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REACTOR STABILITY IN THE SPERT IV FACILITY

J. G. Crocker, Z. R. Martinson,
R. M. Potenza, and L. A. Stephan

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**PHILLIPS
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ATOMIC ENERGY DIVISION

**NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION**

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ABSTRACT

A series of tests to investigate the stability characteristics of a highly enriched, water-moderated, plate-type core has been performed in the Spert IV facility. Most of the tests were initiated at atmospheric pressure, ambient ($\approx 20^{\circ}\text{C}$) bulk water temperature, and with only natural water circulation through the core. The results of these tests showed that, following rampwise reactivity insertions up to 5.0\$ (the maximum used) at an average rate of about 0.15\$/sec, the core under investigation would not become unstable as long as the inlet water temperature to the core remained near ambient. One test was performed in which the bulk water temperature of the system was increased by operation of the reactor at significant power levels with no heat removal equipment in operation. In this test, 4.0\$ of reactivity was inserted in a ramp, and the reactor became unstable when the inlet water temperature reached approximately 70°C after about 12.5 minutes of reactor operation. Upon becoming unstable, the reactor power exhibited successive power oscillations with peaks of 7.5, 15, 25, and 50 MW with a frequency of two cycles per second. A few tests were performed with 1.2 ft/sec forced coolant circulation through the core under ambient bulk water conditions. Under these conditions, the reactor was not made unstable because potential fuel plate burnout problems were encountered with reactivity insertions which were insufficient for obtaining divergent power oscillations. From the various tests which were performed, the rms value of the reactor power noise, spectral power densities, and spectral moment frequencies were determined in order to evaluate the usefulness of these quantities in determining reactor stability characteristics. The results showed that these quantities can be of value in predicting the onset of reactor instability. The experimental and analytical results obtained from the Spert IV stability tests are presented in this report.

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I. INTRODUCTION

Reactor stability has been a subject of considerable interest to reactor designers, operators and others concerned with the safety of nuclear reactors. A large fraction of this interest can be attributed to stability experiments conducted in Borax I [1] and Spert I [2] which demonstrated that highly enriched, plate-type, water-moderated cores can sustain rapid, large-amplitude power oscillations with a potential of growing to destructive proportions.

In the tests performed in Spert I during 1956 and 1957, large amounts of excess reactivity were added to the system with the reactor initially critical at a few watts, with the bulk water temperature either at ambient ($\approx 20^\circ\text{C}$) or at boiling ($\approx 96^\circ\text{C}$) and with hydrostatic heads in the range from two to nine feet above the core. The following conclusions were drawn from these tests: (a) 2\$ of reactivity held in voids constituted a possibly unstable situation for the core under all the system conditions which were investigated; (b) the stability of the core was determined to be quite sensitive to the hydrostatic head above the core, particularly in the tests initiated at boiling; and (c) from the rapid rise of the power during the oscillations, it was concluded that low-pressure, boiling water reactors have an intrinsic mechanism for the rapid return of large amounts of reactivity to the system.

Although contributing significantly to the available information on reactor stability, the tests in Spert I were limited in scope because the design of the Spert I facility [3] did not permit adequate control of the variables which were determined to have an influence on the stability characteristics of the reactor. In particular, the system could not be maintained at a constant subcooled temperature due to the small heat capacity of the system and the lack of heat removal equipment. Also, during some of the tests, water was thrown from the reactor vessel thereby compromising the investigation of the effect of the hydrostatic head. Because of these limitations, a comprehensive study of reactor stability was deferred pending completion of a facility (Spert IV) better suited for stability tests.

One of the objectives in the design of Spert IV was to provide a facility for the continuation of the Spert I reactor stability studies under more closely controlled environmental conditions. Basically, Spert IV is a large-pool facility composed of two connected 20-ft-diameter by 25-ft-deep tanks with provisions for forced coolant circulation at rates up to 5000 gpm and a 1 MW steady state heat removal capacity. The facility was designed for a maximum operating bulk pool temperature of 55°C .

A highly enriched, plate-type core, the D-12/25 core, was chosen for the stability tests in Spert IV. This core was composed of 20 fuel assemblies, 4 control rod assemblies, and one transient rod assembly in a 5 by 5 array. The fuel assemblies each had 12 fuel plates, and the rodded assemblies each had 6 fuel plates. Each fuel plate contained 14 g of U-235. With this core installed in the Spert IV facility, tests could be performed with up to an 18-ft hydrostatic head above the core and with a maximum flow rate of 12 ft/sec through the core. Prior to the performance of the stability tests with this core, a program of static measurements [4] and power excursion tests [5] was performed.

Tests in the Spert IV stability program were performed (a) to determine the necessary conditions required for obtaining divergent power oscillations (the threshold of instability) with various hydrostatic heads above the core, (b) to evaluate the applicability and validity of various experimental techniques for predicting the threshold of instability, and (c) to observe the behavior of the reactor when the threshold of instability has been exceeded. The following types of tests were performed in this program.

- (1) Threshold Tests -- In these tests, reactivity up to 4.2\$ was added in small increments to the core until oscillations of approximately ± 50 percent of the mean power level were observed. Data from these tests were used to evaluate the use of various noise analysis techniques in predicting the threshold of instability.
- (2) Ringing Tests -- In these tests, stepwise insertions of reactivity were made with the reactor initially operating at high power in order to induce a damped oscillatory behavior in the reactor power. In theory, the damping factors obtained from such tests can be extrapolated to zero at the threshold of instability.
- (3) Chugging Tests -- In these tests, large amounts of excess reactivity (up to 5\$) were inserted in ramp-fashion in order to induce large-amplitude, divergent power oscillations. Most of the tests were performed with the bulk pool temperature at ambient, with either a 2- or an 18-ft hydrostatic head, and with only natural circulation through the core.

It was determined in this program that the stability of the reactor was strongly dependent on the inlet water temperature to the core and that, for reactivity insertions up to 5\$, this temperature must be considerably above room temperature in order for the reactor to become unstable. Most of the tests, however, were performed with the inlet water temperature at ambient, this temperature not being a controlled variable at higher temperatures in Spert IV. In the only test in which large-amplitude, divergent power oscillations were obtained, 4\$ of reactivity was inserted as a ramp, and the inlet water temperature increased to approximately 70°C before the reactor became unstable. This report describes the results of the stability tests which have been performed in the Spert IV facility and the conclusions which have been drawn therefrom.

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 tanks, 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-ft-diameter by 25-ft-deep tanks which are connected at the top by a 6- by 6-ft removable gate. The tanks are designed for 25 feet of hydrostatic head plus a 50-psi overpressure. 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 6.

Fig. 1 Spert IV reactor tanks.

2. DESCRIPTION OF REACTOR CORE COMPONENTS

The Spert IV D-12/25 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 by 5 array [a]. Control of the reactor is 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 normally 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.

[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 [7] in the Spert I facility.

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

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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 percent enriched U-Al alloy
Cladding	Al (6061) alloy
U-235 content of plate	14.0 ± 0.2 g
	Control Rods
Number	
Control	4 gang-operated
Transient	1
Composition	"Binal" (7 weight percent boron-aluminum alloy)
Total rod worth	20\$
Withdrawal rate	0 to 12 in./min
Scram time	Approximately 300 msec from upper limit

2.1 Fuel Assembly

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

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 alloy 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

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

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

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 7 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 into guide slots in the standard fuel assemblies. Each rodged assembly has six fuel plates instead of the normal twelve plates.

The transient rod for initiating step reactivity injections 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 into the core to decrease reactivity and dropped out of the core to increase the reactivity of the core.

3. SUMMARY OF STATIC NUCLEAR CHARACTERISTICS OF THE D-12/25 CORE

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. Table II is a brief summary of the static characteristics which were measured for the Spert IV D-12/25 core. For all these measurements, the core was suspended in a 7-ft-diameter by 8-ft-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 measurements can be found in Reference 4.

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

TABLE II

STATIC CHARACTERISTICS OF 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¢/°C
35°C	- 1.2¢/°C
Uniform void coefficient of reactivity	-41.5¢/percent decrease in moderator density
Maximum measured local void coefficient	-60¢/percent decrease in moderator density

4. POWER EXCURSION TESTS WITH THE D-12/25 CORE

A series of self-limiting reactor power excursion tests initiated by stepwise insertions of reactivity was performed in the Spert IV facility [5]. 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 performed with the system at atmospheric pressure and ambient temperature and with forced flow rates of 0 to 12 ft/sec upward through the core. For most of the tests, there was an 18-ft hydrostatic head above the core, although a few short-period tests with no forced coolant circulation were performed with a 2-ft hydrostatic head to investigate the effect of that variable.

A fiducial series of tests was performed with no forced coolant circulation and with an 18-ft 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-ft 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-ft-head tests were slightly lower than for the 18-ft-head tests. The shortest period test which was performed with a 2-ft 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 elimination of the 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 had a thickness which is one-half that of the normal channels, these channels had lower flow rates than the standard channels, and departure from nucleate boiling occurred at substantially lower powers in these channels causing abnormally high temperatures for adjacent fuel plates. This effect was not particularly significant during power excursions since relatively little energy was released during the excursion, but became extremely important for tests with long run-out times and for steady high-power operation.

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 [8].

The data summary for the transient tests performed with the Spert IV D-12/25 core is presented in Table III.

TABLE III

DATA SUMMARY FROM TRANSIENT TESTS

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)
0.80	980	1.02	0.77	2.18	0	71	93	18	0
0.88	598	1.67	1.12	1.56	0	68	103	18	0
0.92	374	2.67	1.55	1.27	0	68	113	18	0
0.98	190	5.26	3.32	1.08	0	63	121	18	0
1.04	107	9.35	6.9	1.18	0	73	122	18	0
1.11	62.4	16.0	17.5	1.64	0	99	128	18	0
1.15	49.8	20.1	25.1	1.83	0	113	124	18	0
1.26	29.6	33.8	68.5	3.00	0	136	141	18	0
1.40	19.6	51.0	147	3.23	3.7	148	162	18	0
1.53	15.8	66.3	220	4.40	8	160	165	18	0
1.54	14.5	69.0	247	4.40	----	165	178	18	0
1.68	11.7	85.5	350	4.84	15	175	179	18	0
1.82	10.1	99.0	467	4.94	16	183	191	18	0
1.91	8.48	118	615	5.66	18	170	212	18	0
2.00	7.8	128	705	6.35	19	167	270	18	0
2.14	7.0	143	875	8.50	21	170	338	18	0
1.40	21.3	47.0	145	3.49	3.3	131	138	2	0
1.68	12.0	83.4	340	4.43	12	155	156	2	0
1.91	8.8	113	570	5.18	18	165	184	2	0
0.88	560	1.79	(a)	(a)	0	(a)	97	18	500
1.14	54.2	18.4	21.5	2.32	0	103	125	18	500
1.66	11.9	84.0	322	5.26	14	163	169	18	500
1.80	10.1	99.0	435	5.97	18	165	173	18	500
0.88	529	1.89	(a)	(a)	0	(a)	99	18	1000
1.04	103	9.60	7.5	1.30	0	64	110	18	1000
1.14	54.5	18.3	22.5	2.35	0	124	136	18	1000
1.39	20.0	50.0	130	4.94	3	138	144	18	1000
1.66	12.3	81.3	292	5.49	14	160	168	18	1000
1.80	10.4	96.1	425	6.05	18	167	184	18	1000
0.88	531	1.89	(a)	(a)	0	(a)	105	18	2500
1.14	52.1	19.2	22.5	2.51	0	91	110	18	2500
1.66	12.2	82.0	320	5.41	14	169	176	18	2500
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
0.88	516	1.94	(a)	(a)	0	(a)	93	18	5000
1.05	104	9.62	9.8	1.79	0	58	102	18	5000
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

TABLE III (Cont.)

DATA SUMMARY FROM TRANSIENT TESTS

R_i (β)	τ (msec)	α (sec^{-1})	$\phi(t_m)$ (MW)	$E(t_m)$ (MW-sec)	$P(\text{max})$ (psig)	$\theta(t_m)$ ($^{\circ}\text{C}$)	$\theta(\text{max})$ ($^{\circ}\text{C}$)	Head (ft)	Flow (gpm)
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
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
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 above delayed critical (from control rod worth curve)

$\tau \equiv$ Reactor period

$\alpha \equiv 1/\tau$

$\phi(t_m) \equiv$ Peak power

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

$\theta(\text{max}) \equiv$ Maximum recorded fuel plate surface temperature

$P(\text{max}) \equiv$ Maximum transient pressure

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

(a) $\equiv$ Power exhibited no peak

(b) $\equiv$ All maximum temperature data for these tests are for narrow channels.

III. SPERT IV STABILITY TESTS

1. GENERAL DESCRIPTION OF TESTS

In the reactor stability program with the Spert IV D-12/25 core, tests were performed to (a) determine the threshold of instability for the core, (b) evaluate the applicability and validity of various experimental techniques for predicting the threshold of instability, and (c) observe the behavior of the reactor once the threshold of instability is exceeded. To accomplish the objectives of the program, three different types of tests were performed: threshold tests, ringing tests, and chugging tests. In all three categories, large amounts of reactivity were inserted into the reactor [a], the principal distinction between test categories being the method of reactivity insertion. In the threshold tests, the reactivity was inserted gradually in small increments; in the ringing tests, small stepwise insertions of reactivity were made with large amounts of reactivity initially compensated by the reactor; and in the chugging tests, the reactivity was inserted in a ramp.

2. THRESHOLD TESTS

The purpose of these tests was to determine the reactivity and/or power level conditions required to make the reactor unstable and to obtain data for noise analysis purposes. It was not the purpose of these tests to allow the reactor to become unstable; therefore, the reactivity was inserted in small predetermined increments, and the reactor was operated at each resulting steady-state power level for several minutes to observe for possible instabilities and for the acquisition of data. In order to set a practical limit for performing the tests without undue risk to the core, an operational limit was defined as that for which power oscillations of ± 50 percent about the mean power level were obtained. This somewhat arbitrary definition is based on the assumption that the reactor conditions required to give ± 50 percent amplitude oscillations are nearly the same as those giving rise to divergent power oscillation behavior. At each power level, an approximate value for the mean amplitude of power oscillation was determined, and extrapolations of these amplitudes were made to the 50 percent amplitude threshold condition. Power data from these tests were recorded on magnetic tape for use in various noise-analysis methods for predicting reactor instability.

Threshold tests were performed in Spert IV for the following initial system conditions: (a) ambient temperature, 18-foot hydrostatic head, no forced coolant circulation; (b) ambient temperature, 2-foot hydrostatic head, no forced coolant circulation; (c) ambient temperature, 18-foot hydrostatic head, 1.2 ft/sec (500 gpm) forced coolant circulation through the core.

2.1 18-Foot Hydrostatic Head, Natural-Circulation Tests

2.11 Reactor Behavior. To determine the threshold of instability of the Spert IV core under system conditions of an 18-foot hydrostatic head and no

[a] Unless otherwise specified, all reactivity data given in this report are values in excess of that required for low-power delayed critical.

forced coolant circulation, numerous tests were performed with total reactivity insertions ranging from 0.2\$ to 4.2\$ above low-power critical. At various reactivity levels, the reactor power (which was displayed on oscilloscopes and examined continuously for evidence of approaching instabilities) was recorded on magnetic tape for about five minutes after the mean power had reached an approximate equilibrium level. Figures 5, 6, and 7 show representative samples of the reactor power for various total reactivity insertions from 0.4\$ to 4.2\$. It can be seen in these figures that the reactor power began to exhibit moderate oscillatory behavior with a total reactivity insertion of 1.8\$. As indicated by closed-circuit television and measured core temperatures, a reactivity insertion of 1.8\$ also marked the beginning of moderate steam generation in the core. The amplitude of the power oscillations increased with increasing reactivity, the operational limit of ± 50 percent oscillations about the mean power level being attained with a minimum total reactivity insertion of 4.1\$.

Figure 8 shows a representative portion of the power trace from the 4.1\$ test, together with the data on fuel plate surface temperatures obtained from a standard-width (0.180-inch) coolant channel and from a narrow-width (0.090-inch) coolant channel [a]. In this test, the mean power level was approximately 7.5 MW, and the power was characterized by oscillations with a frequency of approximately 4 cps. The amplitude (one-half peak-to-peak) of the power oscillations varied somewhat, reaching values of roughly ± 60 percent, but showing no particular tendency to diverge. The noisy temperature traces from the surface thermocouples indicate that considerable moderator boiling was occurring. It is of interest to note the difference in the frequencies of the temperature oscillations obtained from the narrow- and standard-width channels. The fundamental frequency in the standard-channel temperature trace was approximately 1/2 cps; while that for the narrow-channel temperature trace was about 3 cps, or very nearly the same as that of the power trace, implying a strong interaction between the narrow-channel temperature behavior and the reactor power behavior.

Additional temperature data from water-channel thermocouples located between fuel plates in a standard-width coolant channel at +12, +3, and -12 inches (distances above or below the centerline of the active fuel region) are shown in Figure 9. In this figure, it is seen that the inlet water temperature measured by the thermocouple at -12 inches had risen to an average of about 35°C, with occasional temperature spikes up to approximately 90°C. The temperature traces from the coolant channel thermocouples located at +3 and +12 inches indicate that the coolant temperature was fluctuating up to the saturation temperature of 109°C with oscillation amplitudes of about 50°C at the 3-inch position and about 30°C at the outlet of this channel, with a frequency of about 1/2 cps.

In the 4.2\$ reactivity insertion test, the reactor power also developed oscillations greater than ± 50 percent (the operational limit set for these tests); and the reactor was manually scrammed after about 30 sec of operation.

[a] In the temperature traces which are shown in the report, the fuel plate (and water-channel) 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 (or in the water channel next to) the west side of fuel plate number seven, three inches below the core centerline.

Fig. 5 Powertraces for 0.4-, 0.8-, 1.4-, 1.8-, 2.1-, and 2.5-\$ total reactivity insertions in Spert IV.

Fig. 6 Power traces for 3.0- and 3.5-\$ total reactivity insertions in Spert IV.

Fig. 7 Power trace for a 4.2-\$ total reactivity insertion in Spert IV.

Fig. 8 Reactor power and cladding surface temperature behavior in a narrow and a standard coolant channel during an 18-foot hydrostatic head, natural-circulation threshold test (No. SS 8).

Fig. 9 Coolant temperature behavior at three elevations in a standard coolant channel during test No. SS 8.

2.12 Noise Analysis of 18-Foot Hydrostatic Head, Natural-Circulation Tests.

To determine the usefulness of various noise analysis techniques in predicting reactor instability, spectral power densities and the rms values of the reactor power noise were determined for the threshold tests performed with an 18-ft hydrostatic head. Appendix A contains details on the calculation and use of the quantities.

Spectral power density plots of the reactor power from some of the 18-ft head threshold tests with total inserted reactivities ranging from 0.4 to 4.1\$ are shown in Figure 10. (The separation between the curves is arbitrary since the absolute gain of the spectral power density amplitude was not measured.) It can be seen in this figure that the spectral power density function changed markedly as the reactivity was increased. For reactivities less than 3.0\$, various small-amplitude resonances in the power spectrum are apparent; however, no relationship between these resonances and the stability of the reactor could be determined. In the 3.5\$ and 4.1\$ tests, a large and fairly sharp resonance peak in the power spectrum developed at about 4 cps, the same frequency that was obviously dominant in the power oscillations for these tests. The higher and sharper such resonances appear in the power spectrum, the greater the tendency of the system is toward unstable oscillations at the resonant frequency. Because the inlet water temperature was not a controlled variable during these tests, however, reliable instability predictions were not obtained from the spectral analysis of these threshold tests.

Akcasu [9] has pointed out that the rms value of the reactor noise may be used to predict the onset of instability in a noise excited system. The ratio

Fig. 10 Spectral power density as a function of frequency of 0.4-, 1.4-, 2.1-, 2.5-, 3.0-, 3.5-, and 4.1- $\$$ total reactivity insertion threshold tests.

of $\frac{\text{average power}}{\text{rms value}}, \frac{\bar{\phi}}{\Delta\phi_{\text{rms}}}$, can be plotted as a function of some parameter such as power, reactivity, pressure, or temperature. Extrapolation of this curve to the point where the value of $\frac{\bar{\phi}}{\Delta\phi_{\text{rms}}}$

approaches zero indicates the threshold of instability. Figure 11 is a plot of the average power level divided by the rms value of the power noise as a function of total reactivity insertion for the 18-ft hydrostatic head threshold tests. As noted previously, the inlet water temperature remained near ambient for the duration of these tests. Using a linear extrapolation, it can be seen in this figure that for inlet water temperatures near ambient, reactivity insertions greater than 5.3\$ (maximum available excess reactivity) would be required for divergent power behavior of the Spert IV D-12/25 core.

Fig. 11 Average power divided by rms value of reactor power noise as a function of total inserted reactivity for 18-foot hydrostatic head, natural-circulation threshold tests.

2.2 2-Foot Hydrostatic Head, Natural-Circulation Tests

Since tests previously performed in Spert I with the A-17/28 core [2] indicated that the hydrostatic head above the core had an influence on the stability characteristics of the reactor, the hydrostatic head above the D-12/25 core was reduced to two feet, and two threshold tests were performed to determine the conditions required to give ± 50 percent oscillations about the mean power level. In these two tests, the reactor was operated with total reactivity insertions of 0.4, 0.6, 1.0, 1.5, 1.9, 2.8, 3.5, and 4.1\$ above low power critical.

With a 2-ft hydrostatic head and a total reactivity insertion of 4.1\$, the reactor power displayed a definite tendency toward 2-cps divergent oscillations. Figure 12 shows reactor power and fuel plate surface temperature data obtained during this test. The amplitude of the peak oscillation which occurred just prior to the time of reactor scram was about 110 percent of the mean power level of about 3 MW. The mean power reached a maximum level of approximately 5 MW at a time when 1.8\$ of reactivity had been added; but as a result of temperature feedback effects, the power level had decreased to about 3 MW by the time the predetermined reactivity insertion of 4.1\$ had been accomplished.

It can be seen in Figure 12 that before the large power oscillations occurred, the fuel plate surface temperature traces indicate that uniform nucleate boiling was taking place. The fuel plate surface temperatures then began to oscillate with the large power spikes, indicating that a direct coupling between power and temperature had been accomplished, which was not evident in the 18-ft hydrostatic head test.

Figure 13 shows the water-channel temperature traces obtained during the 2-ft hydrostatic head test from the same thermocouple locations indicated in

Fig. 12 Reactor power and cladding surface temperature behavior in a narrow and a standard coolant channel during a 2-foot hydrostatic head, natural circulation threshold test (No. SS 10).

Fig. 13 Coolant temperature behavior at three elevations in a standard coolant channel during test No. SS 10.

Figure 9 for the 18-ft hydrostatic head test. Note that the inlet water temperature indicated by the thermocouple located at -12 inches had risen to about 70°C by the time the large-amplitude oscillations occurred. The temperature trace from the coolant channel thermocouple located at +3 inches has small amplitude oscillations up to the saturation temperature of 96°C with a frequency of about 1/2 cps, while the trace from the thermocouple located at +12 inches indicates that nearly uniform boiling is occurring at the outlet of this channel.

Although the threshold of instability as defined for these tests was reached with a total reactivity insertion of 4.1\$ for both the 18- and 2-ft hydrostatic head tests, the differences in inlet water temperature preclude establishing specific conclusions with regard to the effect of the hydrostatic head above the core on reactor stability.

2.3 18-Foot Hydrostatic Head, 1.2 ft/sec Flow Tests

The stability characteristics of the Spert IV core under system conditions of an 18-ft hydrostatic head and a 1.2 ft/sec forced coolant flow rate were investigated in three threshold tests involving reactivity insertions up to 1.6\$. Under these conditions, power oscillation amplitudes approaching the ± 50 percent operational limit were not attained because the fuel plate surface temperatures in various narrow-width channels exceeded the predetermined scram temperature level of 300°C before sufficient reactivity could be added to the system to yield the threshold condition. During the third flow test, for example, at a time when the mean power level was approximately 10.5 MW (following a total reactivity insertion of 1.6\$), the indicated temperature on one of the monitored narrow-channel fuel plates was observed to increase from approximately 140 to 385°C, resulting in a reactor scram. The power behavior during the test, however, was relatively stable, being characterized by fluctuations of approximately ± 10 percent about the mean level with occasional spikes of ± 15 percent. Figure 14 shows the power trace and the monitored narrow-channel temperature signal obtained in this test.

The hot-channel problems encountered in these flow tests arise as a consequence of the reduced flow in the narrow coolant channels adjacent to the two outside fuel plates in each of the fuel, control-rod and transient-rod assemblies of the core. The reduced flow allows these channels to depart from nucleate boiling at lower power levels than those required for the rest of the core. It was concluded from these tests that, with forced flow, fuel plate melting would occur before the threshold of instability could be reached. Therefore, no further stability tests with flow were performed.

2.4 Threshold Tests Data Summary

Table IV is a data summary from the threshold tests which were performed with the D-12/25 core.

Fig. 14 Reactor power and cladding surface temperature behavior in a narrow coolant channel during an 18-foot hydrostatic head, 1.2-ft/sec forced-flow threshold test (No. SS 13).

TABLE IV

DATA SUMMARY FROM THRESHOLD TESTS

18-Foot Hydrostatic Head, Natural-Circulation Tests						
Stability Test Number	Reactivity (β)	Mean Power (MW)	RMS of Reactor Noise (MW)	Initial Inlet Water Temperature (°C)	Inlet Water Temperature at End of Test (°C)	Remarks
1	0.25 0.35 0.53 0.64 0.79	0.16 0.34 0.67 1.01 1.69		25.1	27.1	
3	0.45 0.61 0.80 1.1 1.2 1.3 1.4	0.51 0.99 1.98 3.48 3.99 4.51 5.01	0.147	25.2	28.7	
5	0.61 1.6 1.8 2.7	0.99 5.63 5.92 7.93	0.694	25.6	31.6	
7	0.62 0.94 1.6 2.1 2.9 3.2 4.0	0.99 3.00 5.34 6.30 7.37 7.86 7.76	0.721 0.778	23.0	38.6	
8	1.6 2.9 3.7 4.1	5.43 7.37 7.25 7.76	0.889	24.0	35.0	Reactor scrammed when power oscillations exceeded ± 50 percent
57	0.40 0.80	0.51 2.05	0.004 0.008	24.4	24.4	
14	1.1	4.01	0.075			
59	1.4	5.86	0.096	24.5	26.5	
53	1.8	5.90	0.226	24.1	26.1	
54	2.1	6.90	0.451	23.3	26.3	
55	2.5	7.37	0.530	22.9	25.9	
60	3.0	9.56	0.774	24.4	27.4	
58	3.5	9.37	0.956	24.2	27.2	
56	4.2	10.2	1.37	24.1	27.1	Reactor scrammed when power oscillations exceeded $\pm 50\%$ after ≈ 30 sec at 4.2%
67	0.50 1.0 1.5 2.0 2.5 2.8	0.6 2.9 5.0 6.4 7.4 7.7		25.1	28.1	

TABLE IV (Cont.)

DATA SUMMARY FROM THRESHOLD TESTS

2-Foot Hydrostatic Head, Natural-Circulation Tests					
Stability Test Number	Reactivity (β)	Mean Power (MW)	Initial Inlet Water Temperature (°C)	Inlet Water Temperature at End of Test (°C)	Remarks
9	0.44	0.54	23.1	29.1	
	0.60	0.97			
	0.95	2.92			
	1.5	4.65			
10	0.61	0.97	22.0	71	
	1.0	2.93			
	1.5	4.70			
	1.9	5.33			
	2.8	5.04			
	3.5	4.17			
	4.1	2.91			
86	3.0	2.4	21.6	70.6	Reactor scrammed when power oscillations exceeded $\pm$ 50 percent
	3.5	2.0			
18-Foot Hydrostatic Head, 1.2 ft/sec Flow Tests					
11	0.04	0.97	20.2	31.2	
	0.13	2.91			
	0.24	4.40			
	0.27	5.49			
	0.37	6.50			
	0.50	7.57			
	0.89	8.62			
	1.1	9.70			
	1.5	10.3			
12	0.08	0.97	18.5	19.0	
	0.51	6.50			
	0.67	7.57			
	0.90	8.73			
	1.2	9.70			
13	0.08	0.97	17.0	19.0	Reactor scrammed due to excessive fuel plate surface temperature
	0.25	2.89			
	0.37	5.47			
	0.42	6.50			
	0.56	7.56			
	0.80	8.82			
	1.0	9.70			
	1.1	10.2			
	1.3	10.8			
	1.6	11.4			

3. RINGING TESTS

Fourteen ringing tests were performed during the Spert IV stability program in which, following the attainment of an equilibrium power level condition for a given initial reactivity insertion, an additional stepwise reactivity insertion (accomplished by rapid ejection of the transient rod) was made in an effort to induce an oscillatory power behavior. In theory [10], if a system is near the limit of stability, the power response resulting from a step-reactivity perturbation will take the form of damped oscillations of constant frequency corresponding to the principal resonance of the system transfer function; therefore, the damping characteristics of the oscillatory behavior should provide a quantitative measure of the stability of the system. All ringing tests were performed with an 18-ft hydrostatic head above the core, with an inlet water temperature of approximately 28°C prevailing at the time of the stepwise reactivity insertion, and with natural circulation flow. The Spert IV tests involved initial reactor-compensated reactivity insertions of up to 3.75\$ combined with stepwise insertions of 0.25, 0.5, or 1.0\$. The largest amount of total reactivity (initially compensated plus step insertion) achieved was 4.25\$. Table V is a summary of the data obtained from the ringing tests.

Figure 15 shows the typical power behavior during a ringing test. The equilibrium power level attained before the stepwise insertion of reactivity displayed a generally oscillatory behavior due to moderator boiling. The oscillations were superimposed on a mean power level ranging from 0.66 MW (for 0.5\$ initial reactivity insertion) to 9.6 MW (for 3.75\$ initial reactivity insertion), the amplitude of the oscillations increasing with the mean power level.

The peak powers resulting from the stepwise reactivity insertions generally fell into two categories. The 0.5\$ ringing tests displayed peak powers approximately 4 MW above the initial equilibrium power levels. The 1.0\$ ringing tests

TABLE V
DATA SUMMARY FROM RINGING TESTS

Stability Test Number	18-Foot Hydrostatic Head, Natural-Circulation Tests							Inlet Water Temperature at End of Test (°C)
	Initial Reactivity (\$)	Reactivity of Step-Input (\$)	Total Reactivity (\$)	Initial Equilibrium Power Level (MW)	Peak Power (MW)	Final Equilibrium Power Level (MW)	Initial Inlet Water Temperature (°C)	
4	1.25	0.25	1.50	4.1	5.8	5.0	22.8	25.8
15	1.00	0.50	1.50	3.2	7.1	5.3	----	----
16	1.50	0.50	2.00	5.5	8.9	7.0	----	----
17	0.50	0.50	1.00	0.66	1.5	2.8	23.9	25.9
22	2.00	0.50	2.50	6.5	10.0	8.1	24.7	27.7
26	2.50	0.50	3.00	8.1	12.4	8.2	24.2	30.2
40	3.00	0.50	3.50	8.1	12.2	8.2	24.3	27.3
44	3.25	0.50	3.75	8.7	13.3	9.1	25.9	34.9
45	3.50	0.50	4.00	8.6	12.7	8.8	24.5	27.5
46	1.50	1.00	2.50	5.3	15.6	7.4	24.3	31.3
52	2.00	1.00	3.00	6.5	15.4	8.1	23.5	28.5
68	2.75	0.50	3.25	7.8	10.4	8.6	24.7	38.7
69	3.25	0.50	3.75	8.7	12.0	9.4	24.5	34.5
74	3.75	0.50	4.25	9.6	14.8	9.6	24.4	27.4

Fig. 15 Reactor power trace during a 4.0-\$ total reactivity insertion ringing test.

displayed peak powers approximately 9 to 10 MW above the initial equilibrium power levels. The only true exceptions to these behaviors were the 0.5\$ ringing test No. SS 17 where the initial equilibrium power level was about 90 percent lower than those of the remaining 0.5\$ tests and ringing test No. SS4 which does not come under the above categorization because the step reactivity insertion was only 0.25\$.

The reactor power after the first power peak returned immediately to a new equilibrium level usually no more than 2 MW higher than the initial equilibrium level. In some of the 0.5 and 1.0\$ step reactivity insertion tests, a secondary power peak was observed; but in no tests were any further power peaks discernible from the boiling noise in the power behavior. As in the tests performed on the VBWR [11], no damped oscillatory power behavior could be observed and only in the larger reactivity insertion tests were secondary power peaks observed. In all cases, the step reactivity induced power rises had peaked before all of the step inserted reactivity could be introduced into the reactor by the transient rod (the reactivity "steps" required about 300 msec).

The temperature behavior during the ringing tests was essentially identical to the temperature behavior during high-power, steady-state operation of the reactor (the fuel plate temperatures oscillating near saturation and the water channel temperatures oscillating between approximately 30°C and saturation). In both cases, the frequency of oscillation was about 1/2 cps. The only noticeable responses in the temperature to the step reactivity insertions occurred in the

fuel plate surface temperature for the larger reactivity insertions. For these cases, the fuel plate surface temperatures displayed an increase of approximately 10°C with the occurrence of the first power peak.

At first, it was thought that the general reactor power noise level was masking the desired damped oscillatory power response to the step reactivity insertion; however, comparison of the system parameters obtained during these ringing tests with those of subsequent chugging tests (especially test No. SS 89) indicates that the reactor was not near enough to being unstable in order for this effect to be observable because of the relatively low inlet water temperature.

4. CHUGGING TESTS

In these tests, large amounts of reactivity (3.0 to 5.0\$) were added to the system in ramp-fashion in an effort to induce divergent oscillations in the reactor power behavior. The name of these tests was derived from the large-amplitude power oscillation phenomenon observed in Borax I [1] and Spert I [2] which has been termed "chugging". The purpose of these tests was (a) to experimentally determine the instability threshold for divergent power behavior with various hydrostatic heads above the core and (b) to obtain data on the dynamic behavior of the core when the instability threshold has been exceeded.

All of the tests were initiated with the bulk pool temperature at ambient (approximately 22°C), with either an 18- or a 2-ft hydrostatic head above the core, and with only natural circulation through the core. In most of the tests, the inlet water temperature remained at near ambient.

The results of the chugging tests which were performed are discussed in the following sections.

4.1 18-Foot Hydrostatic Head, Natural-Circulation Chugging Tests

With an 18-ft hydrostatic head above the core, tests were performed in which reactivities of 3.0, 3.5, 4.0, 4.25, 4.5, 4.75, and 5.0\$ were inserted in ramp-fashion at average rates of about 0.15\$/sec with the reactor initially critical at a power level of about 1 W. In each test, the reactor was allowed to operate at the resulting equilibrium conditions for approximately 5 min in order to observe the stability characteristics of the reactor.

In each test, about 6 sec after the initiation of the reactivity insertion, a transient power burst was observed with a minimum period of 60 msec and with a peak value of approximately 30 MW. The characteristics of this initial peak were primarily functions of the initial reactor conditions and reactivity insertion rate and were not of particular interest in these tests. Following the initial peak, the power typically decreased to a minimum of 2 MW and subsequently increased to an equilibrium level such that the inserted reactivity was compensated by the equilibrium void content and density change in the moderator. Depending on the total reactivity insertion, the equilibrium power level varied from 7.5 to 10 MW.

Figure 16 shows the reactor power and inserted reactivity as a function of time for the 3.0\$ chugging test. The power behavior for this test is typical

Fig. 16 Reactor power and inserted reactivity as a function of time during 18-foot hydrostatic head, natural-circulation chugging test (No. SS 70).

of all the tests which were performed; the only significant difference in result between this test and larger inserted reactivity tests was an increase in the equilibrium power level. In none of the tests did the reactor power exhibit any tendency toward divergent oscillations. Instead, the power was characterized by fluctuations generally within 30 percent with occasional spikes up to 40 percent of the mean power level. The frequency of the fluctuations was approximately 10 cps. These high-frequency reactor power oscillations were probably associated with the rapid growth and collapse of bubbles in the large portion of the core in which nucleate boiling was occurring.

Figure 17 shows a representative sample of the reactor power along with typical surface temperatures from fuel plates adjacent to a standard (0.180-inch-thick) and a narrow (0.090-inch-thick) moderator-coolant channel from the 5.0\$ test. The temperature traces from both the water channels and the fuel plate surface thermocouples in the standard channels exhibited a dominant oscillation of approximately 0.4 cps. These temperature oscillations were probably caused by an oscillation of the natural circulation water velocity through the core. The temperatures from the narrow channels, however, exhibited an oscillation of approximately 3 cps. The character of the oscillations indicated that the occurrence of a transition type of boiling between nucleate and film boiling was associated with these oscillations. The characteristics of the temperature traces, shown in Figure 17, were generally prevalent for all the tests. As the average power increased slightly with increased total reactivity insertion, the average fuel plate temperature also slightly increased. The inlet water temperature remained essentially at ambient during the five to six minute duration of these tests.

Fig. 17 Reactor power and cladding surface temperature behavior in a narrow and a standard coolant channel during an 18-foot hydrostatic head, natural-circulation chugging test (No. SS 78).

It should be noted that the performance of these chugging tests corresponds to a simulated reactor start-up accident where the control rods are continually withdrawn from a low-power, critical position to their upper-position limit. With the reactor initially critical at a power level of about 1 W and using an average reactivity addition rate of 0.15\$/sec, it was determined in these tests that the Spert IV D-12/25 core reached a peak power of about 30 MW and an equilibrium power level of about 10 MW for a 5.0\$ total reactivity insertion.

4.2 2-Foot Hydrostatic Head, Natural-Circulation Chugging Tests

To determine the effect that the hydrostatic head above the core had on the stability characteristics of the reactor, the hydrostatic head was reduced to two feet; and the chugging tests performed with the 18-ft head were repeated. As in the tests with the higher head, the reactor power did not diverge with any reactivity insertion up to the maximum of 5.0\$ which was used. The primary difference observed between the tests with the two hydrostatic heads were those associated with the differences in water saturation temperature (96°C with a 2-ft head compared with 109°C for an 18-ft head). For example, the equilibrium powers in the 2-ft-head tests ranged from approximately 6.5 to 8.0 MW, compared with 7.5 to 10 MW for the higher head tests, and the measured fuel plate surface temperatures were slightly lower, reflecting the decreased energy generation rates. Also, the oscillation frequency of the surface temperatures of fuel plates adjacent to the narrow moderator-coolant channels was more nearly the same as that of temperatures adjacent to the standard channels. Both the amplitude and frequency of the power oscillations remained essentially the same as observed in the 18-ft hydrostatic head tests (ie, power

oscillations were generally within 30 percent of the mean value with a frequency of approximately 10 cps).

4.3 Spectral Moment Frequencies

Spectral moment frequencies [12] were determined for the chugging tests in order to evaluate the usefulness of these quantities in predicting the onset of instability. The spectral moments are given by:

$$\overline{f^n} = \frac{\int_0^\infty f^n P(f) df}{\int_0^\infty P(f) df}$$

where

$P(f)$ is the spectral power density of a noise function, $x(t)$, having a duration (T) , and is defined as:

$$P(f) = \lim_{T \rightarrow \infty} \frac{1}{T} \left| \int_{-\frac{T}{2}}^{\frac{T}{2}} x(t) \exp(-j2\pi ft) dt \right|^2$$

For $n = 2$, the root-mean-square frequency is defined as $f_{rms} = (\overline{f^2})^{1/2}$, and the fourth moment frequency ($n = 4$) is given by $f_{4m4} = (\overline{f^4})^{1/4}$. If any single frequency is dominant in $P(f)$, then all the moments $(\overline{f^n})$ are approximately n th powers of this frequency. As the stability of a reactor is reduced and the resonant frequency (f_0) becomes dominant, f_{rms} and f_{4m4} both approach f_0 . As the ratio of these quantities approaches unity, the reactor should approach the threshold of instability. This technique is described more fully in Appendix A-2.

Figure 18 shows the ratio of f_{rms} and f_{4m4} plotted as a function of inserted reactivity for both the 18- and 2-ft hydrostatic head chugging tests. As can be seen in this figure, this ratio does not approach unity for either condition, thereby indicating that the reactor would not become unstable with the maximum amount of excess reactivity (5.3%) available with the D-12/25 core with ambient inlet water temperature. It also can be seen that the values of f_{rms}/f_{4m4} were consistently greater in the 2-ft hydrostatic head tests than in the 18-ft hydrostatic head tests, thus indicating that the reactor was somewhat less stable with the lower head. The brief duration of these chugging tests allowed the inlet water temperature to remain near ambient for both head conditions.

4.4 Data Summary From Chugging Tests

A summary of the data which were obtained from the Spert IV chugging tests is presented in Table VI.

Fig. 18 Ratio of spectral moment frequencies as a function of total reactivity insertion for 2-foot and 18-foot hydrostatic head chugging tests.

TABLE VI

DATA SUMMARY FROM CHUGGING TESTS

18-Foot Hydrostatic Head, Natural-Circulation Tests						
Stability Test Number	Reactivity (\$)	Initial Peak Power (MW)	Equilibrium Power Level (MW)	RMS of Reactor Noise (MW)	Initial Inlet Water Temperature (°C)	Inlet Water Temperature at End of Test (°C)
70	3.00	26	7.6	0.68	25.6	33.6
71	3.50	33	8.5	0.97	24.9	28.9
73	4.00	33	9.2	1.06	23.8	26.8
75	4.25	32	9.6	0.85	24.5	28.5
76	4.50	34	9.7	0.95	22.4	25.4
77	4.75	31	9.7	0.98	23.3	26.3
78	5.00	32	10.1	1.05	23.7	31.7
2-Foot Hydrostatic Head, Natural-Circulation Tests						
79	3.00	32	6.7	0.64	23.3	26.3
80	3.50	32	7.0	0.71	24.1	30.1
82	4.00	31	7.3	0.76	21.4	28.4
83	4.25	31	7.6	0.80	22.5	27.5
84	4.50	32	7.2	0.93	22.1	28.1
85	4.75	31	6.9	0.88	23.6	30.6
87	5.00	31	8.0	0.89	23.1	38.1
89	4.00	31	Drops from 7.5 to 4 MW	Maximum value 3.03	22.0	69.0
						Reactor power diverged after ~ 12.5 minutes

5. CHUGGING TEST WITH ABOVE-AMBIENT INLET WATER TEMPERATURE

Of the various threshold, ringing, and chugging tests which were performed with the D-12/25 core, in only a few of the threshold tests did the reactor power exhibit any tendency toward divergent oscillations. From a comparison of results, it was concluded that the variation in the core inlet water temperature was responsible for the difference between the tests in which the reactor power exhibited a tendency toward instability and the other tests. In the threshold tests, the reactivity was inserted in small increments; and sometimes a large fraction of an hour was required to complete the insertion. Thus, in the relatively long time that the reactor was at power, the inlet water temperature to the core was significantly increased (especially in the threshold tests in which the reactor power displayed a tendency toward instability). In the chugging tests, however, all reactivity insertions were completed within approximately 30 seconds; and the inlet water temperature remained essentially at ambient for the duration of the tests. Similarly, the ringing tests were performed with the inlet water temperature at near ambient, and the desired damped power oscillations were not obtained, indicating that the reactor was not near the threshold of instability under these operating conditions.

In order to gain a better understanding of the influence of the inlet water temperature on the stability of the D-12/25 core, a chugging test was performed in which 4.0\$ of reactivity was inserted in ramp-fashion; and the reactor was allowed to operate at power until the inlet water temperature had raised sufficiently to allow divergent power oscillations. The initial bulk water temperature for this test was 22°C, and there was a 2-ft hydrostatic head above the core. The results of this test, No. SS 89, are described in the following sections.

5.1 Power Behavior

The initial power behavior was typical of previous chugging tests, being characterized by a ramp-induced peak of approximately 30 MW and a subsequent equilibrium level of 7.5 MW with high-frequency (about 10 cps) oscillations of ± 30 percent about this level. The 10 cps, small-amplitude power behavior prevailed for about 11 minutes after the start of the ramp; following which time, a 2 cps oscillation became increasingly dominant. After approximately 12.5 minutes, the reactor power exhibited divergent oscillations (ie, a series of pulses whose amplitude continued to increase throughout the period of observation with no apparent tendency to approach a limiting value). Successive power peaks reached 7.5, 15, 25, and 50 MW with a frequency of about 2 cps. The last three power oscillations displayed minimum reactor periods of 48, 35, and 24 msec, respectively. The amplitude envelope of the power oscillations was on an exponential rise corresponding to about a 790-msec period. Figures 19 and 20 show the reactor power as a function of selected times for this test. In accordance with criteria established for this test, the reactor was scrammed immediately following the 50 MW peak. At the time that the reactor power diverged, the inlet water temperature to the core had risen from the ambient temperature of 22°C up to 70°C. As a result of temperature feedback effects and fission-product poisoning, the mean power level had dropped from 7.5 to 4 MW by the time the reactor power diverged. A plot of the mean power level versus inlet water temperature for this test is shown in Figure 21.

Fig. 19 Reactor power behavior during test No. SS 89.

Fig. 20 Divergent power behavior during test No. SS 89.

Fig. 21 Mean reactor power level as a function of inlet water temperature during test No. SS 89.

22 displays in detail the reactivity, log power, and the time derivative of log power as a function of time for the last 2 sec of test No. SS 89.

The reactivity corresponding to the final four peaks of the divergent power (Figure 22) had the increasing maximum values of 0.52, 0.83, 0.94, and 1.11\$. These peaks in the reactivity occur simultaneously with the maximum values of the time derivative of the power; the last four oscillations having minimum periods of 97, 48, 35, and 24 msec, respectively. The peak values of reactivity occurring during the divergent power oscillations are less than the values to be found from the inhour curve for the Spert IV D-12/25 core due to the fact that none of these power oscillations ever attained an asymptotic period. The conditions prevailing during the divergent power oscillations are nonequilibrium and the delayed neutron contribution is such that faster rise-times are produced in the same manner as the "prompt jump" is produced during a step-induced transient. It can be seen from the figure that the positive and negative reactivity oscillations occur in the core as large amplitude oscillations or spikes which are not well approximated by step reactivity insertions.

The large changes in the time behavior of the reactivity are interpreted as positive and negative feedback contributions from the moderator coupled with the power where the primary feedback mechanism is oscillation of the void volume.

5.3 Noise Analysis of Test No. SS 89

Of the various stability tests performed with the D-12/25 core, test No. SS 89 provided the best opportunity for the evaluation of the various noise analysis techniques in predicting instability. Therefore, the spectral power density, the rms value of the reactor power noise, and the second and fourth spectral moment

5.2 Reactivity

The net reactivity as a function of time during test No. SS 89 was calculated using the IBM 7040 computer program REACTIVITY [13]. This program calculates the reactivity from the conventional, lumped-parameter reactor kinetic equations using experimental power data as input. This calculation, which gives values of the reactivity as a function of time immediately prior to and during the divergent power oscillations, was performed for the last 30 sec of the test.

During the 30 sec prior to the divergent power oscillations, the reactivity oscillated about the value of zero with an average amplitude of $\pm 0.8\%$ and a characteristic frequency of 2 cps. Occasionally, larger amplitudes of negative reactivity of approximately -1.80 to -1.00% appeared. The reactivity continued with this behavior until the beginning of divergent oscillations. Figure

Fig. 22 Reactor power, reactivity, and reciprocal period as a function of time during test No. SS 89.

frequencies (see Appendix A) were calculated for various times during the test. Analysis of these quantities as functions of the inlet water temperature showed that these quantities can be of value in predicting instability and provide a quantitative evaluation of the importance of the inlet water temperature to the stability of the core.

The last 12 minutes of power data from test No. SS 89 were analyzed in four overlapping sections, each of four minutes duration, in order to evaluate the changes made in the spectral power density functions as the inlet water temperature increased. Figure 23 shows the four spectral power density plots and the associated average inlet water temperature for that interval of the test (the separation between curves is arbitrary). It can be seen in this figure that a resonant peak in the spectral power density occurred at about 10 cps. This peak is assumed to be associated with the rapid growth and collapse of bubbles in the large portion of the core in which nucleate boiling was occurring. In the final four minutes of the test, this peak shifted to a lower resonance frequency of about 8 cps. As the inlet water temperature increased, the spectral power density amplitude in the frequency region from 1.5 to 2.5 cps increased relative to the amplitude at 1 cps where a resonance occurred for the first part of the test. In the final portion of the test, a resonance region from about 1.3 to 2.5 cps developed with the peak at 1.6 cps. A plot of the ratio of spectral power density amplitudes at 1 and 1.6 cps as a function of the inlet water temperature is shown in Figure 24. Extrapolation of the curve to the point where the ratio of these amplitudes approaches zero indicates the onset of divergent power

Fig. 23 Spectral power density as a function of frequency at four different average inlet water temperatures during test No. SS 89.

Fig. 24 Ratio of spectral power density amplitude at 1 cps and 1.6 cps as a function of inlet water temperature during test No. SS 89.

from 30 seconds for the first part of the test to 5 seconds in the final portion of the test. Figure 25 shows the average power level divided by the rms value of the reactor power noise plotted as a function of the inlet water temperature during this test. In this figure, it is seen that there was no strong dependence of reactor stability on inlet water temperature up to about 60°C but that the stability of the system decreased rapidly with increasing temperature beyond this value. From this result, it is concluded that although the method may ultimately predict the correct result, small steps in the independent variable (in this case the inlet water temperature) must be made to prevent the occurrence of an unpredicted unstable situation.

The spectral moment frequencies were calculated by counting the number of maxima and mean crossings contained in an interval of a power trace. Because of the increase of the nonlinear rate in the inlet water temperature, the sampling interval varied from 30 sec for the first part of the test to 5 sec in the final portion of the test.

Figure 26 shows the ratio of the spectral moment frequencies as a function of the inlet water temperature for test No. SS 89. The precision of this method depends on the amount of data processed. Consequently, the brevity of the power data available for any given inlet water temperature during this test resulted in considerable scatter in the calculated quantities. It can be seen in this figure, however, that the stability of the system decreased sharply for inlet water temperatures above approximately 67°C. The accuracy and usefulness of this method should have been improved if the inlet water temperature had been a controlled parameter allowing several minutes of data to be recorded at various steady-state inlet water temperatures.

oscillations at about 70°C. The duration of the low-frequency, large-amplitude oscillations at the higher inlet water temperature was too brief to permit an accurate power spectrum measurement above 64°C. The choice of frequencies associated with the ratio of the spectral power density amplitudes is based on an examination of the four-power spectrum plots. In order to accurately predict the onset of divergent power behavior by such an application of power spectrum measurements, it appears that the reactor would have to be operated for several minutes under controlled system conditions approaching those required for instability.

The rms value of the reactor noise was calculated as a function of the inlet water temperature for test No. SS 89. Each rms value was calculated from 2000 data points. Due to the increasing inlet water temperature, the sampling interval of these 2000 data points varied

Fig. 25 Average power divided by rms value of reactor power noise as a function of inlet water temperature for test No. SS 89.

Fig. 26 Ratio of spectral moment frequencies as a function of inlet water temperature for test No. SS 89.

5.4 Temperature Behavior During SS No. 89

During test No. SS 89, both fuel plate surface and moderator temperatures were measured at various core positions using chromel-alumel thermocouples. All measured temperatures exhibited an oscillatory behavior throughout the test. In general, however, the oscillations in the temperatures were not in phase with each other or with the reactor power until the 2 cps power oscillations began near the end of the test.

5.41 Fuel Plate Surface Temperature Measurements. At the start of the test, all of the measured fuel plate surface temperatures rose to an initial peak (maximum measured value 133°C) with the initial power burst, decreased to a minimum, and then rose to the saturation temperature as the power approached the equilibrium level. The temperatures then developed an irregular cyclic behavior with a frequency of approximately 0.3 to 0.5 cps.

With the onset of boiling, differences in the natural circulation flow characteristics between the standard and narrow moderator channels caused differences in the respective fuel plate temperatures; however, the differences were of degree rather than of kind. In general, the temperatures from fuel plates adjacent to the narrow channels displayed larger and more clearly defined oscillations than temperatures from fuel plates adjacent to the standard channels, but both exhibited the same type of cyclic behavior.

Starting at the low temperature portion of a typical temperature cycle, the following behavior was exhibited: (a) the temperature remained slightly above the saturation level with fluctuations of 2 to 3°C (characteristic of nucleate boiling) for approximately 1 to 2 seconds; (b) the temperature then rose sharply 15 to 20°C and remained at that level exhibiting irregular fluctuations of 2 to 5°C for 1 to 2 seconds; (c) the temperature again rose sharply 15 to 25°C and immediately decreased sharply to the saturation level having formed a sharp spike which ended the cycle. The behavior would then be roughly repeated. Figure 27 shows a typical fuel plate surface temperature, measured during the early portion of the test, from a fuel plate adjacent to a narrow moderator channel.

During the early portion of the test, the fuel plate surface temperatures throughout the core showed no overall phase relationship with each other. Further, there was no apparent relationship between the temperatures and the 10 cps power oscillations, except that the envelope of these power oscillations had a frequency of approximately 0.4 to 0.5 cps, or very nearly the same as that of the temperatures. The frequency of the envelope of the power oscillations, however, was not nearly as evident nor well defined as that of the temperatures.

After about the first two minutes of the test, the frequency of occurrence of the sharp spikes at the end of each temperature cycle gradually decreased. After about six minutes, the sharp spikes were essentially nonexistent; and the amplitude and steepness of the remaining broad peaks began to decrease. After about nine minutes, the initial cyclic behavior had damped out until it was no longer distinguishable; and the temperatures thereafter remained fairly constant at slightly above the saturation temperature with occasional spikes of about 10°C. The amplitude of these spikes continued to decrease until the divergent power oscillations started at the end of the test. Figures 28 and 29 show typical fuel plate surface temperature measurements at approximately six and nine minutes, respectively, after the start of the test.

Fig. 27 Cladding surface temperature behavior during early part of test No. SS 89.

Fig. 28 Cladding surface temperature behavior six minutes after start of test No. SS 89.

Fig. 29 Cladding surface temperature behavior nine minutes after start of test No. SS 89.

When the 2 cps power oscillations started near the end of the test, all measured fuel plate surface temperatures increased and decreased with the power showing a direct linkage between the fuel plate temperature and the reactor power. The fuel plate surface temperature and power behavior at the time of divergence of the power oscillations are shown in Figure 30.

5.42 Water Channel Temperature Measurements. During test No. SS 89, moderator temperature measurements were made at selected vertical positions in three different moderator channels (one in the central region of the core and two near the edge of the core) using 3-mil, chromel-alumel thermocouples. In the presentation of the data from these measurements, temperatures measured at the inlet, vertical center, and outlet of the channels will be discussed.

The temperatures at the center and at the outlet of the moderator channels increased after the initial ramp-induced power burst and approached equilibrium values (proportional to the integrated axial thermal neutron flux) at about 10 sec after the start of the test.

In coincidence with the initiation of the oscillations of the power and the fuel plate surface temperatures, the moderator temperatures began to exhibit well-defined oscillations. In a typical oscillation, the moderator temperatures at the center and at the outlet of the channels would rise to the saturation temperature. The rise at these positions would then be followed by a rapid increase and decrease (temperature spike) in the inlet temperature. About 400 msec following the spike at the inlet, the temperature at the center would decrease rapidly; and about another 400 msec later, the temperature at the outlet would show a rapid decrease. The cycle would then be repeated with a frequency of

Fig. 30 Reactor power and cladding surface temperature behavior at the time the power became unstable during test No. SS 89.

0.3 to 0.5 cps or the same as that of the fuel plate surface temperatures. Although temperatures measured in the various channels exhibited the same type of behavior, the temperature oscillations were not in phase from one fuel assembly to the next.

Occasionally, the moderator temperature at the center of the channel would not reach the saturation temperature during an oscillation or a series of oscillations. When this happened, the temperature spike at the inlet either did not occur or was considerably reduced. This type of behavior was most prevalent during the latter portion of the test. The temperature at the outlet, however, almost invariably reached saturation during an oscillation.

Figures 31 and 32 show the moderator temperatures measured at the inlet, center, and outlet of the center channel of the fuel assembly in grid position E-6 for various times during the test. Figure 33 shows the minimum temperature of the oscillations at the three vertical positions as a function of time. The data points in this figure are for several consecutive oscillations at the selected times.

5.5 Summary of Test No. SS 89

Experimental results from the threshold tests and the chugging tests indicated that the natural circulation inlet water temperature was one of the most significant variables with regard to the stability of the D-12/25 core. Therefore, a test (test No. SS 89) was planned and performed in which 4.0\$ of reactivity was added in a ramp; and the reactor was allowed to operate at the resulting power level, thereby increasing the inlet water temperature, until the reactor power became unstable. The initial bulk water temperature for this test was 22°C, and there was a 2-ft hydrostatic head above the core.

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Fig. 31 Inlet, center, and outlet water channel temperature behavior for selected times during first half of test No. SS 89.

Fig. 32 Inlet, center, and outlet water channel temperature behavior for selected times during last half of test No. SS 89.

Fig. 33 Minimum of water channel temperature oscillations at the inlet, center, and outlet positions as a function of time during test No. SS 89.

The initial reactor behavior during this test was typical of previous tests performed under similar conditions. Following the ramp-induced peak of approximately 30 MW, the reactor power reached an average equilibrium level of 7.5 MW with 10 cps fluctuations of approximately ± 30 percent about this level. In association with the initial power peak, measured fuel plate surface and moderator temperatures increased and became oscillatory in behavior. The fuel plate surface temperatures rose to initial peaks of 120 to 130°C (maximum measured 133°C) and thereafter remained slightly above the saturation temperature with oscillations up to 140 to 150°C. The frequency of these oscillations was approximately 0.3 to 0.5 cps, or approximately the same as that of the envelope of the 10 cps power oscillations. The moderator temperatures were also oscillatory with the same frequency.

The 10 cps, small-amplitude power fluctuations prevailed for about the first 11 minutes of the test. During this time, the average reactor power dropped to about 4.5 MW; the inlet water temperature increased to about 60°C; and the oscillations of the fuel plate surface temperatures decreased considerably in amplitude.

After the first 11 minutes of the test, a 2 cps power oscillation became increasingly dominant; and after 12.5 minutes, the reactor power exhibited 2 cps divergent oscillations with successive peaks of 7.5, 15, 25, and 50 MW (the last of these having a minimum reactor period of 24 msec). In accordance with criteria established for the test, the reactor was scrammed following the 50 MW peak. A maximum fuel plate surface temperature of 130°C and a peak transient pressure of 4.5 psig were measured in association with the 50 MW peak. At the time that the reactor power developed the divergent oscillations,

the inlet water temperature had risen from ambient to approximately 70°C; and the average reactor power had dropped to 4 MW.

In order to obtain a quantitative evaluation of the effect of the inlet water temperature on the stability of the D-12/25 core, the rms value of the reactor power noise, the ratio of the second and fourth spectral moment frequencies, and the spectral power density were calculated from the power behavior during test No. SS 89 for various inlet water temperatures. The inverse rms value of the reactor power noise remained fairly constant for temperatures up to approximately 60°C. For higher temperatures, however, the value dropped sharply; extrapolating to zero (indicating an unstable power behavior) at 70°C. The use of the other quantities showed similar results.

From the results of test No. SS 89, it is concluded that had the reactor not been scrammed, the power oscillations would eventually have reached damaging, if not destructive, proportions as the inlet water temperature continued to increase. Thus, the results from this test clearly demonstrate the importance of the inlet water temperature to the stability of low-pressure, water-moderated reactors.

IV. SUMMARY AND CONCLUSIONS

An experimental program has been conducted in the Spert IV facility to determine the stability characteristics of a highly enriched, plate-type core. In this program, tests were performed to (a) determine the threshold of instability for the core, (b) evaluate the applicability and validity of various experimental techniques for predicting the threshold of instability, and (c) observe the behavior of the reactor once the threshold of instability is exceeded. To accomplish the objectives of the program, three different types of tests were performed: threshold tests, ringing tests, and chugging tests.

- (1) Threshold Tests -- In these tests reactivity, up to 4.2\$, was added in small increments to the core until oscillations of approximately ± 50 percent of the mean power level were observed. Data from these tests were used to evaluate the use of various noise analysis techniques in predicting the threshold of instability.
- (2) Ringing Tests -- In these tests, stepwise insertions of reactivity were made with the reactor initially operating at high power in order to induce a damped oscillatory behavior in the reactor power. In theory, the damping factors obtained from such tests can be extrapolated to zero at the threshold of instability.
- (3) Chugging Tests -- In these tests, large amounts of excess reactivity (up to 5\$) were inserted in ramp-fashion in order to induce large-amplitude, divergent power oscillations.

It was determined in this program that the stability of the reactor was strongly dependent on the inlet water temperature to the core and that this temperature must be considerably above room temperature in order for the reactor to become unstable following its compensation of reactivity insertions of 5.0\$ or less. Most of the tests were performed with the inlet temperature at ambient, this temperature not being a controlled variable at higher temperatures. In the one test in which large-amplitude divergent power oscillations were obtained, 4.0\$ of reactivity was inserted as a ramp; and the inlet water temperature was increased to approximately 70°C as a result of operation of the reactor at the resulting steady power level before the reactor became unstable. The following is a summary of the results of this program, which has now been terminated to permit use of the facility in obtaining urgently needed safety information on other core types (principally low-enrichment oxide cores).

1. THRESHOLD TESTS

The purpose of these tests was to determine the reactivity and/or power level conditions required to make the reactor unstable and to obtain data for noise analysis purposes. It was not intended in these tests that the reactor become unstable; therefore, the reactivity was inserted in small predetermined increments, and the reactor was operated at each resulting steady-state power level for several minutes to observe for possible instabilities (the reactor power was displayed on oscilloscopes and on strip-chart recorders) and for

the acquisition of data. In order to set a practical limit for performing the tests without undue risk to the core, an operational limit was defined as that for which oscillations of ± 50 percent about the mean power level were obtained. Such tests were performed in Spert IV for the following initial system conditions: (a) ambient temperature, 18-ft hydrostatic head, no forced coolant circulation; (b) ambient temperature, 2-ft hydrostatic head, no forced coolant circulation; (c) ambient temperature, 18-ft hydrostatic head, 1.2 ft/sec (500 gpm) forced coolant circulation through the core.

With no forced flow through the core, large amounts of reactivity were inserted in small increments with both an 18- and 2-ft hydrostatic head above the core. In tests with a total of 4.1\$ inserted, the reactor power exhibited 4-cps nondivergent oscillations of up to ± 60 percent of the mean power level with an 18-ft head and showed a definite tendency toward 2-cps divergent oscillations with a 2-ft head. However, these results do not necessarily show that the reactor is more unstable with a 2-ft head than with an 18-ft head because of differences in the core inlet water temperature. By the time that the 4.1\$ insertions had been completed, the inlet water temperature had risen to 70°C in the 2-ft-head test compared with 35°C in the 18-ft-head test.

Power oscillations of ± 50 percent were not obtained during tests with forced coolant circulation because of potential fuel plate burnout problems. With a forced coolant rate of 1.2 ft/sec and a reactivity insertion of 1.6\$, surface temperatures from fuel plates adjacent to narrow moderator-coolant channels in the core oscillated from approximately 140°C up to 300 to 400°C. The power behavior at that time, however, was relatively stable, being characterized by regular fluctuations of less than ± 15 percent of the mean level. From this result, it was concluded that fuel plate burnout would probably occur prior to divergent power oscillations during tests with flow; and no further tests with forced flow were performed.

2. RINGING TESTS

In these tests, stepwise insertions of reactivity were made, following the attainment of an equilibrium power level condition for a given initial reactivity insertion, in an effort to induce an oscillatory power behavior. In theory, if a system is near the limit of stability, the power response resulting from a step-reactivity perturbation will take the form of damped oscillations of constant frequency corresponding to the principal resonance of the system transfer function; therefore, the damping characteristics of the oscillatory behavior should provide a quantitative measure of the stability of the system.

In the ringing tests performed in Spert IV, initial reactivities of up to 3.75\$ were inserted with an 18-ft head at ambient bulk water temperature conditions. Following the attainment of equilibrium power conditions, stepwise reactivity insertions in the range from 0.25 to 1.0\$ (usually 0.5\$) were made to induce the damped power oscillations. In all of the tests which were performed, however, the damped oscillations were either not present or were masked by the general noise level. It was concluded that if the tests had been performed at higher inlet water temperatures, the damped power oscillations may have been obtained; but these tests did not allow any conclusions as to the applicability of the technique in predicting instability.

3. CHUGGING TESTS

In the chugging tests, large amounts of excess reactivity (3.0 to 5.0%) were added to the system in ramp-fashion in an effort to induce divergent oscillations in the reactor power. The tests were initiated with the bulk pool temperature at ambient, with either an 18- or 2-ft hydrostatic head above the core, and with only natural circulation through the core. In most of the tests, the inlet water temperature through the core was maintained at ambient. Under these conditions, however, the reactor power could not be made to diverge with any of the combinations of hydrostatic head and reactivity insertion used. The power oscillations generally remained within 30 to 40 percent of the mean value with a frequency of approximately 10 cps. Subsequently, a test was performed in which a 4% reactivity insertion was made and the reactor allowed to operate at equilibrium power, thereby increasing the inlet water temperature until the reactor power oscillations diverged. In this test, successive power peaks of approximately 7.5, 15, 25, and 50 MW with a frequency of 2 cps were obtained when the inlet water temperature reached approximately 70°C, and approximately 2% of reactivity was held in steam. In the 5.0% chugging test, with the inlet water temperature at ambient, approximately 3% was held in steam without resulting in divergent power oscillations. This result demonstrates the importance of the inlet water temperature to the stability of the reactor and also that the total reactivity in voids is not a sufficient quantity by which to evaluate the stability of a reactor.

For most of the Spert IV stability tests, power spectral densities and the rms value of the reactor power noise were determined as functions of the total reactivity inserted above low-power critical, the reactivity in voids, and the steady-state reactor power level. Since the inlet water temperature was not a controlled variable, reliable predictions of reactor instability were not obtained. Qualitatively, however, the reactor power spectra obtained did exhibit a resonance which increased with the total reactivity insertion, and the rms value of the reactor noise also increased with reactivity. Although quantitative results were not obtained which would verify the use of the various noise analysis techniques, it is concluded on the basis of the qualitative results obtained, that quantities derived from noise analysis can be useful in reactor stability analysis and/or as operational indicators in the relative stability of reactor systems.

The rms value of the reactor noise was calculated as a function of inlet water temperature in the one test in which large-amplitude, divergent power oscillations were obtained. The plot of the inverse rms value versus inlet temperature remained essentially flat for temperatures up to approximately 60°C and thereafter decreased rapidly to zero at 70°C. From this result, it is concluded that although the method may ultimately predict the correct result, small steps in the independent variable must be used in order to prevent reaching an unexpected unstable condition.

It is apparent from the results and conclusions presented in this report that additional testing and analysis is required before a complete understanding of the various factors which influence the stability of cores such as the D-12/25 core can be obtained. Much of the additional testing, as indicated by the tests in Spert IV, would require the performance of tests with controlled inlet water temperatures significantly above ambient. The Spert IV facility was designed

for a maximum bulk pool temperature of approximately 55°C, making such tests infeasible for performance in Spert IV.

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APPENDIX A

NOISE ANALYSIS TECHNIQUES

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APPENDIX A

NOISE ANALYSIS TECHNIQUES

Various techniques for predicting instability thresholds have been proposed by various authors in several articles on stability. At this time, the most successful method of predicting instability is based on the reactor transfer or describing function which is usually measured by rod-oscillator techniques. In the rod-oscillator experiment, the reactivity of the system is oscillated at various frequencies, usually by the sinusoidal oscillation of a neutron absorber, to measure the transfer function, $G(f)$, between the reactivity amplitude and the resulting reactor-power fluctuation:

$$G(f) = \frac{\delta n / N_0}{P}$$

where N_0 is the mean reactor power about which fluctuations, δn , occur. It is the most precise technique presently available; but it does have disadvantages in the expense of time, money, manpower, and equipment required.

It has been suggested that a number of characteristics of reactor noise may be determined in a relatively simple manner in order to obtain information about reactor stability [12]. These are:

- (1) Resonances in the spectral power density function. The higher and sharper such resonances are, the greater the tendency of the system is toward unstable oscillations at the resonant frequency.
- (2) The spectral moment frequencies. As instability is approached, the ratio of the root-mean-square frequency and the fourth moment frequency approach unity.
- (3) The root-mean-square value of the reactor power noise. The ratio of average power/rms value approaches zero for unstable, divergent reactor power behavior.

1. SPECTRAL POWER DENSITY

The spectral power density, $P(f)$, of a noise function, $x(t)$, having a duration, T , is defined as:

$$P(f) = \lim_{T \rightarrow \infty} \frac{1}{T} \left| \int_{-\frac{T}{2}}^{\frac{T}{2}} x(t) \exp(-j2\pi ft) dt \right|^2$$

The spectral power density and reactor transfer function are related for the usual linear stationary system by:

$$P(f) = (\text{input reactivity spectral power density}) (|\text{transfer function}|^2).$$

The spectral power density function and the square of the transfer function will have the same frequency dependence only if the spectrum of the boiling noise reactivity input is constant or so-called "white noise". It has been shown, however, in other boiling water reactors [15] where both the transfer function has been measured by the rod oscillator technique and the power spectral density function has been determined that the boiling noise spectrum is not white but varies with frequency. The primary disadvantage of power-spectral analysis is that the phase relationship between the input reactivity noise and the output power response is not directly determined as in direct transfer-function experiments. The power spectrum function is a useful quantity in itself, however, since the reciprocal value of the spectral power density amplitude at the resonant frequency should approach zero as a reactor approaches instability.

For the Spert IV stability tests, spectral power densities were determined by analyzing FM magnetic tape records of reactor power data with a Boonshaft and Fuchs Analog Power and Cross Spectrum Computer. In most cases, five minutes of reactor power noise from each test was spectrum analyzed in the frequency range from 0.25 to 20 cps.

2. SPECTRAL MOMENT FREQUENCIES

Spectral moment frequencies [12] were determined in order to evaluate the usefulness of these quantities in predicting the onset of instability. The spectral moments are given by:

$$\overline{f^n} = \frac{\int_0^\infty f^n P(f) df}{\int_0^\infty P(f) df}$$

where $P(f)$ is the spectral power density of a noise function, $x(t)$, having a duration, T , and is defined as:

$$P(f) = \lim_{T \rightarrow \infty} \frac{1}{T} \left| \int_{-T/2}^{T/2} x(t) \exp(-j2\pi ft) dt \right|^2$$

For $n = 2$, the root-mean-square frequency is defined as $f_{rms} = (f^2)^{1/2}$, which Rice [16] has shown to be equal to $(1/2 \text{ multiplied by the number of times the noise signal crosses its mean per second})$. For $n = 4$ the fourth moment frequency may be defined as, $f_{4m4} = (f^4)^{1/4}$ which can be evaluated by counting the number of maxima in a noise function:

$$f_{4m4} = \left(\frac{\text{number of maxima}}{\text{sec}} \right)^{1/2} \left(1/2 \times \frac{\text{the number of mean crossings}}{\text{sec}} \right)^{1/2}$$

If any single frequency is dominant in $P(f)$, then all the moments, $\bar{f}^n$, are approximately n th powers of this frequency. As the stability of a reactor is reduced and the resonant frequency, f_0 , becomes dominant, f_{rms} and f_{4m4} both approach f_0 . As the ratio of these quantities approaches unity, the reactor should approach the threshold of instability.

The spectral moment frequencies were calculated as described above by counting the number of maxima and mean crossings found in a 30 sec interval of a power trace from a recording oscillograph. Due to the increasing inlet water temperature in test No. SS 89, the sampling interval varied from 30 seconds for the first part of the test to 5 seconds in the final portion of the test.

3. ROOT-MEAN-SQUARE VALUE OF REACTOR POWER NOISE

Akcasu [9] has pointed out that the rms value of the reactor noise may be used to predict the onset of instability in a noise excited system. The ratio of $\frac{\text{average power}}{\text{rms value}}$, $\frac{\bar{\phi}}{\Delta\phi_{rms}}$, is plotted as a function of some parameter such as power, reactivity, pressure, or temperature. Extrapolation of the curve to the point where the value of $\frac{\bar{\phi}}{\Delta\phi_{rms}}$ approaches zero indicates divergent power behavior.

Power data was recorded in analog form on FM magnetic tape and then digitized by an analog-to-digital converter. An IBM 7040 computer was used to calculate the rms value of the reactor power noise from the following relationship:

$$\Delta\phi_{rms} = \frac{\left(\sum_{i=1}^N (X_i)^2 - (\bar{X})^2 \right)^{1/2}}{N}$$

where

$$\bar{X} = \frac{\sum_{i=1}^N X_i}{N}$$

X_i = discrete power level values

N = number of data points.

For most of the tests, ten thousand data points with a digitizing interval from 10 to 20 msec were used in the calculation of the rms value.

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APPENDIX B

INSTRUMENTATION

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APPENDIX B

INSTRUMENTATION

During the Spert IV stability 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 was recorded on two 36-channel recording oscillographs, with 39 of these channels also being recorded on magnetic tape. The primary variables which were recorded were reactor power, fuel plate surface temperature, water channel temperature, transient pressure, and reactor tank strain.

During the Spert IV stability tests, recording of the power behavior required maximum coverage due to the possible occurrence of divergent power oscillations. In order to properly record the characteristics of the transient power behavior under these conditions, the power-measuring instrumentation gave coverage over approximately six decades of power rise. For each test, normally six chambers were used, each chamber covering from 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 chamber. The types of chambers which were used were miniature ion chambers (both boron- and uranium-lined), semirad chambers, and conventional B¹⁰-lined ion chambers. Both the miniature ion chambers and the semirad chambers had linear outputs up to approximately 10^{16} nv and were, therefore, positioned adjacent to the core in the reflector and also in the core proper. The conventional chambers had linear outputs up to only 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 and water channel 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 may be deduced.

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 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 and to provide temperature data from the suspected core hot-spots.

The water-channel temperatures were measured using single-junction chromel and alumel thermocouples which were fixed in selected moderator

channels midway between fuel plates at various vertical positions. During the Spert IV stability tests, water channel temperature measurements were made in three assemblies in grid positions E3 and E6. The water-channel temperature measurements were made at distributed vertical positions in the moderator. A more detailed vertical coverage for a single water channel was accomplished in the assembly in grid position E3. There, the thermocouples were located at +12, +6, 0, -6, and -12 inches from the core centerline.

In the temperature traces which are shown in the report, the fuel plate (and water-channel) thermocouple locations are 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 (or in the water channel next to) the west side of fuel plate number seven, and three inches below the core centerline.

Transient pressure pulses can be developed during large rapid changes in reactor power, owing to moderator and metal expansion and steam formation. In general, these pulses are significant only for short-period power rises and tend to increase in amplitude for shorter periods. Since the pressures are generated between the fuel plates and since 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 17.

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.

The Spert IV reactor tanks were designed for a 25-ft hydrostatic head plus a 50 psi overpressure. Since pressure pulses have been observed in reactor destructive testing (eg, in Borax I [18] and Spert I [19]) 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 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 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 tank wall.

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