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NUCLEAR START-UP OF THE SPERT IV REACTOR

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**PHILLIPS
PETROLEUM
COMPANY**

ATOMIC ENERGY DIVISION

**NATIONAL REACTOR TESTING STATION
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ABSTRACT

The Spert IV reactor facility is a large pool-type facility designed for the study of the kinetic behavior of a wide variety of reactor types. The first portion of the Spert IV experimental program will be a study of reactor instability as exhibited by highly enriched uranium-aluminum, water-moderated and reflected cores. The first core to be used in this program is a close-packed array of Spert type D 12-plate assemblies. Initial criticality was achieved with this core on July 24, 1962. The initial critical loading contained 21 fuel assemblies which was equivalent to 3.08 kg of U-235. The operational core, a 5 x 5 array of fuel assemblies containing 3.75 kg of U-235, has a maximum cold excess reactivity of 5.3\$.

This report describes the initial critical experiment, the operational core loading, and measurements of various static characteristics (void and temperature coefficients, control rod worth, neutron flux distributions, etc) which were made prior to the initiation of the kinetics testing program with this core. Also presented is a brief description of the Spert IV facility.

NUCLEAR START-UP OF THE SPERT IV REACTOR

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NUCLEAR START-UP OF THE SPERT IV REACTOR

I. INTRODUCTION

In August 1956, a series of tests ^[1] was performed in the Spert I reactor in which severe oscillations of the reactor power were observed. These tests dramatically demonstrated the hazards that can exist when a highly enriched, plate-type, water-moderated reactor is allowed to exceed the threshold of instability and indicated the need for further testing to better understand the nature of such reactor behavior. However, the limitations of the Spert I facility (a low-heat-capacity system with no provisions for forced coolant circulation or heat removal) did not permit adequate control of test conditions and, therefore, the reproducibility of the tests was poor, making a more comprehensive study in the Spert I facility relatively unprofitable. Therefore, in order to continue these studies of reactor instability under more closely controlled environmental conditions, to extend the range and type of controlled test parameters, and to provide a facility for the kinetic testing of reactor cores of advanced design, the Spert IV facility was constructed.

Spert IV is a large pool-type facility which includes provisions for forced coolant circulation at rates up to 5000 gpm and steady-state heat-removal capacity of 1 Mw. The reactor (like all Spert reactors) was designed to be controlled remotely from the Spert control center which is approximately one-half mile from the reactor building.

The initial portion of the Spert IV experimental program will be performed with a highly enriched, plate-type, light-water-moderated core. In particular, the behavior of the core when allowed to exceed the threshold of instability is of prime interest in this program. This type of core was chosen because of its extensive use in research and test facilities, and to help complete the information available on the kinetic behavior of such cores. Initial criticality was achieved on July 24, 1962, with the first core to be used in this program, the Spert IV D-12/25 core. In this report are presented brief descriptions of the Spert IV facility, the initial core, and the core parameter measurements made prior to the initiation of the kinetics testing program.

II. DESCRIPTION OF FACILITY AND REACTOR CORE COMPONENTS

A. Brief Description of the Spert IV Facility

The Spert IV facility, in general, is divided into two areas of operation, the control center area and the reactor building area. Because of the hazardous nature of some of the tests in the Spert IV experimental program, all nuclear operation of the reactor will be accomplished, remotely, from a control room in the Spert control center located about 1000 yd from the reactor building. The reactor building is a pumice block structure which houses the reactor pools, a forced coolant system, process and experimental instrumentation, a water treatment system, and other auxiliary equipment necessary to the operation of the facility. Operation of the major plant equipment may be performed from either the control room or the reactor building. A reactor building plan and section are shown in Figures 1 and 2.

The pool portion of the facility (Figure 3) is composed of two 20-ft-diam x 25-ft-deep tanks which are connected at the top by a 6- x 6-ft removable gate. The tanks are fabricated from 5/16-in.-thick stainless steel and are designed for 25-ft of hydrostatic head plus a 50-psi overpressure. The north, or operating pool, has a 16-in flanged nozzle centrally located in the bottom of the tank to allow forced circulation through the core. Each pool has two 12-in. nozzles in the pool wall for connection to the two 2500-gpm coolant pumps to allow various water circulation patterns to be established between the two pools. The coolant system includes a 1-Mw heat exchanger for stabilizing the pool temperature

Fig. 1 Spert IV reactor building Plan.

Fig. 2 Spert IV reactor building Section.

during power operation and cooling the pools following tests in which the bulk-water temperature is raised significantly above room temperature. The maximum design operating temperature for the reactor pools and coolant system is 130°F.

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

A detailed description of the Spert IV facility may be found in IDO-16745 [2].

B. Description of Reactor Core Components

The initial Spert IV core (Figure 4) is a highly enriched uranium-aluminum, plate-type core consisting of 20 fuel assemblies, 4 control rod assemblies, and 1 transient rod assembly in a 5 x 5 array. 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 similar to the control rods with the exception that its poison section is normally below the active core region. The core support structure includes an 81-position bottom core grid, a flow skirt to direct forced coolant circulation through the core, and fuel assembly hold-down bars. The principal characteristics of the Spert IV "D" core are summarized in Table I.

Fig. 3 Spert IV reactor vessels.

1. Fuel

The Spert type D assembly (Figures 5 and 6) which is used consists of a 3-in.-square x 27-5/8-in.-long 6061-T6 aluminum retaining can, a 2.7-in.-square 6061-T6 aluminum lower end box, two grooved side plates, a lifting

Fig. 4 Plan view of Spert IV D-12/25 core.

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 g of U-235 alloyed with aluminum melting stock to produce a "meat section" 0.020 in. thick, 2.45 in. wide, and 24 in. long. This fuel section is clad with 6061 aluminum to form a plate 2.704 in. wide, 25-1/8 in. long, and 0.060 in. thick. A fully loaded fuel assembly contains 168 g of U-235. The water channel spacing is nominally 0.179 in., but can be varied by use of different side plates or by the removal of some of the fuel plates.

TABLE I
CHARACTERISTICS OF SPERT IV REACTOR

	<u>Vessels</u>
Composition	Welded, rolled, type 304 stainless steel plate
Height	25 ft
Inside diameter	20 ft
Wall thickness	Top 23 ft: 5/16 in. Bottom 2 ft: 5/8 in.
Bottom thickness	1/2 in.
Design pressure	Below gate: 25-ft hydrostatic load plus 50-psi static surcharge. Above gate: 6-ft hydrostatic load plus 10-psi static surcharge.
	<u>Core</u>
Size	15 x 15 x 24 in.
Total core volume	5.4×10^3 in. ³
Moderator volume	3.26×10^3 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
Critical U-235 mass	3.0 kg (21 fuel assemblies)
Operational U-235 mass	3.75 kg
Cold excess reactivity	5.3\$
	<u>Fuel Assemblies</u>
Type	Plate
Number	
- standard	20
- modified (control)	5
Size	3 x 3 x 25-1/8 in.
Plates per assembly	
- standard	12
- modified	6
Total number of fuel plates in core	270
Fuel plate thickness	0.060 in.
Meat thickness	0.020 in.
Clad thickness	0.020 in.
Normal coolant channel thickness	0.179 in.
Outside coolant channel thickness	0.090 in.
Active length of fuel plate	24 in.
Meat	93% enriched U-Al alloy
Cladding	Al
U-235 content of plate	14.0 ± 0.2 g
	<u>Control Rods</u>
Number	
- control	4 gang-operated
- transient	1
Composition	"Binal" (7 wt% boron-aluminum alloy)
Total rod worth	20\$
Withdrawal rate	0 to 12 in./min
Scram time	Approximately 300 msec from upper limit

Fig. 5 Isometric view of 12-plate type D assembly.

are withdrawn. There are four control rod assemblies, each having two blades which fit in guide slots in the standard fuel assemblies. Each rodded assembly has 6 fuel plates instead of the normal 12 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 to decrease reactivity and dropped to increase the reactivity of the core.

Fig. 6 Plan view of 12-plate type D fuel and control rod assemblies.

2. Control Rods

The control rods for the "D" core are 5/16-in.-thick x 2-in.-wide blades made of Binal* which contains 7 wt% natural boron. The upper section of the control rod blade is the "Binal" section and is 32-7/8 in. long. The lower section is a 25-3/8-in.-long aluminum follower that serves as both a guide and flux peak suppressor as the control rod blades

* Trade name for the Sintercast Corp. aluminum-boron powder metallurgy processed material.

3. Control Rod Drive Units

The control rod drive units for the Spert IV D core were designed for open-pool operation with cores up to 36 in. long. Basically, the central rod drive units consist of four inverted hollow shaft screw jacks mechanically connected and driven by a variable-speed transmission. The control rod speed is variable from 0 to 12 in./min. A synchro-transmitter connected to the transmission output drives a digital indicator on the control console which indicates the control rod drive position to 0.01 in.

Control rod pickup is accomplished by means of an electromagnet suitable for underwater service (Figure 7). An adjustable spring is provided for initial acceleration when the control rods are scrambled. The transient rod magnet, also shown in Figure 7, is equipped with a mechanical latch to prevent an inadvertent rod drop, since dropping the transient rod increases the core reactivity. Hydraulic shock absorbers are integral with the drive units and are suitable for operation either above or below the water level.

4. Core Support Structure

The Spert IV core support structure (Figure 8) is a 9 x 9 lattice of 3- x 3-in. cells supported from the reactor control bridge. The grid is attached to a mounting plate, which also serves as the mounting surface for a core flow adapter which connects through an expansion joint to the 16-in. coolant nozzle in the bottom of the vessel. The square flow skirt provides lateral support for the assemblies, and serves as the hold-down bar mount as well as providing an enclosure for directing the water flow through the core. A sway guide that is fastened to the pool floor prevents lateral movement of the core structure. This sway brace is not mechanically connected to the grid structure, and allows vertical movement of the grid and support structure. The fuel assembly hold-downs are 3-1/4-in.-high x 3/8-in.-wide x 30-in.-long aluminum bars bolted to the top of the flow skirt weldment. These bars restrain the fuel assemblies in a vertical position between the lower grid assembly and the top surface of the flow skirt weldment.

Fig. 7 Control and transient rod magnet assemblies.

Fig. 8 Spert IV core support structure.

III. DESCRIPTION OF STATIC EXPERIMENTS

A. General

Prior to the initiation of the kinetics testing program with any Spert reactor core, a series of tests is performed in which various parameters of the particular core variant are determined. These "static" tests, which typically include measurements of control rod worths, flux distributions, void and temperature coefficients, and power calibration factors, are performed for the purpose of gaining sufficient information about the core to allow (a) a safely conducted kinetics testing program; (b) appropriate placement of core instrumentation, flux detectors, thermocouples, pressure transducers, etc; and (c) correlations to be made between the dynamic behavior of the reactor and static reactor parameters.

The following discussion includes descriptions of the loading of the Spert IV D-12/25 core and the measurements of the static parameters of the core. For all of the measurements described in this report, the core was suspended in a 7-ft-diam x 8-ft-high calibration tank which was installed in the north, or operating, reactor vessel. 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.

B. Initial Critical Experiment and Operational Core Loading

1. Initial Critical Experiment

The purpose of this experiment was to determine the minimum critical loading for the Spert IV 12-plate D core and to provide a safe and efficient method for loading the operational core. From previous studies in Spert I [3], using a core with similar geometry and equal fuel plate enrichment, the critical mass was expected to be approximately 3 kg of U-235.

In the approach-to-critical, the multiplication, M , of the core was determined from neutron counting data and the inverse-multiplication, $1/M$, technique was used to experimentally determine the number of assemblies or mass of U-235 that would be required for criticality. Following each fuel addition, neutron counting rates were obtained with the control rods fully inserted, raised to 6, 12, and 18 in., and to the upper limit of control rod travel, 23.2 in. For each core loading and control rod position, plots were made of reciprocal multiplication vs the number of assemblies in the core. Straight line extrapolations of these curves to $1/M = 0$ gave the predictions of the number of assemblies required for criticality.

Figure 9 is a diagram of the core and vicinity showing relative positions of neutron-sensing devices, and Figure 10 is a block diagram of the instrumentation used in the critical experiment. Channels A through F utilized B-10-lined pulse chambers with readout in the control room on decade scalars. The operational linear and operational log systems utilized gamma compensated current chambers with readout on strip-chart recorders in the control room. Chambers B and C were positioned so as to permit reliable neutron counting should systems A, D, E, and F become saturated.

A Ra-Be source, with an emission rate of approximately 10^6 neutrons per second, was fitted into the transient rod assembly and placed into position E-5

Fig. 9 Plan view of D-12/25 core showing operational and start-up instrumentation.

Fig. 10 Block diagram of operational and start-up instrumentation channels.

of the core (Figure 4). With the source in this position and no fuel in the core, initial neutron counting rates (C_0) were determined for systems A through F.

Table II summarizes the critical loading sequence by showing the load number, assembly numbers, their respective positions in the core, the mass

TABLE II
DATA FROM SPERT IV INITIAL CRITICAL EXPERIMENT

Load Number	Core Positions	Mass Addition (kg U-235)	Total Mass (kg U-235)	Total Number of Assemblies Loaded	Criticality Predictions A + D + E + F System	
					Number of Assemblies	Mass of U-235 (kg)
1	E - 5 D - 4 D - 6 F - 4 F - 6	0.415	0.415	5	—	—
2	F - 5 D - 5 E - 6 E - 4	0.665	1.08	9	14	1.85
3	G - 5 C - 5	0.321	1.41	11	17	2.32
4	E - 7 E - 3	0.333	1.74	13	17	2.57
5	C - 4 G - 6	0.332	2.08	15	19	2.65
6	D - 7 F - 3	0.334	2.41	17	20	2.86
7	F - 7	0.166	2.58	18	20	2.87
8	D - 3	0.167	2.74	19	21	2.97
9	G - 4	0.169	2.91	20	21	3.04
10	C - 6	0.169	3.08	21	[a]	[a]

[a] Critical with 21 assemblies and control rods at 21.6 in. withdrawn.

of U-235 added for each loading, the total mass of U-235 contained in the core, and the control-rod-at-upper-limit critical predictions from the combined counting rate of systems A, E, D, and F. Figure 11 shows the $1/M$ curve obtained from the combined rates of these four systems. Systems B and C have not been included in this table because of relatively poor statistics during the early part of the experiment. However, as loading progressed and consequently the counting rates from systems B and C increased, these agreed with the more sensitive systems in predicting the critical configuration would contain 21 assemblies.

Fig. 11 Inverse multiplication curve obtained during initial critical experiment.

Initial criticality was achieved on July 24, 1962 with 21 assemblies (3.08 kg of U-235) loaded in the core and with the control rods withdrawn to 21.6 in. Extrapolation of the $1/M$ curves indicated that the critical mass would be 3.04 kg of U-235 with the control rods withdrawn to the upper limit. The critical configuration can be most easily described as a 5 x 5 array with the four corners missing.

2. Operational Core Loading

Following the initial critical experiment, the loading of the Spert IV 12-plate, type D core was continued for the purpose of assembling a core with sufficient excess reactivity to allow a thorough investigation of reactor instabilities at various initial conditions of temperature, flow, and upper-reflector height. Based on the Spert I stability studies,^[1] where it was determined that

the core could stably compensate about 3.5\$ at ambient conditions and no forced circulation, a minimum excess reactivity requirement of 5\$ was decided upon for the Spert IV operational core. In addition, for safety reasons, a minimum shutdown margin of 3\$ was required.

Following each fuel assembly addition to the initial critical core, the reactor power was allowed to rise on several relatively long ($\tau > 10$ sec) periods, where the prompt term of the inhour equation is small compared with the delayed (summation) term, and, therefore, the reactivity of the system can be determined from the summation term. From these data, an approximate differential control rod worth curve was produced, the integration of which allowed the determination of approximate values for the reactivity worth of that fuel addition and the total available excess reactivity. By this method a core composed of a 5 x 5 array of 12-plate, type D control and fuel assemblies was determined to have an excess reactivity of 5.4\$. The total fuel loading in this core, the Spert IV D-12/25 core, was 3.75 kg of U-235. Figure 12 shows the total available excess reactivity and the control rod positions at critical vs the number of fuel assemblies in the core.

Fig. 12 Available excess reactivity and critical control rod position vs number of fuel assemblies in core.

Table III summarizes the excess loading sequence giving the load number, the grid position loaded, the mass of U-235 in the loading as well as in the core, the critical control rod positions,

TABLE III
DATA FROM OPERATIONAL CORE LOADING

Load Number	Core Position	Mass Addition (kg U-235)	Total Mass (kg U-235)	Total Number of Assemblies Loaded	Critical Control Rod Position (in. withdrawn)	Total Available Excess Reactivity (\$)
11	C - 3	0.166	3.25	22	18.78	1.62
12	G - 7	0.168	3.42	23	17.02	2.86
13	G - 3	0.167	3.58	24	15.58	4.12
14	C - 7	0.166	3.75	25	14.45	5.27

reactivity worth of the load, and the approximate excess reactivity of the core.

The shutdown margin for each intermediate core formed during the operational loading was determined by use of the integral-count, rod-drop method [4] where the shutdown margin, R, in dollars is given by:

$$R(\$) = \frac{N(o)}{n(o)} \sum_{i=1}^6 \frac{\gamma_i \beta_i / \gamma \beta}{\lambda_i}$$

where:

$N(o)$ = Neutron counting rate at critical (sec^{-1})

$n(o)$ = Total number of neutrons counted from time of rod drop until the count rate reaches source level.

γ_i = Delayed neutron effectiveness of the i^{th} delayed neutron group

γ = Average delayed neutron effectiveness

β_i = Fractional yield of the i^{th} delayed neutron group

λ_i = Decay constant of the i^{th} delayed neutron group (sec^{-1})

The shutdown margin for the Spert IV D-12/25 core was determined to be approximately 15\$ by this method. Table IV is a summary of the data taken in this experiment.

C. Control Rod Calibration

The control rods of the Spert IV D-12/25 core were calibrated as a bank from the cold, clean critical position to the upper limit of travel by the period technique using boric acid as a reactivity shim. The purpose of this experiment was to obtain a more precise value of the available excess reactivity of the core and to obtain operational information which is necessary for accurate adjustment of the control rods during the kinetics testing program.

TABLE IV
DATA FROM SHUTDOWN MARGIN EXPERIMENT^[a]

Number of Assemblies in Core	Control Rod Position at Critical (in. withdrawn)	Average Neutron Count Rate at Critical $N(0)$ (c/sec)	Total Number of Neutrons Counted After Scram $n(0)$	Shutdown Margin (\$)	Available Excess Reactivity of Core (\$)	Total Control Rod Worth (\$)
21	21.63	22,700	14,800	20.0	0.3	20.3
22	18.78	24,000	17,000	18.3	1.6	19.9
23	17.02	24,400	18,600	17.1	2.9	20.0
24	15.58	22,300	18,100	16.0	4.1	20.1
25	14.45	22,500	19,200	15.2	5.3	20.5

[a] Delayed neutron data from Keepin et al was used in this experiment and β/λ was assumed to be unity.

The differential worth of the control rods was found to vary, almost linearly, from 1.07\$/in. at the cold, clean critical position of 14.5 in. to 0.19\$/in. at their upper limit of travel 23.2 in. The available excess reactivity of the core was determined to be 5.3\$ from the data taken in this experiment. The differential and integral control rod worth curves for the Spert IV D-12/25 core are shown in Figure 13 and 14, respectively.

D. Void Coefficient Measurements

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

Fig. 13 Differential control rod worth.

Fig. 14 Integral control rod worth.

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

The uniform void distribution was simulated for several void volumes by placing an equal number of 3/4-in.-wide x 1/8-in.-thick, full-core-length aluminum strips in each of the 20 non-rodded fuel assemblies. Reactivity loss was calculated from the change in critical rod position using the differential control rod worth curve. The results of the uniform void distribution measurements as a function of total volume voided are shown in Figure 15. The measured coefficient is -41.5ϕ per percent decrease in moderator density or $-0.080\phi/\text{cm}^3$ of water replaced. The coefficient is independent of the total void volume in the core for the range investigated (about 11% moderator displacement).

For the radial importance measurement, the critical position was determined with a fuel assembly containing 24 full-core-length aluminum strips loaded at different representative lattice positions. The total volume of the aluminum strips in the voided fuel assembly was 864 cm^3 , which is equivalent to 1.66% of the total moderator volume in the core. For the measurement in the transient rod assembly position, 18 aluminum strips were used, equivalent to 1.24% of

Fig. 15 Reactivity loss vs moderator volume displaced by void strips for uniform void distribution.

the total moderator volume. The reactivity worth of the voided assembly was determined from the change in the critical control rod position from the unvoided critical position. A reactivity correction was made for small differences in fuel content between the special voided test assembly and the fuel assembly it replaced in the core. The reactivity worth of fuel used for this correction was evaluated at various fuel assembly positions by removing one of the central fuel plates of an assembly and replacing it with an aluminum dummy plate. The resultant loss in reactivity was determined by the change in the critical position. Figure 16 shows the void coefficient results obtained for various lattice positions.

The reactivity worth of 2-in.-high void strips distributed throughout the core as shown in Figure 17 was determined as a function of vertical position in the core. The total volume of the 96 strips used was 281 cm³, which is equivalent to 0.5% of the core moderator volume. The void strips were attached by nylon cord to the transient rod drive mechanism to permit control of the vertical position of the void strips. The strips were initially positioned with their center-lines 1 in. above the bottom of the active fuel region and were then raised in 2-in. increments. The reactivity worth as a function of vertical position was determined from changes in the critical position of the calibrated control rods. The results of this experiment are shown in Figure 18. The peak of the void worth curve has a value of approximately -80¢ per percent decrease in moderator density. The peak occurs at about 8.5 in. above the bottom of the fuel. The integrated average value of the void worth curve in Figure 18 is -41.7¢ per

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

Fig. 16 Local void coefficient in cents per percent moderator density change.

	CR-4		CR-1	
		TR		
	CR-3		CR-2	

Fig. 17 Location of void strips for vertical void importance measurement.

Fig. 18 Reactivity loss vs vertical void position.

percent decrease in moderator density, which is in good agreement with the value obtained from the uniform void coefficient measurement.

The void coefficient results reported here have not been corrected for absorption or scattering in the aluminum. PDQ [5] calculations, which have been made to determine the relative effectiveness of aluminum and air voids, show that air gives approximately a 7ϕ per percent larger negative void coefficient than aluminum. In addition, previous void coefficient measurements using both aluminum and air voids with various cores at Spert have shown that for undermoderated cores, such as the D-12/25 core, aluminum is a good simulated-void material. For example, the magnitude of the air-void coefficient was approximately 8% larger than the aluminum-void coefficient for the Spert I B-24/32 core [6], and no difference could be determined between the two types of voids with the Spert I oxide core [7].

E. Flux Distribution Measurements

During a typical short-period power excursion test, such as the type performed in the Spert experimental program, the reactor power may rise by a factor of 10^{10} or more during the course of the transient. The energy released during this rapid surge of power causes certain physical changes (moderator heating, fuel plate expansion, void formation, etc) to take place which not only

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

Therefore, as an aid in evaluating the validity of power measurements made during power excursion tests, a series of neutron flux distribution measurements were made in the Spert IV reactor. These measurements were made with the Spert IV D-12/25 core for three distinct core variants: (a) the normal operational (clean) core, (b) the operational core with the central region partially voided with aluminum strips worth 2\$ of reactivity, and (c) the operational core poisoned with boron-plastic strips worth 2\$ of reactivity and, hence, having the same critical control rod position as the voided core.

The voided core was used to simulate a core at the time of peak power during a power burst. In the actual test the control rods would be pulled to the supercritical position representative of the amount of inserted excess reactivity and the core would contain the voids formed during the power burst. In these experiments, the central region of the core was voided with 1/8- x 3/4-in. full-core-length aluminum strips with a total reactivity worth of 2\$. The positions of the voids within the core are shown in Figure 19.

Fig. 19 Spert IV core diagram showing positions of aluminum "void" strips.

The poisoned core was used to simulate the condition in which the control rods have been pulled to a super critical position representative of the amount of inserted excess reactivity and the reactor is on a positive period at a power level low enough that no significant amount of voids have formed in the core. In this case the core was uniformly poisoned by the insertion of six boron-impregnated, polyethylene strips in each of the non-rodded fuel assemblies of the core. These full-core-length strips have cross sectional dimensions of $1/2 \times 1/32$ in. and contained 1% boron by weight (amounting to 0.014 grams of boron per linear foot). The total reactivity worth of these strips was 2\$. The positions of the poison strips are shown in Figure 20.

Fig. 20 Location of boron-impregnated polyethylene poison strips in Spert IV fuel assembly.

For each core condition, the flux was determined from activation of 40-mil cobalt wires located at the core positions shown in Figure 21. The wires in the core were taped to the surfaces of the fuel plates along the center-lines and extended the full length of the assemblies. The wires in the reflector regions were held in place within a dummy fuel can as shown in Figure 22. In addition, one horizontal wire was taped to the outside of the fuel assemblies on the south side of the core at a position 10 in. from the bottom of the fuel plates. The wires indicated in Figure 21 as having cadmium sleeves were covered in two positions (12-1/2 and 18-1/2 in. from the top of the fuel plates) with 60-mil-OD $\times$ 1-1/4-in.-long cadmium tubes for the purpose of measuring the cadmium ratios. For each irradiation the reactor was operated at a power level of approximately 40 Kw for about 30 min.

Figures 23 through 29, 30 through 36, and 37 through 43 show the vertical flux profiles for the clean, voided, and poisoned core respectively, and Figures

IN FIGURES 22 THROUGH 43, POSITION DESIGNATIONS REFER TO WIRE POSITIONS SHOWN IN THIS FIGURE AS FOLLOWS: PROCEEDING NORTHWARD FROM THE CORE CENTER IN THE CENTRAL COLUMN OF FUEL ASSEMBLIES, THE WIRES ARE DESIGNATED E-5₁, D-5₁, D-5₂, C-5₁, AND C-5₂. PROCEEDING WESTWARD FROM THE CENTER IN THE CENTRAL ROW, THE WIRES ARE DESIGNATED E-5₁, E-5₂, E-4, AND E-3.

Fig. 21 Location of cobalt wires for measurement of neutron flux distribution.

44 through 49 show the horizontal flux profiles at three positions (at the flux peak, 6.5 in. above the peak, and 4.5 in. below the peak) in vertical planes through the core center in the north-south and east-west directions for the three core conditions.

Table V shows, for each power distribution, the average nvt values at the centerlines of various fuel assemblies (by applying symmetry, all fuel assemblies in the core are represented by these values), the average core nvt, the maximum core nvt, and the maximum-to-average nvt. These values in this table have been normalized to account for the variations in total irradiated time among the three irradiations.

TYPICAL OF DUMMY ASSEMBLY IN CORE
POSITION B-5.

Fig. 22 Spert IV dummy fuel can showing
locations of reflector-positioned flux wires.

Fig. 23 Vertical flux profiles in selected fuel
assemblies for clean core.

Fig. 24 Vertical flux profiles in selected fuel
assemblies for clean core.

Fig. 25 Vertical flux profiles in selected fuel
assemblies for clean core.

Fig. 26 Vertical flux profiles in selected fuel assemblies for clean core.

Fig. 27 Vertical flux profiles in selected fuel assemblies for clean core.

Fig. 28 Vertical flux profiles in selected fuel assemblies for clean core.

Fig. 29 Vertical flux profiles in selected fuel assemblies for clean core.

Fig. 30 Vertical flux profiles in selected fuel assemblies for voided core.

Fig. 31 Vertical flux profiles in selected fuel assemblies for voided core.

Fig. 32 Vertical flux profiles in selected fuel assemblies for voided core.

Fig. 33 Vertical flux profiles in selected fuel assemblies for voided core.

Fig. 34 Vertical flux profiles in selected fuel assemblies for voided core.

Fig. 35 Vertical flux profiles in selected fuel assemblies for voided core.

Fig. 36 Vertical flux profiles in selected fuel assemblies for voided core.

Fig. 37 Vertical flux profiles in selected fuel assemblies for poisoned core.

Fig. 38 Vertical flux profiles in selected fuel assemblies for poisoned core.

Fig. 39 Vertical flux profiles in selected fuel assemblies for poisoned core.

Fig. 40 Vertical flux profiles in selected fuel assemblies for poisoned core.

Fig. 41 Vertical flux profiles in selected fuel assemblies for poisoned core.

Fig. 42 Vertical flux profiles in selected fuel assemblies for poisoned core.

Fig. 43 Vertical flux profiles in selected fuel assemblies for poisoned core.

Fig. 44 Horizontal flux profiles through central N-S row of fuel assemblies for clean core.

Fig. 45 Horizontal flux profiles through central E-W row of fuel assemblies for clean core.

A comparison of the vertical flux profiles for the three core conditions indicates that the flux peak was shifted upward for the poisoned and voided cores, as would be expected from the changes in the critical positions of the control rods.

Fig. 46 Horizontal flux profiles through central N-S row of fuel assemblies for voided core.

Fig. 47 Horizontal flux profiles through central E-W row of fuel assemblies for voided core.

Fig. 48 Horizontal flux profiles through central N-S row of fuel assemblies for poisoned core.

Fig. 49 Horizontal flux profiles through central E-W row of fuel assemblies for poisoned core.

The horizontal flux profiles and Table V show that the addition of voids in the center of the core definitely depressed the neutron flux in that region. This is shown by the peak-to-average flux ratio of 1.95 for the voided core compared with 2.35 for the clean core. The poisoned-core neutron flux shows the same general shape as the clean core; however, the neutron flux values at each position in the core are higher for the poisoned core. It is felt that this difference is mainly due to the method used to poison the core. Although the core, as a whole, was fairly uniformly poisoned, the strips were positioned asymmetrically within the individual fuel assemblies, and, therefore, the neutron flux was depressed in the vicinity of the poison strips and peaked in the vicinity of the flux wires.

Figure 50 shows a comparison of the cadmium ratios for the different power distributions at 12 in. below the top of the fuel plates in a vertical plane through the center of the core in the east-west direction. The cadmium ratios for the various cores show that voiding the central region of the core hardened the flux in that region; however, it made little change in the outer region of the core and in the reflector. The cadmium ratio for the poisoned core differed little from the clean core within the core, but was slightly greater in the reflector.

TABLE V
NORMALIZED^[a] FLUX VALUES FOR THE THREE CORE VARIATIONS

Poisoned Core		Voided Core		Clean Core	
Wire	Average (nvt)	Wire	Average (nvt)	Wire	Average (nvt)
C - 7[b]	3.08×10^{14}	C - 7	2.97×10^{14}	C - 7	2.85×10^{14}
C - 6	4.13	C - 6	3.90	C - 6	3.47
C - 5 ₂	4.40	C - 5 ₂	4.15	C - 5 ₂	3.97
D - 3	4.40	D - 3	3.76	D - 3	3.62
D - 4	5.30	D - 4	4.79	D - 4	4.46
D - 5 ₂	5.71	D - 5 ₂	4.24	D - 5 ₂	5.18
F - 5	5.90	F - 5	4.63	F - 5	5.24
E - 3	4.27	E - 3	4.26	E - 3	4.17
E - 4	5.74	E - 4	4.46	E - 4	5.25
E - 5 ₁	6.84	E - 5 ₁	5.46	E - 5 ₁	6.21
		<u>Core Average</u>			
		4.50		3.97	
				3.95	
		<u>Maximum</u>			
		10.85		7.75	
				9.30	
		<u>Peak to Average</u>			
		2.41		1.95	
				2.35	

[a] All values have been normalized to operation at 36 Kw for 30 min.
[b] See Figure 21 for explanation of wire nomenclature.

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

Fig. 50 Cadmium ratios for the clean, voided, and poisoned cores 12 in. below top of fuel plates.

F. Calorimetric Power Calibrations

Consideration has been given to the possibility that the formation of voids in the central region of the core during a power burst could change the neutron flux at a chamber position relative to the average core flux by an amount large enough to make it a necessary consideration in the chamber calibration. In order to determine the extent that voids affect the current-to-power relationships of chambers in various positions and to determine chamber constants to be used in the transient testing program, calorimetric power calibrations were made for the unvoided operational core and the voided core. The voided core was the same as the core used in the voided flux measurements and is shown in Figure 19. In these experiments, the Spert IV calibration tank (7 ft diam x 8 ft high) was used as a calorimeter, and the bulk temperature of the reactor and moderator-reflector was raised approximately 10°C by operating the reactor at a constant power level as indicated by the operational linear-power channel. Since the heat

capacity of the complete system (core, moderator, and tank) could be computed, the actual operating power was determined from the observed temperature rise and the reactor operating time. Since the calibration tank is not a perfect calorimeter, cooling curves also were obtained following the heating to aid in estimating the heat losses during heating and during the time required to bring the system to isothermal conditions, at which time the final temperature measurements were made.

The change in temperature of the calorimeter from the initial temperature to the equilibrium temperature after nuclear heating was determined by use of the following instrumentation:

- (a) Chromel-alumel thermocouple in a water channel at the center of the core
- (b) Chromel-alumel thermocouple attached to the top of the center fuel assembly
- (c) Four-junction chromel-alumel thermopile located immediately under the center of the core
- (d) Seven-junction chromel-alumel thermopile in the reflector. The thermocouple junctions were distributed vertically from just above the bottom of the calibration tank to just under the surface of the water.

Using this technique, the current-to-power relationships were determined for the operational linear-power chamber and for a miniature chamber positioned in the reflector (Figure 51). During both experiments the operational linear-power

Fig. 51 Location of neutron chambers with respect to Spert IV core during power calibrations.

chamber current was held constant at 4×10^{-4} amp by suitable adjustments of the control rods. In each case the miniature neutron chamber current was initially 2.7×10^{-5} amp; however, at the end of the heating period (approximately 20 min later) the current had dropped to approximately 2.5×10^{-5} amp. It is believed that this effect was due to the buildup of the delayed gamma field which would have less of an effect on the miniature chamber (which is in the reflector neutron flux peak) than on the operational linear chamber which was approximately $10\text{-}1/2$ in. from the edge of the core.

From these experiments it was determined that for an operational linear-power chamber current of 4×10^{-4} amp the average reactor power during the tests was 173 ± 3 kw for the operational core, and 170 ± 3 kw for the voided core. On the basis of these results it is concluded that the power calibration of the neutron chamber is relatively insensitive to voids which would be formed in the central region of the core during a power burst.

The chamber constants which have been determined in these experiments will be used as references for the intercalibration of other chambers which will be positioned in and around the core following the removal of the calibration tank. This is necessary since removal of the calibration tank precludes calibration of these chambers by calorimetric methods.

G. Isothermal Temperature Coefficient Measurement

Isothermal temperature coefficient measurements have been made for the Spert IV D-12/25 core over the temperature range from approximately 20 to

35°C. In the measurements, the bulk-water temperature in the Spert IV calibration tank was raised from room temperature to 35°C by operating the reactor core at a steady-state power level of approximately 170 kw. The water was then cooled in steps by draining some of the hot water from the tank and refilling with cooler makeup water. After each addition of cooler water, the water was stirred continuously until the water temperature reached isothermal conditions.

The instrumentation used in this experiment was the same as that used in the calorimetric power calibration, a 4-junction thermopile below the core, a 7-junction thermopile in the reflector region, and single thermocouples in a moderator channel in the core and above the core.

At each temperature the critical position of the calibrated control rods was determined, and the reactivity difference between any two temperatures could then be determined by the change in the critical position of the calibrated control rods. By this technique the temperature coefficient was found to vary as the coefficient of expansion of water from approximately $-0.7\phi/^{\circ}\text{C}$ at 20°C to $-1.2\phi/^{\circ}\text{C}$ at 35°C.

Figure 52 shows the temperature coefficient vs temperature over the range of the measurement, and Figure 53 shows the temperature coefficient vs the expansion coefficient of water, β .

- THESE POINTS AND CURVE ARE RELATIVE DENSITY OF WATER. $\rho @ 4^{\circ}\text{C} = 1$.
- EXPERIMENTAL TEMPERATURE COEFFICIENT POINTS. UNCERTAINTY LIMITS ARE BASED ON $\pm 0.25\phi$ IN $\Delta\rho$ IN THE RAW DATA.

Fig. 52 Isothermal temperature coefficient as a function of temperature.

Fig. 53 Expansion coefficient of water vs isothermal temperature coefficient of reactivity.

TABLE VI

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

Core Designation	Spert IV D-12/25 Core
Fuel Assembly Type	Spert D
Number of Fuel Plates Per Assembly	12
Number of Fuel Assemblies	20
Number of Control Rod Assemblies	4
Number of Transient Rod Assemblies	1
Core Configuration	5 x 5 array
Mass of U-235	
Critical Loading	3.0 kg
Operational Loading	3.75 kg
Available Excess Reactivity (20°C)	5.3 \$
Control Rod Worth at Cold Clean, Critical	1.07 \$
Core Shutdown Margin	15 \$
Average Neutron Flux Per Watt	$6.1 \times 10^6 \frac{\text{nv}}{\text{watt}}$
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¢/% decrease in moderator density
Maximum Measured Local Void Coefficient	-60¢/% decrease in moderator density

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