-water-moderated core was chosen for this program in order to have a core design which is similar to that used in the Spert I tests and because of the extensive use which is made of this type of core in research and test facilities.

Initial criticality was achieved in July 1962, with the first core to be used in the Spert IV stability studies designated the Spert IV D-12/25 core following which, an extensive series of static tests and core parameter measurements was conducted [8]. Following these tests and prior to the initiation of the stability test program, a series of power excursion tests was performed to establish the response characteristics of the D-12/25 core to stepwise insertions of reactivity. In these tests, the reactor power was allowed to rise on asymptotic periods in the range from approximately 1 sec to 7 msec under system conditions of ambient initial temperature, various forced coolant flow rates in the range from 0 to 12 ft/sec, and with a hydrostatic head above the core of approximately 18 feet. In addition, a few tests were performed with a 2-foot water head above the core. In all of the tests which were performed, the inherent reactivity compensating mechanisms were allowed to limit the initial power burst, and the reactor was not scrammed until approximate equilibrium conditions had been reached. This report summarizes the results which were obtained from these power excursion tests. Also presented in this report are brief descriptions of the Spert IV facility and of the D-12/25 core.

II. DESCRIPTION OF SPERT IV FACILITY AND INITIAL CORE

1. BRIEF DESCRIPTION OF SPERT IV FACILITY

Spert IV is a large-pool reactor facility designed to permit the experimental investigation of the kinetic behavior of a variety of reactor types. The facility is divided into two areas of operation, the control area and the reactor building area, with all nuclear operation of the reactor being accomplished, remotely, from a control room in the Spert control center located about 1000 yards from the reactor building. The reactor building houses the reactor pools, a forced coolant system, process and experimental instrumentation, a water treatment system, and other auxiliary equipment necessary to the operation of the facility. Operation of the major plant equipment may be performed from either the control room or the reactor building.

The pool portion of the facility (Figure 1) is composed of two 20-foot-diameter by 25-foot-deep tanks which are connected at the top by a 6- by 6-foot removable gate. The tanks are designed for 25 feet of hydrostatic head plus a 50-psi over-pressure. The north, or operating tank, has a 16-inch flanged nozzle centrally located in the bottom of the tank to allow forced circulation at rates up to 5000 gpm through the core. The coolant system includes a 1 MW heat exchanger for stabilizing the pool temperature during power operation and cooling the pools following tests in which the bulk water temperature is raised significantly above room temperature.

The reactor core support structure is suspended from a control bridge which spans the width of one tank and which is movable, on rails, for the full length of both tanks.

A detailed description of the Spert IV facility may be found in Reference 9.

2. DESCRIPTION OF REACTOR CORE COMPONENTS

The initial Spert IV core (Figure 2) is a highly enriched, uranium-aluminum, plate-type core consisting of 20 fuel assemblies, 4 control rod assemblies, and one transient rod assembly in a 5 x 5 array^[a]. Control of the reactor is

[a] Except for the position of the control rods within the core, this core is identical to the core which was used in a destructive test series^[10] in the Spert I facility.

Fig. 1 Spert IV reactor vessels.

Fig. 2 Plan view of Spert IV D-12/25 core.

accomplished by the use of four control rods, each of which contains two neutron-absorbing blades. The transient rod, which is used for step insertions of reactivity, is similar to the control rods with the exception that its poison section is below the active core region and normally is pulled into the core only just prior to the initiation of a transient test. The principal design characteristics of the Spert IV D-core are summarized in Table I.

2.1 Fuel Assembly

The Spert D-12 type assembly (Figures 3 and 4) which is used consists of a 3-inch-square by 27-5/8-inch-long aluminum retaining can, a 2.7-inch-square

TABLE I

DESIGN CHARACTERISTICS OF SPERT IV D-12/25 CORE

	Core
Size	15 x 15 x 24 in.
Total core volume	5.4×10^3 in. ³
Moderator volume	3.25×10^3 in. ³
Metal volume	2.14×10^3 in. ³
Heat transfer area	230 ft ²
Metal-to-water ratio	0.66
H/U ratio	360
	Fuel Assemblies
Type	Plate
Number	
-standard	20
-modified (control)	5
Size	3 x 3 x 25-1/8 in.
Plates per assembly	
-standard	12
-modified	6
Total number of fuel plates in core	270
Fuel plate thickness	0.060 in.
Meat thickness	0.020 in.
Clad thickness	0.020 in.
Normal coolant channel thickness	0.179 in.
Outside coolant channel thickness	0.090 in.
Active length of fuel plate	24 in.
Meat	93% enriched U-Al alloy
Cladding	Al (6061)
U-235 content of plate	14.0 $\pm$ 0.2 g
	Control Rods
Number	
-control	4 gang-operated
-transient	1
Composition	"Binal" (7 wt% boron-aluminum alloy)
Total rod worth	20\$
Withdrawal rate	0 to 12 in./min
Scram time	Approximately 300 msec from upper limit

aluminum lower end box, two grooved side plates, a lifting bail, and 12 removable fuel plates. The lifting bail and the lower end box perform the secondary function of restricting the vertical movement of the fuel plates within the assembly.

Fig. 3 Isometric view of 12-plate type D assembly.

Fig. 4 Plan view of 12-plate type D fuel and control rod assemblies.

The fuel in each plate consists of 14 grams of U-235 alloyed with aluminum melting stock to produce a "meat section" 0.020 inch thick, 2.45 inches wide, and 24 inches long. This fuel section is clad with 6061 aluminum to form a plate with overall dimensions of 2.704 inches wide, 25-1/8 inches long, and 0.060 inch thick. A fully loaded fuel assembly contains 168 grams of U-235. The standard water channel thickness between plates is nominally 0.179 inch. The outside water channels in each assembly have a thickness which is one-half that of the standard channels so that the outside fuel plates of adjacent assemblies will have the same amount of water between them as between adjacent interior plates.

2.2 Control Rods

The control rods for the D core are 5/16-inch-thick by 2-inch-wide blades made of Binal ^[a] which contains seven weight percent natural boron. The upper section of the control rod blade is the Binal section, and the lower section is an aluminum follower that serves as both a guide and flux peak suppressor as the control rod blade is withdrawn. There are four control rods, each having two blades which fit in guide slots in the standard fuel assemblies. Each rodded assembly has six fuel plates instead of the normal twelve plates.

The transient rod for initiating step reactivity insertions is essentially identical to the control rods, except that the upper portion of the blade is aluminum and the lower portion is Binal. This rod is raised to decrease reactivity and dropped to increase the reactivity of the core.

3. SUMMARY OF STATIC EXPERIMENTS

3.1 General

Prior to the initiation of the kinetics testing program with any Spert reactor core, a series of tests is performed in which various parameters of the particular core variant are determined. These "static" tests, which typically include measurements of control rod worths, neutron flux distributions, void and temperature coefficients, and power calibration factors, are for the purpose of gaining sufficient information about the core to allow (a) safe conduct of the kinetics testing program, (b) appropriate placement of core instrumentation, flux detectors, thermocouples, pressure transducers, etc, and (c) correlations to be made between the dynamic behavior of the reactor and static reactor parameters. The following discussions are brief summaries of the static experiments which were performed with the Spert IV D-12/25 core. For all these tests, the core was suspended in a 7-foot-diameter by 8-foot-high calibration tank which was installed in the north, or operating reactor tank. The reason for using this tank was to provide a small, low-heat-capacity system in which to perform calorimetric power calibrations and temperature coefficient measurements. More detailed descriptions of these tests can be found in Reference 8.

[a] Trade name for Sintercast Corporation aluminum-boron powder metallurgy processed material.

3.2 Initial Critical Experiment and Operational Core Loading

Initial criticality was achieved on July 24, 1962, with 21 assemblies (3.08 kg of U-235) loaded in the core and control rods withdrawn to 20.6 inches. The critical configuration can be most easily described as a 5 x 5 array with the four corners missing. Following the initial critical experiment, the loading of the core was continued for the purpose of assembling a core with sufficient excess reactivity to allow a thorough investigation of reactor instabilities at various initial conditions of temperature, flow, and upper reflector height. Based on the Spert I stabilities studies, a minimum excess reactivity of 5\$ was decided upon for the Spert IV operational core, and, for safety reasons, a minimum shutdown margin of 3\$ was required. It was determined that a core composed of a 5 x 5 array of 12-plate, type D control and fuel assemblies fulfilled these operational requirements. The total fuel loading in this core, the Spert IV D-12/25 core, was 3.75 kg of U-235. The shutdown margin for this core was determined to be 15\$ by use of the integral-count, rod-drop method [11].

3.3 Control Rod Calibration

The control rods of the Spert IV D-12/25 core were calibrated as a bank from the cold, clean critical position to the upper limit of travel by the period technique using boric acid as a reactivity shim. The differential worth of the control rods was found to vary, almost linearly, from 1.07\$/in. at the cold, clean critical position of 14.5 inches to 0.19\$/in. at their upper limit of travel, 23.2-inches. The available excess reactivity of the core was determined to be 5.3\$ from the data taken in this experiment.

3.4 Void Coefficient Measurements

The void coefficient of reactivity was measured in the D-12/25 core using aluminum strips to simulate voids. Three kinds of measurements were made:

- (1) A measurement for a uniform core distribution of full-core-length void strips;
- (2) A radial importance measurement using full-length void strips in various fuel assembly positions; and
- (3) A measurement of the vertical void importance.

The uniform void distribution was simulated for several void volumes by placing an equal number of 3/4-inch-wide x 1/8-inch-thick, full-core-length aluminum strips in each of the 20 non-rodded fuel assemblies. In this experiment, the uniform void coefficient was determined to be -41.5¢ per percent decrease in moderator density or -0.08¢/cm³ of water replaced. The coefficient is independent

of the total void volume in the core for the range investigated (up to about 11 percent moderator displacement).

For the radial importance measurements, a fuel assembly containing 24 full-core-length aluminum strips was loaded at different representative lattice positions and the resultant reactivity change determined. Figure 5 shows the void coefficient results obtained for various lattice positions.

For the vertical void importance, the reactivity worth for various vertical positions of 2-inch-high void strips distributed in a horizontal plane throughout the core was determined. The results of this experiment are shown in Figure 6.

21.1 (0.040)	32.7 (0.063)	39.9 (0.077)		
34.6 (0.066)	CR-4	60.4 (0.116)	CR-1	
41.7 (0.080)	59.4 (0.114)	TR 47.9 (0.092)	58.4 (0.112)	
	CR-3	59.1 (0.114)	CR-2	
NOTE: NUMBERS IN PARENTHESES ARE ϕ / CM^3				

Fig. 5 Local void coefficient in cents per percent moderator density change.

Fig. 6 Reactivity loss vs vertical void position.

The peak of the void worth curve has a value of -80ϕ per percent decrease in moderator density. The peak occurs at about 8.5 inches above the bottom of the fuel. The integrated value of the void worth curve in Figure 6 is -41.7ϕ per percent decrease in moderator density, which is in good agreement with the value obtained from the uniform void coefficient measurement.

3.5 Neutron Flux Distribution Measurements

During a typical short-period power excursion test, such as the type performed in the Spert experimental program, the reactor power may rise by a factor of 10^{10} or more during the course of the transient. The energy released

during this rapid surge of power causes physical changes (moderator heating, fuel plate expansion, void formation, etc) to take place which not only limit the power

rise, but also may affect some of the core characteristics, such as the neutron flux distribution, which are usually considered to be constant. Since reactor power measurements by external neutron detectors are dependent on constant ratios existing between the average core flux (more properly the fission rate) and the neutron detection rates at various ionization chamber positions, any significant change in neutron flux distribution could result in errors both in power measurements and in calculations which are based on the power behavior.

Therefore, as an aid in evaluating the validity of power measurements made during power excursion tests, a series of neutron flux distribution measurements was made in the Spert IV reactor. These measurements were made with two distinct core variants; (a) the normal operational core, and (b) the operational core with the central region partially voided with aluminum strips worth two dollars of reactivity. For each core condition, neutron flux distributions were determined from activation of 40-mil-diameter cobalt wires located at selected positions throughout the core.

Results from these tests showed that the presence of voids in the core, as during a short-period power excursion, can cause appreciable changes in the core neutron flux distribution. In the case of the voided core, for example, the neutron flux in the center of the core was definitely depressed relative to the average core flux; however, the average core neutron flux was the same for both cores. Since the operational-linear power chamber current was the same for both tests, it was concluded that the power calibration factor for the chamber in that position was not changed by the presence of voids in the core and that valid power measurements could be made at that chamber position during power excursion testing.

3.6 Calorimetric Power Calibration

In order to determine the extent the presence of steam voids affect the current-to-power relationships of chambers in various positions and to determine chamber constants to be used in the transient testing program, calorimetric power calibrations were made for the unvoided operational core and the voided core. The voided core was the same as the core used in the voided flux measurements. In these experiments, the Spert IV calibration tank was used as a calorimeter, and the bulk temperature of the reactor and moderator-reflector was raised approximately 10°C by operating the reactor at a constant indicated power level. From these experiments it was determined that for an operational-linear power chamber current of 4×10^{-4} amperes the average reactor power

during the tests was 173 ± 3 kW for the operational core, and 170 ± 3 kW for the voided core. Similar results were obtained for the other chambers which were used. On the basis of these results, it was concluded that the power calibrations of the neutron chambers to be used for the transient tests are relatively insensitive to voids which would be formed in the central region of the core during a power burst.

3.7 Isothermal Temperature Coefficient Measurements

Isothermal temperature coefficient measurements have been made for the Spert IV D-12/25 core over the temperature range from 20 to 35°C. The temperature coefficient was found to be proportional to the coefficient of expansion of water and to vary from approximately $-0.7\phi/^\circ\text{C}$ at 20°C to $-1.2\phi/^\circ\text{C}$ at 35°C.

3.8 Table of Static Characteristics

Table II is a summary of the static characteristics of the Spert IV D-12/25 core which were determined during these tests.

TABLE II

STATIC CHARACTERISTICS OF THE SPERT IV D-12/25 CORE

Critical U-235 mass	3.0 kg (21 fuel assemblies)
Operational U-235 mass	3.75 kg
Cold excess reactivity	5.3\$
Control rod worth at cold clean, critical	1.07\$/in.
Core shutdown margin	15\$
Average neutron flux per watt	6.1×10^6 nv/watt
Average core cadmium ratio	7.6
Ratio of peak-to-average core flux	2.35
Temperature coefficient of reactivity	
20°C	$-0.7\phi/^\circ\text{C}$
35°C	$-1.2\phi/^\circ\text{C}$
Uniform void coefficient of reactivity	-41.5\$/% decrease in moderator density
Maximum measured local void coefficient	-60\$/% decrease in moderator density

III. DESCRIPTION OF SPERT IV D-12/25 CORE POWER EXCURSION TESTING PROGRAM

1. GENERAL

The information which can be obtained from power excursions initiated by the stepwise insertion of reactivity aids in determining the relationships between reactor theory, core design parameters, and reactor dynamic behavior, and helps establish the relative importance of some of the physical mechanisms which determine the system behavior.

Following a step insertion of reactivity, the reactor power rises exponentially until the energy release of the excursion, acting through various reactivity coefficients, causes the power to reach a maximum value and seek some lower equilibrium value, either monotonically or by damped oscillations. The significant variables which are measured during these tests are those which indicate the severity of the test and include reactor period, power, temperature, pressure, and energy.

The purpose for conducting such tests with the D-12/25 core was to establish the transient response characteristics of the core in preparation for the forthcoming stability test program. Since analysis of step-transient tests can yield information on reactivity feedback and, consequently, provide insight on the stability characteristics of the reactor, it was desirable that these tests be performed first, to aid in the planning of a significant program of stability tests which could be safely conducted without undue risk to the reactor facility.

2. DESCRIPTION OF TESTS

In the performance of power excursions initiated by stepwise reactivity additions, the reactor was initially brought to criticality at a power level of approximately 50 watts by the withdrawal from the core of the control rods. The exact control rod position necessary for the insertion of the desired excess reactivity was then determined from the control-rod-worth curves. Then the poison section of the transient rod was pulled into the core to make the reactor subcritical, and the control rods were withdrawn to the predetermined test position. Following these rod adjustments the reactor was still subcritical and would or would not be returned to the critical state by adjustment of the transient rod

prior to the initiation of the test. The test was then initiated by the sudden ejection of the poison section of the transient rod from the core, making the reactor supercritical. Tests of this type were performed in Spert IV under all system conditions for which tests are to be performed in the stability test series, with the controlled variables being the initial reactor power, reactivity insertion, bulk moderator-coolant temperature, hydrostatic head, and coolant circulation rate.

Most of the tests in the series were initiated with the reactor critical at a power level of approximately one watt. For the tests with periods shorter than 15 msec, however, it was necessary to begin with the reactor subcritical with the power level in the milliwatt range to allow completion of the reactivity insertion before the power reached significant levels.

The reactivity insertion for these tests varied from 0.8 to 2.14 dollars, resulting in asymptotic reactor periods from approximately 1 sec to 7 msec. The bulk moderator-coolant was at ambient temperature ($\approx 20^{\circ}\text{C}$) for the initiation of each test.

Tests in this series were performed under coolant circulation rates of 0, 1.2, 2.4, 6, and 12 ft/sec through the reactor core. The last mentioned value corresponds to the maximum flow rate available at the Spert IV facility, 5000 gpm. The hydrostatic head above the core for most of the tests was approximately 18 feet. However, since the earlier Spert I ^[7] tests had indicated a significant influence on the stability of the reactor for a variation in the water head, a limited investigation of the effect of this variable on power excursion tests was made by the performance of a few short-period tests with a 2-foot head.

3. TRANSIENT MEASUREMENTS AND INSTRUMENTATION

During the tests, electrical signals from the various transducers in the reactor were transmitted to a transient instrumentation room adjacent to the main reactor room. The signals were then amplified by driver amplifiers and transmitted approximately 1000 yards to the Control Center for recording on magnetic tape and high-speed recording oscillographs. For each test, a total of 72 channels of information were recorded on two 36-channel recording oscillographs with 13 of these channels also being recorded on magnetic tape. The primary variables which were recorded were reactor power, fuel plate surface temperature, transient pressure, transient flow, and reactor tank strain.

The recording of an accurate power measurement is of prime importance in reactor kinetics testing. Not only are the rate of rise of the power and the magnitude to which the power rises direct indicators of the severity of a particular excursion, but analysis of the power burst also results in the determination of other important time-dependent variables, principally energy and reactivity of the test. The principal features of the behavior which were of interest in the Spert IV transient tests were the initial asymptotic rise, the peak power, the power burst shape, and equilibrium power. To adequately record the power in these areas of interest required, in general, that the power-measuring instrumentation cover approximately six decades of power rise. Normally the reactor power will begin to deviate from the exponential approximately one to two decades below the peak. Therefore, approximately four decades of power coverage are required below the expected peak since approximately two decades are desired for the period measurement. In addition, for backup purposes, one to two decades of power coverage above the expected peak are required. For each test, normally six chambers were used, each chamber covering from approximately 2.5 to 4 decades of power rise, thus providing considerable overlap for comparison of results and coverage in the event of failure of any power channel. The types of chambers which were used were miniature ion chambers (both boron- and uranium-lined), semi-rad chambers, and conventional B¹⁰-lined ion chambers. Both the miniature ion chambers and the semi-rad chambers had linear outputs for neutron fluxes up to at least 10^{16} nv and were, therefore, positioned in the immediate reflector and in the core proper. The conventional chambers had linear outputs up to only about 10^{11} nv and were positioned correspondingly further from the core. Calibration of these chambers to the absolute power level was done by calorimetric methods.

Fuel plate surface temperature measurements are of importance in reactor kinetics testing since the more important reactivity compensating mechanisms (moderator expansion, void formation, etc) in highly-enriched, water-moderated systems depend on energy transfer from the fuel plate surfaces to the moderator. From the temperature data, such information as the transient energy distribution, heat transfer rates to the moderator, and the onset and nature (nucleate, film, etc) of moderator boiling can be deduced. In addition, extrapolations of temperature measurements from the core "hot-spots" provide information as to the maximum reactivity insertion which a core can withstand without encountering clad melting.

This threshold was used as the upper limit for the tests which were performed in the Spert IV transient test series.

The transducers used in the measurements of fuel plate surface temperatures were thermocouples composed of 0.010-inch-diameter, chromel and alumel wires which were flattened to a thickness of approximately 0.003 inch at the junction and spot welded to the fuel plate surfaces. Each junction (Figure 7) consisted of three contact points to reduce the perturbations which can be caused by the inhomogeneity of the fuel in the plate. These thermocouples were distributed throughout the core in order to obtain a three-dimensional picture of the temperature behavior, in addition to positioning them at the suspected core hot-spots.

In the temperature traces which are shown in this report, the thermocouple locations will be specified by a coordinate designation, eg, (E5 7W-3). In this nomenclature, the thermocouple would be located in the fuel assembly in grid position E5, on the west side of fuel plate number seven, three inches below the core centerline.

Fig. 7 Photograph of typical surface-type thermocouple.

Transient pressure pulses can be developed during power excursion testing, owing to moderator and metal expansion and steam formation. In general, these pulses are significant only for short-period tests and tend to increase in amplitude for shorter periods. Since the pressures are generated between the fuel plates, and the transducers can normally be positioned no closer to the pressure source than the end boxes of the fuel assemblies, an initial uncertainty due to pressure attenuation along the length of the channels is introduced. This measurement is further complicated by wave reflection, pressure multiplication, possible shock wave buildup, and the fact that most pressure-measuring devices having the required dynamic response characteristics are sensitive to radiations emitted as a result of the fission process. A detailed discussion of the interpretation of transient pressure measurements in plate-type cores is presented in Reference 12.

Pressure measurements near the core were made using unbonded-strain-gauge, diaphragm-type transducers mounted in protective steel housings. These transducers were mounted beneath the core and radially from the edge of the core to the reactor tank wall. Radiation sensitivity characteristics of the transducers were determined prior to their use, and the least sensitive were used near the core. In addition, because of their fast response characteristics, crystal-type transducers were used at the tank wall for backup purposes. Crystal transducers are unsuitable for use near the core because of their extreme radiation sensitivity.

Transient flow measurements which are made during power excursion testing are those related to the expulsion of water from the fuel assemblies by such causes as moderator heating and void formation. The flow meter to be used for these measurements should have a wide operating range, fast response characteristics, and be relatively insensitive to radiation. In addition, since these measurements are made in the fuel assembly end boxes and/or the fuel assembly tops, the flow meter should be relatively small so as to not perturb the characteristics of the particular assembly to which it is attached. In the Spert IV tests, flow measurements were made using small discs mounted in the ends of selected assemblies. Displacement of these discs (proportional to flow) by water expulsion was measured by the use of strain gauges attached to a beam supporting the discs. These meters were calibrated in a steady-state flow loop prior to their installation in the core.

The Spert IV reactor tanks were designed for a 25-foot hydrostatic head plus a 50 psi overpressure. Since pressure pulses have been observed in reactor destructive testing (eg, in Borax I ^[13] and Spert I ^[14]) which were two orders

of magnitude higher than this design value, it was decided to routinely measure the strain on the reactor tank wall in addition to the pressure at that point during all power excursion or stability tests to be conducted in the Spert IV experimental program. No test was to be performed for which a strain equivalent to more than two-thirds of the vessel yield stress would be expected on the basis of previous test results. These measurements were made using conventional strain gauge techniques involving the measurement of the change in electrical resistance of strained wires which were attached to the outside of the tank wall. No significant tank strains have been measured in any of the tests which have been accomplished thus far in the Spert IV program.

In addition to these previously mentioned measurements, high-speed motion pictures were taken of each test through ports in the reactor vessel wall.

IV. POWER EXCURSION TESTS PERFORMED WITHOUT FORCED COOLANT CIRCULATION

1. FIDUCIAL TESTS (18-FOOT HYDROSTATIC HEAD)

In order to establish a reference series of tests with which tests performed under other initial conditions could be compared, a series of self-limiting power excursion tests was performed in the period range from approximately 1 sec to 7 msec with an 18-foot hydrostatic head above the core and with no forced coolant circulation. Most of the tests were initiated with the reactor critical at a power level of approximately one watt; however, for tests with periods shorter than 15 msec, it was necessary to initiate the tests with the power in the milliwatt region in order to complete the reactivity insertion before the power reached significant levels. All tests were initiated with the bulk water at ambient temperature ($\approx 20^{\circ}\text{C}$).

1.1 Power Behavior

The power behavior during the fiducial series of tests was very similar to that observed during previous test series with various other cores in the Spert experimental programs [2]. In particular, the behavior was similar to that of the Spert I DU-12/25 core [14] except that the peak power reached with the Spert IV core was approximately 30 percent higher. In all of the tests which were performed, the reactor power exhibited an initial peak the value of which increased monotonically as the reactor period was shortened. Figure 8 shows the peak powers reached as a function of reciprocal period (α) for the tests in this series. Two straight lines with slopes on the logarithmic plot of approximately 0.8 and 1.7 for the upper and lower sections, respectively, have been fitted to the experimental data points. This characteristic is typical for cores of this type with the change of slope in the curve occurring at a reciprocal period corresponding to a reactivity insertion of approximately one dollar.

Figure 9 shows the power behavior, along with other significant variables, as a function of time for a 7.0-msec-period test, the shortest period test which was performed in this series. In this test, the power reached a peak value of approximately 875 MW before being self-limited by the energy released during the excursion. The only damage to the reactor attributable to this test was a slight increase in the fuel plate bowing already present from previous short-period tests.

Fig. 8 Peak power vs reciprocal period for Spert IV excursion tests with 18-foot hydrostatic head and no flow.

Fig. 9 Reactor power, fuel plate surface temperature, transient pressure and energy release for Spert IV 7.0 msec-period excursion with 18-foot hydrostatic head and no flow.

1.2 Energy

The amount, distribution, and rate of release of energy during a power excursion causes various physical changes to occur in the reactor system which limit the power burst and determine the post-burst behavior of the reactor. In the long-period region, where the rate of energy release is low compared with the thermal time constants of the system, moderator heating by conduction from the fuel plates is one of the most important phenomena in limiting the power burst. As the reactor period is shortened and the energy generation rate increases, a larger fraction of the total generated energy remains in the fuel plates during the excursion, and fuel plate expansion becomes a significant contributor to reactor self-shutdown. For short-period excursions, the fuel plate surface temperatures rise sufficiently that steam void formation combines with the other mechanisms in limiting the excursions. These phenomena are not considered to be the only important reactivity compensating mechanisms but are discussed to illustrate their relationships with the energy release.

The energy released during the transient tests in Spert IV was obtained as a function of time by digital integration of the power traces. Figure 10 shows

Fig. 10 Energy release at the time of peak power vs reciprocal period for Spert IV excursion tests with 18-foot hydrostatic head and no flow.

the total energy released up to the time of peak power as a function of reciprocal period for the fiducial tests. From this plot, it can be seen that the energy released decreases over the range of reciprocal period from 1 to 6 sec⁻¹ and, thereafter, increases with increasing reciprocal period. This behavior is typical of this type of core and is consistent with the theory [15] that, for reactors with shutdown processes that are relatively simple functions of the energy release, the reactivity necessary to limit a power excursion passes through a minimum in the region of prompt critical (ie, reactivity insertions of one dollar).

1.3 Fuel Plate Surface Temperatures

From a viewpoint of safety, probably the most important variables which were measured during the fiducial tests were the fuel plate surface temperatures. Although these temperatures have as their source the energy released during the power excursions, the energy if adequately dissipated will not cause damage to the reactor. All reactor materials, however, undergo physical changes (eg, melting) at fairly definite temperatures which can result in damage to the system. The fuel plate surface temperatures, therefore, are more direct indicators of the relative safety of a particular excursion test, and the fiducial series of tests was limited to those tests for which no melting would be predicted. For the short-period tests, fuel plate surface temperature predictions were obtained by extrapolating temperature results from the longer period tests which had previously been performed in the series. During the tests, fuel plate surface temperatures were measured at 24 representative positions in the core, using 0.003-inch-thick chromel-alumel thermocouples in order to obtain data for predicting the minimum-period test which could safely be performed, and to obtain the temperature distribution throughout the core.

Data showing the fuel plate surface temperature at the time of peak power and the maximum recorded fuel plate surface temperature are shown in Figures 11 and 12 as functions of the reciprocal period. These data were obtained from a thermocouple (E5 7W-3) positioned on one of the center fuel plates in the transient rod assembly which usually produced the highest recorded temperature. This position also corresponds to the peak neutron flux in the core.

It can be observed from Figure 11 that the saturation temperature (about 109°C at the core center with an 18-foot head) was not reached at the time of the power peak for tests having reciprocal periods less than 20 sec⁻¹, and that nonboiling mechanisms are sufficient to halt power excursions with reciprocal periods in this region. For larger reciprocal periods, the temperatures at the

Fig. 11 Fuel plate surface temperature at the time of peak power vs reciprocal period for Spert-IV excursion tests with 18-foot hydrostatic head and no flow.

Fig. 12 Maximum fuel plate surface temperature vs reciprocal period for Spert IV excursion tests with 18-foot hydrostatic head and no flow.

time of the power peak increase beyond the saturation point, and the boiling region spreads out from the center of the core making boiling an increasingly significant contributor in limiting the excursions. The temperature at the time of the power peak is observed to slightly decrease over the range of reciprocal periods from 1 to 6 sec^{-1} and, therefore, increase with reciprocal period, reflecting the energy dependence.

The maximum fuel plate surface temperature generally increased with increasing reciprocal period throughout the range covered in this series of tests. There is a slight increase in the slope of the maximum temperature versus reciprocal period curve for $\alpha > 20 \text{ sec}^{-1}$ and a sharp increase for $\alpha > 120 \text{ sec}^{-1}$. For $\alpha < 20 \text{ sec}^{-1}$; conduction of thermal energy to the moderator and the effective heat removal mechanism of nucleate boiling prevent the fuel surfaces from reaching temperatures much above the saturation point. In the region of reciprocal period from 20 to 120 sec^{-1} occasional film boiling occurs resulting in the increase in the slope of the maximum temperature curve. For $\alpha > 120 \text{ sec}^{-1}$ film boiling is established while the power is still in the region of the initial peak, thus allowing the temperatures to exhibit a rapid rise due to the high energy generation rates within the fuel plates. The rapid rise in the temperature trace in the region of the power peak shown in Figure 9 is due to this type of behavior. Extrapolation of maximum temperature data indicates that fuel plate melting would be expected for self-limiting excursions with periods shorter than 6 msec.

1.4 Transient Pressure

In this series of tests, pressure generated during the power excursions was measured using unbonded-strain-gauge-type transducers positioned as shown in Figure 13. Measurable pressure pulses were obtained only from positions A, beneath the core, and B, at the side of the core, the other positions being too remote to sense any pressure even for the 7-msec-period test. The pressures recorded beneath the core were the result of the downward expulsion of water due to the rapid energy release during the excursion and were characterized by a damped oscillatory behavior. Figure 14 shows the relationship between the transient pressure measured beneath the core and the transient flow of water out of the bottom of the core. Figure 15 shows the maximum transient pressure measured beneath the core as a function of reciprocal period. The transient pressures which were measured at the side of the core were small (a factor of 5 to 10 less) compared with those measured beneath the core. Among the reasons

Fig. 13 Locations of pressure transducers around Spert IV core. (Transducer "A" located 26-5/16 inches below core centerline; all others are located at centerline).

for this difference is that the mass movement of water in the horizontal direction was restricted by the fuel assembly cans, and the position at the side of the core was subject to more pressure relief owing to reflections from the free surface than for the position beneath the core.

2. TWO-FOOT HEAD TESTS

The water head above the core was reduced from 18 to 2 feet, and tests with asymptotic periods of 20, 12, and 8.8 msec were performed to investigate the effect of variations in hydrostatic head on the response of the Spert IV D-12/25 core to stepwise reactivity additions. These tests were limited to the short-period

Fig. 14 Transient pressure and flow vs time for 10-msec test with 18-foot hydrostatic head and no forced flow.

Fig. 15 Maximum transient pressure vs reciprocal period for Spert IV excursion tests with 18-foot hydrostatic head and no flow.

region, where boiling contributes to the shutdown process and where the change in hydrostatic pressure of 6.9 psi (change in boiling point of 14°C) should produce the greatest change in reactor response. The following results and conclusions were obtained from this series of tests.

(1) There was no significant change observed in the peak power, burst shape, or energy release as a result of lowering the hydrostatic head. Figure 16 shows a power burst comparison for tests with periods of approximately 8.5 msec, the shortest period for which comparative tests were performed. In these tests, the peak power for the 2-foot-head test was only about seven percent lower than with the 18-foot head. Considering the slightly shorter period for the 18-foot head test (8.5 versus 8.8 msec), there is essentially no difference either in peak power or burst shape between the two tests.

Fig. 16 Reactor power vs time for 8.5-msec-period tests with 18-foot and 2-foot hydrostatic heads.

(2) The measured fuel plate surface temperatures were higher for all tests with the higher head. The difference in temperatures at the time of peak power (Figure 17) was about 15°C and the differences in the maximum temperatures (Figure 18) ranged from approximately 25 to 35°C. The lower temperatures are

Fig. 17 Fuel plate surface temperature at the time of peak power vs reciprocal period for Spert IV excursion tests with 18- and 2-foot hydrostatic heads and no flow.

Fig. 18 Maximum fuel plate surface temperature vs reciprocal period for Spert IV excursion tests with 18- and 2-foot hydrostatic heads and no flow.

to be expected for lower-head tests since the highly effective heat transfer mechanism of nucleate boiling is established earlier in the excursion, thereby reducing the energy stored in the fuel plates in comparison with the higher-head tests. The fuel plate surface temperatures measured in the two tests with periods of approximately 8.5 msec are shown in Figure 19. Examination of the temperature data from these tests shows that the "break" from the initial exponential rise, corresponding to the onset of nucleate boiling, occurs at a temperature approximately 20°C lower for the 2-foot head than for the 18-foot head.

Fig. 19 Fuel plate surface temperature vs time for 8.5-msec-period tests with 18- and 2-foot hydrostatic heads.

(3) Although the data are somewhat conflicting, it appears that there was no significant change in the transient pressures generated within the core as a result of the change in hydrostatic head. For the pressures measured beneath the core, there was little difference for the two conditions, except for the 12-msec tests where the measured pressure was 12 psig for the 2-foot head compared with 15 psig for the 18-foot head. The pressures measured at the side of the core, however, were approximately a factor of three higher for the 18-foot-head tests than for the 2-foot-head tests. This difference is thought to be a consequence

of the measurements at the side of the core being affected more by pressure relief from the free surface during tests with the lower head. Since the pressure measurements beneath the core are considerably larger and apparently are not significantly affected by the free surface, it is concluded that the pressure measurements at that position are more representative of the pressure generation within the core than those measurements made at the side of the core. Hence, the conclusion is drawn that there was no significant change in pressure generation due to the change in head.

These results can be compared with results from previous tests involving a change in hydrostatic head or initial system pressure which have been performed in the other Spert reactors.

A series of transient tests was performed in Spert I with the A-17/28 core in which the tests were performed with both a 2-foot and 9-foot hydrostatic head above the core [16]. There was essentially no difference in the test results for the tests initiated at ambient temperatures. For the tests initiated with the bulk water temperature at boiling, however, the peak power reached in the 9-foot tests was approximately a factor of two higher than for the 2-foot tests, and the power bursts were considerably broader for the higher head tests.

A series of tests was performed in Spert III in which the initial system pressure was raised from atmospheric to 2500 psig [17]. Results from these tests showed that for the shorter period tests, where boiling contributes to the shutdown mechanism at atmospheric pressure, the peak powers were increased by approximately 30 percent as the pressure was increased to 2500 psig with most of the change occurring in the first 50 psi of pressure increase. Also, in the higher-pressure tests, the power burst shape was somewhat broadened, and, of course, the energy release was increased. These tests demonstrated that non-boiling shutdown mechanisms can be effective in limiting short-period excursions.

Transient pressure measurements which have been made during power excursions initiated at atmospheric pressure and at elevated pressures in Spert II [18] and Spert III [6] have shown that an increase in system pressure up to approximately 75 psig causes increases in the transient pressures observed. With further increases in initial system pressures, the transient pressures either cease to increase or even decline.

The limited amount of data available from the investigation of the hydrostatic head effect in Spert IV are in general agreement with the results from the other reactors and support the Spert I results where it was found that

raising the hydrostatic head above the core had no significant effect on transients initiated at ambient temperatures.

V. THE EFFECT OF FORCED COOLANT CIRCULATION ON REACTOR POWER EXCURSION BEHAVIOR

1. GENERAL

A series of tests was performed to investigate the effect of forced coolant circulation on the transient behavior of the Spert IV D-12/25 core. The tests were performed over the range of asymptotic periods from approximately 500 to 10 msec with forced-coolant flow rates of 5000, 2500, 1000, and 500 gpm bulk coolant flow, corresponding to rates of 12.0, 6.0, 2.4, and 1.2 ft/sec, respectively, through the core. The coolant flow was directed upward through the 25-assembly core from a 16-inch-nozzle opening in the floor of the reactor pool and a transition flow piece connecting the nozzle to the bottom of the core support structure. Coolant pump suction was taken from a 12-inch connection at the pool wall. Grid filler pieces were installed in those grid positions not occupied by fuel assemblies in order that all the flow be directed through the core.

By providing a continuous supply of ambient-temperature water to the core inlet and by increasing the water velocity through the core, both the heat transfer rates from the fuel to the moderator and the energy removal rate from the core are increased. Depending on the rate of increase of the power, the reactivity feedback mechanisms which depend on the energy stored in the core are delayed, suppressed, or, as in the case of moderator boiling, even eliminated by the addition of flow. The extent of the effect on these physical phenomena, therefore, determines the effect of flow on the power excursion.

2. THE EFFECT OF FLOW ON POWER BEHAVIOR

In the long-period region, the addition of forced flow resulted in a substantially different power behavior than that resulting during tests without flow. For instance, in the tests which were performed with periods of approximately 500 msec, a definite power peak was eliminated for all tests with flow. Instead of peaking, the power continued to rise until thermal compensation of reactivity caused the power to approach an equilibrium value. In addition, the magnitudes of the equilibrium values were higher for the higher flow rates. The tests in this period region which were performed without flow exhibited definite initial peaks of approximately one megawatt and a subsequent equilibrium power of approximately

one-half megawatt. Figure 20 shows the power as a function of time for tests in the period region of 500 msec for various forced circulation rates.

Fig. 20 Reactor power as a function of time for 500-msec excursion tests with various flow rates.

For tests in the period region from 100 msec to 50 msec, a definite power peak occurred with both the peak and equilibrium values increasing with the flow rate. In addition, it was observed that the power bursts were broadened as the flow rate was increased. The peak power reached for the tests with reactor periods of 50 msec ranged from approximately 25 MW for the tests without flow up to 30 MW for the test with 12 ft/sec. Figure 21 shows the power as a function of time for the tests with initial periods of 50 msec for the various flow rates.

For tests with periods shorter than about 50 msec, little change was observed in the initial power burst with variation in the water circulation rate. For these short-period excursions, the power is rising too fast for the flow to significantly influence the energy distribution during the initial burst. For example, in a 10-msec-period transient with the highest available flow rate at Spert IV, the coolant moves only about three inches in the D-12/25 core in the time that the power

Fig. 21 Reactor power as a function of time for 50-msec excursion tests with various flow rates.

risks by a factor of ten. It is expected, however, that higher flow rates would have a more significant effect on the short-period burst behavior. Figure 22 shows the power as a function of time for the 10-msec-period tests which were performed with various coolant circulation rates.

The addition of flow caused significant changes in the post-burst power behavior and/or the equilibrium level reached for a given reactivity insertion. In general, the post-burst behavior was observed to be oscillatory in nature for the tests with flow. In the long-period region these oscillations took the form of small, random fluctuations about the equilibrium level. In the short-period tests, however, the equilibrium power level was approached in a damped series of large, secondary power bursts. As the initial period becomes shorter, steam formation becomes a more significant shutdown mode and the magnitudes of the secondary power oscillations become larger because the growth and collapse of the steam bubbles following the initial burst cause larger changes in reactivity. For a particular period, the steam bubble collapse becomes more complete as the flow rate is increased, thereby causing greater amplitudes in the post-burst oscillations. In the 10-msec-period test with 12 ft/sec, shown in Figure 22, the

Fig. 22 Reactor power as a function of time for 10-msec excursion tests with various flow rates.

initial peak of 480 MW was followed by successive peaks of approximately 140, 100, 70, and 50 MW with a frequency of between 5 and 6 cps.

The frequency of the observed oscillations was determined to increase with both the reactivity insertion and the flow rate. As the reactivity insertion is increased, a greater amount of reactivity is compensated by steam void and, therefore, is readily available to be returned to the system. This greater amount of returned reactivity causes faster secondary power rises and a consequent increase in the oscillation frequency. For the tests with 12 ft/sec flow, the oscillation frequency was 2-1/2 cps with a reactivity insertion of 1.14\$ ($\tau = 50$ msec) and increased to 5-1/2 cps for an insertion of 1.8\$ ($\tau = 10$ msec). For the 10-msec-period tests, the oscillation frequencies were approximately 2-1/2, 4, and 5-1/2 cps for flow rates of 2.4, 6.0, and 12 ft/sec, respectively.

Although many of the tests were terminated prior to the establishment of true equilibrium power levels, it was determined that the equilibrium levels varied approximately proportionally with the flow rate and reactivity insertion.

3. ENERGY RELEASES

In general, the addition of forced coolant circulation to power excursion tests produced higher energy releases over the entire range of reactor periods investigated. These higher energy releases occurred due to the higher peak powers and broadened power bursts which resulted from an increase in flow. For the energy which was released during the initial power burst, the addition of flow made the greatest percentage increase in the longer period tests with the effect diminishing for shorter periods. For the tests which were performed with periods of approximately 100 msec (the longest period for which well-defined bursts occur with flow), the energy release during the power burst for the test with 12 ft/sec flow was approximately 50 percent higher than for the test without flow. This difference decreased to about 20 percent for the tests with initial periods of 10 msec. Figures 23, 24, and 25 show power burst comparisons for reactor periods of approximately 100, 50, and 10 msec for tests without flow and tests with 12 ft/sec flow. Figure 26 shows the energy released up to the time of the power peak as a function of reciprocal period for both the tests without flow and the tests with 12 ft/sec flow. Intermediate flow rate tests had energy releases bounded by these curves.

Fig. 23 Reactor power and fuel plate surface temperature for 100-msec-period tests with and without forced coolant flow.

Fig. 24 Reactor power and fuel plate surface temperature for 50-msec-period tests with and without forced coolant flow.

Fig. 25 Reactor power and fuel plate surface temperature for 10-msec-period tests with and without forced coolant flow.

Fig. 26 Energy release at the time of peak power vs reciprocal period for various water head and flow conditions.

The energy releases following the initial burst were considerably higher for the tests with flow reflecting the dependence of the equilibrium power on the flow rate.

4. EFFECT OF FLOW ON FUEL PLATE SURFACE TEMPERATURE

The D-12/25 core has two types of moderator-coolant channels with distinctly different thermal-hydraulic characteristics. In each assembly, the two outermost moderator-coolant channels are one-half the thickness of the standard interior channels so that the two outside fuel plates of adjacent assemblies have the same amount of moderator between them as between two adjacent interior fuel plates in order to reduce flux peaking between assemblies. This design presented no particular problems for tests without flow; however, the increased flow resistance in the narrow channels resulted in decreased water velocities in those channels relative to the standard channels, and burnout problems were encountered in the tests with flow. Therefore, the discussion of the effect of forced coolant on fuel plate surface temperatures is presented with respect to these two types of channels.

4.1 Surface Temperatures from Fuel Plates Adjacent to the Standard (0.180 inch) Moderator-Coolant Channels

Figures 27 and 28 show the fuel plate surface temperature measured at the time of peak power and the maximum fuel plate surface temperature measured during the tests as functions of reciprocal period for tests without flow and tests with 12 ft/sec forced coolant circulation. As can be seen in the Figures, the addition of flow caused a reduction in both of these temperatures with the most significant change occurring in the longer period regions where the water velocities were significant compared with the rate of power rise.

In the long-period region, the temperatures at the time of peak power were approximately 30°C cooler in the tests with flow than for the tests without flow. This difference decreased to about 10°C for tests with periods of approximately 100 msec and became insignificant for the 10-msec-period tests. As can be seen in Figure 27, there is a short range in reciprocal periods (from $\alpha = 10$ to 20 sec^{-1}) where the tests with and without flow yield approximately the same dependence of temperature at the time of peak power on reciprocal period, as evidenced by the uniformity of slopes between the two cases. The point at which this uniformity begins to fail corresponds to the point where the heat transfer

Fig. 27 Fuel plate surface temperatures at the time of peak power vs reciprocal period for various water heads and flow conditions.

Fig. 28 Maximum fuel plate surface temperatures vs reciprocal period for various water heads and flow rates.

characteristics for the tests without flow change due to the onset of nucleate boiling. Under conditions of flow, the continuous supply of cooler water to the core inlet extended this region of essentially linear dependence of temperature on reciprocal period to approximately $\alpha = 40 \text{ sec}^{-1}$ beyond which the two temperature curves converge.

The behavior of the maximum fuel plate surface temperatures (Figure 28) was quite similar to that of the temperature at the time of peak power (Figure 27). An interesting difference lies in the region between $\alpha = 10$ to 30 sec^{-1} . In this region the maximum temperature curve does not have the sharp increase in slope seen in the temperatures at the time of peak power. The reason for this difference is that the maximum temperatures reached in this region of reciprocal period were at or near the saturation temperature and were held near that temperature because of the high heat transfer rates to the moderator which accompany the nucleate boiling phase. The temperatures at the time of peak power, however, were still below the saturation temperature in this region and increased with the energy release for increasing reciprocal periods. ||

Figures 23, 24, and 25 show the fuel plate surface temperatures along with the power bursts for reactor periods of 100, 50, and 10 msec, both with and without

flow. In each test it was evident that although the power levels with flow were higher at a given instant and consequently the energy being released in the plate was greater, the fuel plate surface temperatures were lower, indicative of the increased heat removal capacity of the system with flow.

Figures 29 and 30 present fuel plate surface temperatures as functions of time for various flow-rate tests with periods of 50 and 10 msec. A comparison

Fig. 29 Fuel plate surface temperature rise during Spert IV 50-msec-period excursion tests with various flow rates (thermocouple near flux peak on centrally located fuel plate E5 7W-3).

Fig. 30 Fuel plate surface temperature rise during Spert IV 10-msec-period excursion tests at various flow rates (thermocouple near flux peak on centrally located fuel plate E5 7W-3).

of these figures with Figures 21 and 22 shows that, as the flow rate was increased, the temperatures more closely followed the reactor power. This again can be attributed to the increased heat removal capacity as the flow rate was increased. One of the primary factors which influences the fuel plate surface temperature is the mode of heat removal, and during a transient test, the heat transfer rates may vary considerably as the moderator changes through the subcooled, nucleate

boiling, and film boiling phases. By reducing the energy stored in the moderator, the addition of flow reduces the randomness of transitions through these various phases, thus allowing the fuel plate surface temperature to more accurately reflect the energy generation within the plate.

4.2 Surface Temperatures from Fuel Plates Adjacent to the Narrow(0.090-inch) Moderator-Coolant Channels

In general, for the tests with flow, the fuel plate surfaces adjacent to the narrow moderator-coolant channels reached higher temperatures than fuel plate surfaces from the same core region which were adjacent to the wider channels. Also, the narrow channels showed a greater tendency toward instability.

In the long-period region with 12 ft/sec flow, neither the power nor the temperature exhibits peaks but gradually rise to steady-state values about which small oscillations with a random frequency appear. For the 500-msec-period test, the power rose to approximately 20 MW with oscillations of approximately ± 1 MW about the steady-state value. Characteristic of the difference in the behavior in this period region between temperatures in the wide and narrow channels are the results from two thermocouples in the transient rod assembly (E5 7W-3) and (E5 1W-3). In the wide channel, the fuel surface temperature rose to 115°C with fluctuations of $\pm 1^\circ\text{C}$, while the temperature in the narrow channel reached 130°C with fluctuations of $\pm 10^\circ\text{C}$.

For reactor periods shorter than 100 msec, both the reactor power and fuel plate surface temperatures exhibited an initial peak, with the post-peak region having a tendency to oscillate. For shorter periods, the initial peak fuel plate surface temperatures were a function of the core neutron flux distribution, and, therefore, temperatures from the same region of the core reached approximately the same initial peak temperatures. The post-peak temperature behavior, however, was strongly dependent on the core hydraulics, and the fuel plate surfaces next to the narrow coolant channels reached higher temperatures than those next to the wide channels. Figure 31 shows the power trace from one of the 10-msec-period tests, with temperature traces from a wide channel (E5 7W-3) and a narrow channel (E5 1W-3).

One of the most significant differences in behavior between temperatures from the two types of channels was in their predictability in the short-period region. For fuel plate surfaces adjacent to the wide channels, the maximum temperatures measured during any given test were always associated with the initial power peak, and extrapolations of previous test results provided reliable predictions for the maximum expected fuel plate surface temperatures for all

Fig. 31 Reactor power and fuel plate surface temperature from wide and narrow channels vs time for 10-msec tests (wide channel E5 7W-3; narrow channel, E5 1W-3).

tests which were performed with forced flow. For the narrow channels, however, the maximum temperatures were not necessarily associated with the initial power peak and could occur at anytime following the initial rise. Figure 32 shows a temperature trace from a narrow channel (E6 1W-3) which was in a fuel assembly adjacent to the transient rod assembly. This trace shows a large temperature spike which started just prior to the initiation of the reactor scram and continued rising thereafter. Three thermocouples on that particular plate all showed the same behavior. The relative predictability of fuel plate surface temperatures from the two types of channels can be seen from the range of maximum temperatures reached during the three 10-msec-period tests which were performed in this series. The temperatures recorded at (E5 7W-3), one of the hotter-reading, wide-channel thermocouples, ranged from 188 to 200°C, while one of the narrow-channel thermocouples (E6 1W-3) ranged from 203 to 253°C.

Fig. 32 Temperature from narrow channel (E6 1W-3) vs time for 10-msec-period test with 12 ft/sec forced coolant flow.

5. MELTED FUEL PLATE

A detailed inspection of the core following one of the short-period tests revealed a large hole (Figure 33) in the fuel plate in channel number 1 of the transient rod assembly (Figure 34). The hole was approximately 7 inches long and 1-1/2 inches wide at its widest point. Inspection of the rest of the core revealed no damage other than fuel plate bowing and rippling. Investigation into the cause of the melting indicated that melting occurred during a 10-msec-period test with 5000 gpm forced coolant circulation. In this test the initial power peak of 480 MW was followed by successive peaks of approximately 140, 100, 70, and 50 MW and subsequent small oscillations about a steady power level of 30 to 35 MW for approximately 3.3 seconds. It is postulated that mechanical deformation of the plate, either prior to the test or during the initial burst, caused blocking of the narrow, outside moderator-coolant channel, resulting in steam-blanketing and burnout of the fuel plate. The normal flow velocity in the narrow channels of this assembly at 5000 gpm is 5 ft/sec compared with 12 ft/sec for the average core flow velocity. Reexamination of the moving pictures of this test showed several pieces of what appeared to be fuel plate material falling in the

FLOW →

Fig. 33 Photograph of Spert IV melted fuel plate.

reflector region of the core immediately following the termination of the test. The presence of air activity following this test added further evidence that the melting occurred at this time. Samples of the pool water which were taken immediately following the test, however, revealed no contamination.

Subsequent metallurgical examination of the melted fuel plate, a fuel plate from a symmetrical position which did not melt, and a plate from the center of the transient rod assembly showed that the only evidence of fuel plate damage was in the immediate vicinity of the melted area. The examination also showed that the side of the fuel plate which was adjacent to the narrow, outside moderator-coolant channel reached a considerably hotter temperature than the other side of the plate. By the presence of UAl_3 in the fuel section and on the outside of the plate next to the narrow channel, it was estimated that the hot side of the plate immediately adjacent to the hole reached a temperature between 730 and 860°C. The other side of the plate, however, only showed grain boundary melting which occurs at temperatures between 582 and 649°C.

Fig. 34 Cross section drawing of Spert IV transient rod assembly showing location of melted fuel plate.

Examination of a sample of the melted plate which was taken at the top of the melted hole supported the conclusion that the melting occurred during a test with up flow. This sample appeared to have been melted, but closer examination revealed that the melted-appearing area was lying over sound plate material.

In order to gain a better understanding of the conditions which resulted in the melting and to reveal any problems which may be encountered as a result of the restricted flow in the outside moderator-coolant channels during the Spert IV stability tests, an additional series of power excursion tests with 12 ft/sec flow was performed. Tests with initial asymptotic periods of 500, 50, 20, 15, 12, and 10 msec were performed in this series. Following each of the tests a detailed inspection of the core was made in order to determine the effects of a given test on the physical condition of the core and to ascertain the condition of the core prior to the initiation of the succeeding test.

Consideration was given in this test series to the various mechanisms which could cause blockage of a coolant channel and consequently the melting of a portion of a fuel plate. One such mechanism considered was the bending of a fuel plate due to thermal stresses from a large temperature differential across the

plate during the test. In order to evaluate the effect of this mechanism, a fuel plate with thermocouples on both sides was installed in the core position which had been occupied by the melted plate. In all of the tests which were performed, a differential of 15 to 25°C existed across the plate. Figure 35 shows this differential as it existed across the plate during a 20-msec-period test. Although

Fig. 35 Temperature traces from opposite sides of a fuel plate in a narrow channel during a transient test with 10 ft/sec forced coolant flow.

the magnitude of this temperature differential was not sufficient to explain the channel blockage, should the narrow channel be partially blocked either prior to a test or during a test, a larger differential could exist which could make this mechanism self-propagating.

Another mechanism which could cause blockage is the tendency of the fuel plates to bend due to thermal expansion when tightly constrained in the fuel assembly between the lifting bail at the top and the lower end box at the bottom. To determine the effect of this mechanism, a special fuel assembly was installed in which one-half of the fuel plates were tightly constrained and the other half were free to expand. This assembly was in the core for several of the short-period tests, including three 10-msec-period tests. Inspection of this fuel assembly revealed that the degree of fuel plate constraint had little effect on permanent

fuel plate bowing. The possibility of a temporary bowing of constrained fuel plates existing during a test when the fuel plates are at elevated temperatures cannot be ruled out, however.

Other mechanisms which could conceivably cause blockage of a coolant channel are the deformation of a plate due to a pressure pulse during the initial power burst, mechanical damage to the assembly during loading or unloading of the core (the narrow outside channels are particularly susceptible to this kind of damage), foreign objects becoming lodged between fuel plates, or any combination of the various mechanisms.

From the results of this series of tests, it was concluded that the melting of the fuel plate was not a typical result of a 10-msec-period transient with 12 ft/sec forced coolant flow. This conclusion was supported by the fact that three additional 10-msec-period transients were performed without resulting damage to the core other than minor fuel plate bowing.

The physical condition of the core prior to the initiation of a test has a strong influence on the ability of the test to be performed without resulting fuel plate melting. For example, the reactivity insertion required to produce a 500-msec-period also will produce a steady state power level of 20 MW with 12 ft/sec flow. This means that any of the transient tests which were performed with 12 ft/sec flow could have resulted in fuel plate melting, should a coolant channel have been blocked either before or during the test. Naturally, a channel is more likely to become blocked during the short-period tests because of the stresses to which the core is subjected during the high rate of energy release accompanying the initial power burst.

It also was concluded that the fuel plate melting was not a reasonably predictable event on the basis of previous test results. Extrapolation of maximum temperature data from all tests with periods longer than 10 msec indicated that the maximum temperature expected for a 10-msec-period test should be less than 220°C. To predict a melting incident, similar to the one which occurred, would require complete knowledge of the temperature behavior of each fuel plate surface during the previous tests and of the physical condition of each fuel plate and coolant channel prior to the next test. Even with this information, prediction of a melting event would be improbable if the channel blockage were initiated during the test in which melting occurs, unless, of course, enough information were available to ascertain that the test would likely result in melting, regardless of the initial physical condition of the core.

VI. CONCLUSIONS

From the series of tests described in this report, the following conclusions have been derived concerning the response of the Spert IV D-12/25 core to stepwise insertions of reactivity and the effect of variations in forced coolant circulation and hydrostatic head on such response.

In the fiducial tests which were performed with an 18-foot hydrostatic head, it was determined that the reactor could safely self-limit a stepwise reactivity insertion of 2.14\$ which resulted in an asymptotic reactor period of 7 msec, a peak power of 875 MW, a burst energy release of 12 MW-sec, a maximum fuel plate surface temperature of 338°C, and a peak transient pressure of 21 psig. It also was determined during these tests that the onset of core damage and minor fuel plate bowing occurred for a reactor period of 15 msec, and that fuel plate melting could be expected for periods shorter than 6 msec. Results from these tests showed that nonboiling shutdown mechanisms (moderator heating, fuel plate expansion, etc) are sufficient to limit power excursions with periods longer than 50 msec, but as the period is shortened beyond this value, boiling void formation became one of the significant shutdown mechanisms.

It was determined in this test series that a change in the hydrostatic head from 18 feet to 2 feet at ambient temperature produces no significant change in the response of the system for periods as short as 8.8 msec, the shortest period test which was performed with the reduced head.

In the long-period region ($\tau \approx 500$ msec), the addition of flow eliminated the formation of an initial power peak for all flow rates investigated, 1.2 ft/sec to 12 ft/sec. In these tests, the power continued to rise monotonically to an equilibrium value. For reactor periods between 100 and 50 msec, all tests exhibited an initial peak with the magnitude of the peak being increased and the burst shape being broadened for increasing flow rates. This effect results in higher energy releases as the flow rate is increased. For the shorter period tests, the addition of flow produced little effect on the initial burst, although there was a tendency for the tests with flow to have slightly higher powers and broader burst shapes.

In the post-peak region, the power in the tests with flow exhibited an oscillatory behavior with the mean power level being approximately proportional to the flow rate. It was determined that both the amplitude and the frequency of the oscillations increase for increased flow rates.

In general, the fuel plate surface temperatures were lower for the tests with flow; with the difference decreasing as the reactor period was decreased. There was, however, an additional problem with respect to fuel plate surface temperatures which was observed during the tests with flow. This problem was a consequence of the reduced flow rates in the narrow outside moderator-coolant channels (0.090 inch thick compared with 0.180 inch for the standard channels) in the fuel, control-rod, and transient-rod assemblies, which allowed these channels to depart from nucleate boiling at substantially lower power levels than the remainder of the core. The test results showed that the temperatures of fuel plates adjacent to these channels were higher, showed a greater tendency toward instability, and were more difficult to predict than temperatures of fuel plates adjacent to the wide channels. It was concluded, however, that the melting of a fuel plate which occurred during a 10-msec test with 12-ft/sec flow was not a typical result. This conclusion was supported by the performance of three identical tests which resulted in only minor fuel plate bowing. It was further concluded that any of the tests with flow, with their higher equilibrium powers, could have resulted in fuel plate melting if a moderator channel became blocked and the reactor allowed to operate at equilibrium power for a few seconds.

It should be pointed out that the conclusions which were drawn from this test series are not unique to this core, and except for the use of specific numbers, can be applied in general to all cores of similar design. In fact, most of these conclusions were anticipated prior to the performance of these tests from tests previously performed in the various Spert reactors. The uncertainties which are associated with the performance of instability testing, however, made it desirable to experimentally verify the transient response of the specific core under consideration for all the various available operating conditions prior to the performance of the instability test program.

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APPENDIX A

REDUCED PROMPT NEUTRON GENERATION TIME
($\Lambda/\beta_{\text{EFF}}$) FOR THE SPERT IV D-12/25 CORE

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APPENDIX A

REDUCED PROMPT NEUTRON GENERATION TIME ($\Lambda/\beta_{\text{EFF}}$) FOR THE SPERT IV D-12/25 CORE

The value of $\Lambda/\beta_{\text{eff}}$ for the Spert IV D-12/25 core was determined from the tests which were conducted in this series under various initial plant conditions of flow and hydrostatic head. In these transients the initially injected reactivity, R , varied from about 0.80 to 2.14\$. The reactivity was inserted as a step by rapid ejection of the transient rod, and the amount of reactivity inserted was determined from a prior reactivity calibration of the control rods. The rod calibration itself was accomplished by long-period measurements, corresponding to reactivities that were essentially determined by the delayed reactivity term

$$R_d(\alpha) = \sum_i \frac{a_i \alpha}{\lambda_i + \alpha}$$

of the inhour relation,

$$R = \frac{\alpha \ell}{k \beta_{\text{eff}}} + R_d \quad (1)$$

or

$$R = \frac{\alpha \Lambda}{\beta_{\text{eff}}} + R_d \quad (2)$$

where

R = total (prompt and delayed) reactivity (\$),

ℓ = prompt neutron lifetime (sec),

Λ = effective mean prompt-neutron generation time,

$a_i = \beta_{i\text{eff}}/\beta_{\text{eff}}$ = absolute delayed-neutron yield per fission for the i^{th} delayed-neutron group,

α = reciprocal reactor period (sec^{-1}),

λ_i = i^{th} delayed-neutron-group decay constant, and

k = static effective multiplication constant.

Rearranging terms in Equations (1) and (2),

$$\left(\frac{\Lambda}{\beta_{\text{eff}}}\right) \alpha = (R - R_d) \quad (3)$$

and

$$\left(\frac{\ell}{\beta_{\text{eff}}}\right) \alpha = k(R - R_d). \quad (4)$$

Lewins [19] has pointed out that for a particular core perturbation there is a distinction to be made as regards which parameter, Λ or ℓ , will be a constant during that perturbation. If departure from criticality is achieved by movement of a "black" neutron absorber, Λ , not ℓ , is constant during the change. However, if departure from criticality is achieved by means of a "gray" fuel-follower control rod, then ℓ and not Λ would be constant during the perturbation.

Because the step transients in Spert IV were accomplished by the perturbation of black neutron absorbers, $\Lambda/\beta_{\text{eff}}$ is the proper parameter to measure. A comparison of Equation (3) with Equation (4) shows that if k does not depart significantly from the value of unity, $\ell \approx \Lambda$. This latter point is demonstrated by the values given in the following paragraphs.

From Equation (3), $\Lambda/\beta_{\text{eff}}$ was obtained as the slope of the linear plot of $(R - R_d)$ as a function of α . The experimental plot obtained for the Spert IV D-12/25 core is shown in Figure A-1. A least squares linear fit of the data gave a value of

$$\begin{aligned} (\Lambda/\beta_{\text{eff}})_{\text{transient}} = \\ (7.94 \pm 0.26) \times 10^{-3} \text{ sec} \end{aligned}$$

where the error represents the standard deviation from the mean.

Setting $k = (1 - R\beta_{\text{eff}})^{-1} \approx 1 + R\beta_{\text{eff}}$ in Equation (3) (where, without introducing appreciable error, the value of β_{eff} in the correction term $R\beta_{\text{eff}}$ may be estimated from the buckling) one may rewrite Equation (3) as

Fig. A-1 Prompt reactivity vs reciprocal period for the Spert IV core.

$$(R - R_d) (1 + R\beta_{\text{eff}}) = \frac{\ell}{\beta_{\text{eff}}} \alpha.$$

Thus, ℓ/β_{eff} was obtained as the slope of the linear plot of $(R - R_d)(1 + R\beta_{\text{eff}})$ as a function of α . A least squares linear fit of this data gave a value

$$\ell/\beta_{\text{eff}} = (8.06 \pm 0.26) \times 10^{-3} \text{ sec}$$

where, again, the error represents the standard deviation from the mean.

Based on these results it is noted that the difference between $\Lambda/\beta_{\text{eff}}$ and ℓ/β_{eff} is not determinable within experimental error.

An independent measurement of $\Lambda/\beta_{\text{eff}}$ for the Spert IV D-12/25 core through a sub-critical statistical method also has been performed^[1]. In brief, the method employed the particular statistic, the variance-to-mean ratio of the count rate as a function of counting time. At the basis of such an approach is the premise that because the statistical behavior of the neutron population in a multiplying system is intrinsic to the system, a study of the neutron population statistics will yield information on fundamental parameters of the system.

The method takes into account those fluctuations in the neutron population due to both the prompt and the delayed neutrons. A model was developed which describes the statistical fluctuations in the neutron population as observed by a neutron detector in terms of the variance-to-mean ratio of the counting rate as a function of counting time.

The neutron counting data were taken at four subcritical values of reactivity ranging from $-0.7530 \pm 0.0045 \$$ to $-0.1334 \pm 0.0009 \$$. The raw counting data were recorded on magnetic tape and statistically sampled at a later time, thus reducing reactor operating time. A stainless steel regulating rod, located in the transient rod position with a small reactivity worth per unit length, was used for the accurate measurement of reactivity.

The raw counting data were statistically sampled by playing the signal from the magnetic tape through a gated scaler at gate times ranging from 0.0010 to 2.0000 sec. Approximately 800 gates were taken at each gate time and automatically recorded on punched paper tape. The unbiased variance-to-mean ratio was computed for each gate time directly from the paper tape by use of an IBM 1620 computer. The technique used for determination of $\Lambda/\beta_{\text{eff}}$ utilized a general program written for the IBM 7090. This program predicted a least-squares value of the prompt root of the Inhour equation.

From these data points, a plot, similar to Figure A-1, was generated and the slope gave a value of

$$\Delta/\beta_{\text{eff}} = (8.07 \pm 0.17) \times 10^{-3} \text{ sec}$$

for the Spert IV D-12/25 core, where the uncertainty represents the standard deviation from the weighted mean.

APPENDIX B
POWER, ENERGY, TEMPERATURE, AND PRESSURE DATA FIGURES

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APPENDIX B

POWER, ENERGY, TEMPERATURE, AND PRESSURE DATA FIGURES

Appendix B contains time plots of the reactor power, the energy released, the maximum recorded fuel plate surface temperature, and the pressure generated for self-limiting power excursion tests that were performed as part of the experimental program conducted with the Spert IV D-12/25 core.

The power data were taken from the recording oscillograph traces of ion chambers whose current-to-power relationships had been determined previously by calorimetric methods.

The energy data were obtained by integrating the power data.

The temperature data were taken using 0.003-inch-thick chromel-alumel thermocouples resistance-welded to the fuel plate surfaces. The temperature traces shown in the figures generally are those for the hottest measured temperature for each test.

The pressure data were taken using unbonded-strain-gauge, diaphragm-type transducers mounted in the flow transition piece below the core. Pressure data is given for only the tests having a measurable pressure pulse.

The tests are presented in groups with respect to the initial conditions in the following order: 18-foot hydrostatic head with no forced flow, 2-foot hydrostatic head with no forced flow, 18-foot hydrostatic head with 500 gpm forced flow, 18-foot hydrostatic head with 1000 gpm forced flow, 18-foot hydrostatic head with 2500 gpm forced flow, and 18-foot hydrostatic head with 5000 gpm forced flow. The tests are presented in order of decreasing asymptotic period within the groups.

A summary of the data which were obtained from the Spert IV power excursion tests is presented in Table B-1.

TABLE B-I

DATA SUMMARY FROM SPERT IV D-12/25 CORE TRANSIENT TESTS

Figure Number	R ₁ (\$)	τ (msec)	α (sec ⁻¹)	$\phi(t_m)$ (MW)	E(t _m) (MW-sec)	P max (psig)	$\theta(t_m)$ (°C)	$\theta(\max)$ (°C)	Head (ft)	Flow (gpm)
B-1	0.80	980	1.02	0.77	2.18	0	71	93	18	0
B-2	0.88	598	1.67	1.12	1.56	0	68	103	18	0
B-3	0.92	374	2.67	1.55	1.27	0	68	113	18	0
B-4	0.98	190	5.26	3.32	1.08	0	63	121	18	0
B-5	1.04	107	9.35	6.9	1.18	0	73	122	18	0
B-6	1.11	62.4	16.0	17.5	1.64	0	99	128	18	0
B-7	1.15	49.8	20.1	25.1	1.83	0	113	124	18	0
B-8	1.26	29.6	33.8	68.5	3.00	0	136	141	18	0
B-9	1.40	19.6	51.0	147	3.23	3.7	148	162	18	0
B-10	1.53	15.8	66.3	220	4.40	8	160	165	18	0
B-11	1.54	14.5	69.0	247	4.40	---	165	178	18	0
B-12	1.68	11.7	85.5	350	4.84	15	175	179	18	0
B-13	1.82	10.1	99.0	467	4.94	16	183	191	18	0
B-14	1.91	8.48	118	615	5.66	18	170	212	18	0
B-15	2.00	7.8	128	705	6.35	19	167	270	18	0
B-16	2.14	7.0	143	875	8.50	21	170	338	18	0
B-17	1.40	21.3	47.0	145	3.49	3.3	131	138	2	0
B-18	1.68	12.0	83.4	340	4.43	12	155	156	2	0
B-19	1.91	8.8	113	570	5.18	18	165	184	2	0
B-20	0.88	560	1.79	(a)	(a)	0	(a)	97	18	500
B-21	1.14	54.2	18.4	21.5	2.32	0	103	125	18	500
B-22	1.66	11.9	84.0	322	5.26	14	163	169	18	500
B-23	1.80	10.1	99.0	435	5.97	18	165	173	18	500
B-24	0.00	529	1.89	(a)	(a)	0	(a)	99	18	1000
B-25	1.04	103	9.60	7.5	1.30	0	64	110	18	1000
B-26	1.14	54.5	18.3	22.5	2.35	0	124	136	18	1000
B-27	1.39	20.0	50.0	130	4.94	3	138	144	18	1000
B-28	1.66	12.3	81.3	292	5.49	14	160	168	18	1000
B-29	1.80	10.4	96.1	425	6.05	18	167	184	18	1000
B-30	0.88	531	1.89	(a)	(a)	0	(a)	105	18	2500
B-31	1.14	52.1	19.2	22.5	2.51	0	91	110	18	2500
B-32	1.66	12.2	82.0	320	5.41	14	169	176	18	2500
B-33	1.80	10.0	100	435	6.13	18	182	182	18	2500
----	0.87	554	1.81	(a)	(a)	0	(a)	149 (b)	18	5000
B-34	0.88	516	1.94	(a)	(a)	0	(a)	93	18	5000
B-35	1.05	104	9.62	9.8	1.79	0	58	102	18	5000
B-36	1.14	49.3	20.3	30.2	2.71	0	90	108	18	5000
---	1.14	47.2	21.2	31.5	---	0	---	129 (b)	18	5000
B-37	1.39	20.7	48.3	169	4.78	4.0	149	152	18	5000
---	1.39	20.2	49.5	173	---	4.0	---	167 (b)	18	5000
---	1.53	14.3	70.0	267	---	10	---	177 (b)	18	5000

TABLE B-I (Cont.)

DATA SUMMARY FROM SPERT IV D-12/25 CORE TRANSIENT TESTS

B-38	1.66	12.2	82.0	370	5.26	14	167	173	18	5000
---	1.66	11.5	87.0	373	---	15	---	184 (b)	18	5000
B-39	1.80	10.1	99.0	505	6.08	17	185	192	18	5000
---	1.80	10.0	100	481	---	23	---	200 (b)	18	5000
---	1.80	9.7	103	497	---	19	---	270 (b)	18	5000

R_i $\equiv$ Initial reactivity step (from control rod worth curve)

τ $\equiv$ Reactor period

α $\equiv$ $1/\tau$

$\phi(t_m)$ $\equiv$ Peak power

$\theta(t_m)$ $\equiv$ Fuel plate surface temperature at time of peak power

$\theta_{\max}$ $\equiv$ Maximum recorded fuel plate surface temperature

$P_{\max}$ $\equiv$ Maximum transient pressure

$E(t_m)$ $\equiv$ Energy release at time of peak power

(a) - Power exhibited no peak

(b) - All maximum temperature data for these tests are for narrow channels.

Fig. B-1 Test 2, 980-msec period.

Fig. B-2 Test 3, 598-msec period.

Fig. B-3 Test 5, 374-msec period.

Fig. B-4 Test 4, 190-msec period.

Fig. B-5 Test 6, 107-msec period.

Fig. B-6 Test 9, 62.4-msec period.

Fig. B-7 Test 7, 49.8-msec period.

Fig. B-8 Test 8, 29.6-msec period.

Fig. B-9 Test 10, 19.6-msec period..

Fig. B-10 Test 49, 15.8-msec period.

Fig. B-11 Test 11, 14.5-msec period.

Fig. B-12 Test 12, 11.7-msec period.

Fig. B-13 Test 13, 10.1-msec period.

Fig. B-14 Test 14, 8.48-msec period.

Fig. B-15 Test 52, 7.83-msec period.

Fig. B-16 Test 53, 7.0-msec period.

Fig. B-17 Test 15, 21.3-msec period.

Fig. B-18 Test 16, 12.0-msec period.

Fig. B-19 Test 17, 8.83-msec period.

Fig. B-20 Test 34, 560-msec period.

Fig. B-21 Test 38, 54.2-msec period.

Fig. B-22 Test 42, 11.9-msec period.

Fig. B-23 Test 46, 10.1-msec period.

Fig. B-24 Test 35, 529-msec period.

Fig. B-25 Test 37, 103-msec period.

Fig. B-26 Test 39, 54.5-msec period.

Fig. B-27 Test 41, 20.0-msec period.

Fig. B-28 Test 43, 12.3-msec period.

Fig. B-29 Test 45, 10.4-msec period.

Fig. B-30 Test 26, 531-msec period.

Fig. B-31 Test 40, 52.1-msec period.

Fig. B-32 Test 44, 12.2-msec period.

Fig. B-33 Test 47, 10-msec period.

Fig. B-34 Test 18, 516-msec period.

Fig. B-35 Test 19, 104-msec period.

Fig. B-36 Test 20, 49.3-msec period.

Fig. B-39 Test 23, 10.1-msec period.

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