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NUCLEAR START-UP OF THE
SPERT II REACTOR
WITH HEAVY-WATER MODERATOR

MASTER

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April 20, 1962

PHILLIPS
PETROLEUM
COMPANY

ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION

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SPECIAL POWER EXCURSION REACTOR TESTS

NUCLEAR START-UP OF THE
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By

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Errata for IDO-16762

1. In Table IV (p. 27) substitute -2.79 for -3.79 in the 4th column, 6th row.
2. In Table VII (p. 34) substitute -0.017 for -0.022 in the 6th column, 6th row.
3. In Fig. 33 (p. 35), in the upper right box, substitute -0.030 for -0.026.
4. In Table VIII (p. 39), line 7, the "Average Thermal Flux Per Watt: should be 6.1×10^6 (not 10^{16}).

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ABSTRACT

A brief description is presented of the Spert II facility along with a description and the results of the static experiments with the initial D₂O-moderated core to be used in self-limiting power excursion tests. The core is loaded with three-inch square by 24-inch long plate-type, highly enriched uranium-aluminum fuel assemblies on six-inch centers. A core loading of 3.7 kg of U-235 has an available excess reactivity of approximately \$8.1 at 70°F and atmospheric pressure. The temperature coefficient of reactivity varies from about -2.0 ϕ /°F at 80°F to about -4.4 ϕ /°F at 300°F. The reactivity coefficients of temperature and void may be expressed as a density coefficient of -94 ϕ /(per cent density change) for uniform density changes or -7 ϕ /(per cent density change) when confined to the total water volume within the fuel assemblies. The reduced prompt neutron lifetime, ℓ/β_{eff} , was determined to be 0.11 sec.

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I. INTRODUCTION

A primary purpose of the Spert program, which is conducted by Phillips Petroleum Company as a part of the Atomic Energy Commission's Reactor Safety Program, is the determination of the kinetic behavior of various classes of reactor systems, particularly under conditions similar to those encountered in accidents. One of the facilities used in this program is the Spert II reactor, which is the third of a series of heterogeneous, water-cooled and -moderated reactors constructed in the Spert reactor area of the National Reactor Testing Station in Idaho. One of the initial objectives of the Spert II experimental program is a study of the influence of the prompt neutron lifetime on reactor power excursions and reactor safety by use of various moderators and reflectors, especially light and heavy water. The reactor vessel is designed to facilitate changes in core types and configurations. The reactor vessel and coolant system are designed for operation at pressures to 375 psig, temperatures to 400°F and coolant flow rates to 20,000 gpm. Both upward and downward coolant flow through the core is possible; however, since the facility is intended primarily as a transient test facility, no power removal equipment is included.

This report presents a brief description of the Spert II facility, a description of the experimental work and results of the static experiments with the initial D₂O-moderated core to be used in self-limiting power excursion tests. The core is comprised of 3-inch square by 24-inch long, highly enriched, uranium-aluminum plate-type fuel assemblies, which are spaced on 6-inch centers. The experimental procedures are presented in the body of the report and the results are summarized at the end of the report.

II. BRIEF DESCRIPTION OF THE FACILITY

A. Introduction

The Spert II facility consists of the reactor area and a control center area that is located approximately 1/2 mile from the reactor. The reactor building encloses the reactor vessel, coolant flow systems, heavy water system, and the necessary auxiliary equipment for operating a reactor of this type. The reactor vessel and coolant flow systems are designed for operation at pressures to 375 psig, temperatures to 400°F, and flow either up or down through the core at rates to 20,000 gpm with either light or heavy water. The control center building houses those controls necessary for remote operation of the reactor and plant during nuclear operation. The following sections describe those reactor and plant components that were unique to the initial D₂O core and are pertinent to this report. More complete descriptions of the complete facility have been published previously^(1,2).

B. Reactor Components

The Spert II reactor assembly, Figure 1, consists of the reactor core with the control rods and drives, a transient rod with its drive, the reactor core structure, and the reactor pressure vessel with a full-diameter flanged top head. Inlet and exit coolant piping is attached to the reactor vessel bottom head.

1. Reactor Vessel

The reactor vessel, a picture of which is shown in Figure 2, is a stainless steel clad, carbon steel vessel, 10 ft ID by about 16 ft high. The vessel is designed for a pressure of 375 psig at 400°F and is equipped with a full-opening hemispherical head upon which is mounted a 42-inch diameter top plug containing the control rod drives. During normal shutdown operations only the top plug is removed; however, the full opening head may be removed to change or remove the larger internal structure.

The vessel bottom head contains a 42-inch diameter spool extending downward from the head. The spool is equipped with electric immersion heaters having a capacity of 200 kw for heating the moderator-coolant to the desired temperature. This spool also contains a 24-inch coolant nozzle. In addition to the spool, the vessel bottom head is penetrated by four additional nozzles for the introduction or removal of coolant, depending upon the desired direction of coolant flow through the core.

Flanged nozzles for the removal of instrument leads are provided in the top head and at several elevations around the vessel shell. In addition to the electric heaters in the bottom well, 400 kw of heating capacity is supplied by side-arm heaters in the vessel shell.

Fig. 1 Spert II Reactor Vessel Assembly

Fig. 2 Photograph of the Spert II Reactor Vessel

2. Internal Structure

The reactor core is supported by an internal structure from the bottom of the vessel. An aluminum flow skirt, 42 in. in diameter and 1/2 in. thick, directs the flow of coolant through the core and separates the core from the balance of the vessel to a height of about 12 in. above the top of the fuel. Coolant entering the bottom well flows either upward through the core, over the top of the core skirt, downward outside the core skirt and exits from the bottom of the vessel or vice versa.

Figure 3 is a cross-sectional view of the reactor core support grid. This grid is divided into four symmetrical quadrants by a 3/4-in. thick cross of aluminum and water. In the center of the cross is a blade-type transient rod containing a poison in the bottom section.

The reactor is controlled by eight control rods of the simple poison type which operate in guides in the positions labeled "C". The grid has 88 positions available for the loading of fuel assemblies. Unoccupied fuel positions are blocked by filler pieces or plugs during experiments requiring coolant flow.

Fig. 3 Spert II Reactor Cross Section

3. Fuel Assemblies

The fuel assembly used in the initial operational core is the Spert BD-type assembly which is shown in Figure 4. Basically, the fuel assembly consists of a 3-in. square by 27-5/8 in. long aluminum retaining can; an aluminum lower end box with locking lugs; two grooved side plates; a lifting bail; and 22 removable fuel plates. The lifting bail and lower end box perform the secondary function of holding the plates in the assembly. Complete disassembly of the fuel assembly is easily accomplished by removal of eight screws which fasten the lifting bail and end box to the retaining can.

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

4. Control Rods and Transient Rod

The control rods are a simple poison type, with cadmium in the upper section and an aluminum guide tube for the lower section. The poison section consists of a hollow stainless steel tube, 1.5 in. in diameter, with cadmium and stainless steel laminated on the outer surface. A pair of control rods is joined by a yoke to form a control rod assembly, Figure 5, which is driven by a single drive mechanism.

The transient rod, Figure 6, is a blade-type rod with poison in the lower section and aluminum in the upper section. The poison section is 0.020 in. cadmium sheet clad with aluminum.

5. Control Rod Drive Mechanisms

Five rod drive mechanisms are mounted on the reactor top plug as shown in Figure 1. Four of the drives position the control rods and the fifth positions the transient rod. Although the function of the two rod drive types is somewhat different, the mechanical design is essentially identical.

A pictorial drawing of a control rod drive assembly is presented in Figure 7. The drive consists basically of a cylinder, a piston, a shock absorber, a drive screw and a motor. No direct connection exists between the piston, which is mechanically joined to the control rod, and the drive screw. Air pressure (Hold-Air) on the bottom side of the piston maintains the piston in contact with the screw, thus permitting positioning the control rod by means of the drive motor and screw. Air pressure (Scram-Air) is also maintained on the upper side of the piston. The Scram-Air is at slightly less pressure than the Hold-Air, yet sufficient to drive the control rod in against reactor pressure. Rapid insertion of the rods (Scram) is accomplished by exhausting the Hold-Air. An air reservoir is attached to the upper cylinder of each drive to provide a

Fig. 4 Spent BD-Type Fuel Assembly with a 22 Fuel Plate Loading

Fig. 5 Spert II Tubular Control Rod Assembly

Fig. 6 Spert II Transient Rod Assembly

Fig. 7 Pictorial Drawing of Spert II Control Rod Drive Mechanism

low impedance source of scram air and to protect against certain types of pneumatic failures. Spring-locked latches are provided to hold the rod pistons down once they are seated. A key switch on the reactor console must be operated to unlatch the pistons and permit rod withdrawal.

The transient rod drive is similar to the control rod drives except that in this case a latch is provided to secure the rod piston, and hence the rod itself, to the rod drive screw until unlocked by a key switch on the reactor console. Inasmuch as the transient rod is dropped to insert reactivity and initiate a power excursion, the air pressure on the top of the transient rod piston (Fire-Air) may be applied only as a part of the arming sequence immediately prior to the rod drop.

The drive motors are three-speed, 1 hp, 480 volt, 3 phase, induction motors. A speed reducing gear box permits control rod speeds of approximately 0.1, 0.2, and 0.3 in./sec.

Selsyn transmitters are geared to each drive motor shaft to provide rod position information. The selsyn receivers at the console drive mechanical position indicators, which read in hundredths of inches of rod travel.

6. Reactor Control System

The reactor control system will be described by reference to those features which are unusual or which are pertinent to an understanding of this report. A picture of the reactor control console is shown in Figure 8.

Fig. 8 Photograph of the Spert II Reactor Control Console

The reactor control instrumentation basically consists of two boron-lined ionization chambers used for neutron level determination under operational conditions (above ~ 0.1 watt reactor power) and two boron-lined neutron pulse chambers for low-level neutron measurements (up to ~ 1.0 watt reactor power). The outputs of the ionization chambers are displayed on two neutron level recorders (one linear power and one log power) mounted in the reactor console. The outputs of the pulse chambers are displayed on scalers mounted in the reactor console; in addition, one of these provides a signal to a count-rate meter mounted in the console.

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

During normal shutdown operations the top plug, shown in Figure 9, which contains the control rod drives, may be moved to the top head storage area (drydock). The control system air and electrical connections are mounted on a jib crane which swings from the reactor vessel to the drydock area. This allows operational checkout of the complete control system when it is in drydock.

C. Process Systems

A schematic flow diagram of the Spert II process system is shown in Figure 10. The principal components are the primary circulating loop, a pressurizer vessel with helium pressurizing system, and a heavy water system.

1. Primary Coolant System

The primary coolant circulation loop is designed for use with either light or heavy water. A crossover in the piping permits the circulation of coolant either upward or downward through the reactor core. The primary loop piping is stainless steel, type 304L, and is designed to assure maximum integrity of the primary loop and minimum holdup of the coolant. Where feasible, valves have been located in vertical runs to minimize liquid holdup. When heavy water is in use

Fig. 9 Photograph of Control Rod Drives
and Jib Crane

Fig. 10 Spart II Process Flow Diagram

in the reactor vessel and coolant loop, the light water system is completely isolated from the loop by removing connecting spools.

The two primary coolant pumps are conventional horizontal, single stage, centrifugal pumps having a maximum capacity of 10,000 gpm each and a net differential head of 175 ft of water. A unique feature of the system is the fluid coupling of the pumps to the electric motors for adjustment of pump speed. This allows the selection of any flow rate from a maximum of 10,000 gpm to a minimum of 2,000 gpm. An 8 in. flow control valve permits further flow reduction to 200 gpm. The flow rate is measured over the entire range by either of two flow tubes.

2. Pressurizing System

The reactor and coolant system are pressurized by means of the pressurizer vessel, utilizing high pressure helium. The pressurizer is directly connected to the reactor vessel and is mounted in relation to the reactor vessel approximately as shown in Figure 10.

In addition to system pressurization, low pressure helium is utilized to purge and/or blow down the system as required. Helium released to the atmosphere for any purpose, such as depressurization, is first directed through the cold trap to recover D_2O vapor from the helium.

3. Heavy Water System

The heavy water system is comprised of a number of small systems including the heavy water storage vessel, a transfer system, a clean-up loop and a D_2O recovery system.

Heavy water is stored in a 14,500 gal stainless steel tank, which is provided with a helium gas blanket at a positive pressure of about 0.5 in. of water. Heavy water may be transferred from the storage tank to the reactor vessel, or vice versa, by means of a 75 gpm canned motor pump. A forced draft fin-tube heat exchanger is incorporated in the transfer loop in order to cool the heavy water in the vessel below the boiling point before returning it to storage, or before opening the vessel after operation at elevated temperatures. The heat exchanger is capable of reducing the temperature of the D_2O from 400°F to 100°F in about 12 hours.

Heavy water purity is maintained by a clean-up loop consisting of a 5 gpm canned motor pump, two stainless steel filters which pass particles of 35 microns or smaller and two mixed-bed ion exchange columns. Heavy water located in storage or in the primary system may be directed through the loop and returned as desired. The ion exchange columns contain about 1 ft³ of mixed anion-cation resin each and are valved for removal without draining. Secondary equipment includes pH and conductivity meters as well as a sampling station. A 130 cfm vacuum pump capable of reducing the pressure to about 40 microns is provided for drying the system; a necessary procedure to prevent contamination of the D_2O with H_2O from previous operation with light water.

III. PREPARATION FOR NUCLEAR START-UP

A. Introduction

Construction of the Spert II facility was completed on February 1, 1960. From this date until early March, the installation and checkout of equipment proceeded and various minor plant modifications were undertaken. From March until August 1960, numerous exploratory static tests were performed with light-water-moderated cores.^(3,4) Following these tests, the system was converted for D₂O-operation and a program of investigation was undertaken on D₂O-moderated cores. This section describes those activities which were directly related to the nuclear start-up of the reactor with the core configuration which was to be used for the initial transient testing program in D₂O. The core is designated the expanded D₂O core.

B. Reactor Core Coordinate System and Core Designation

The 96 lattice positions in the Spert II reactor have been assigned numbers as shown in Figure 11. The coordinate system can be described by considering the core to be divided into four quadrants (N, E, S, and W) by the cross. Numbers are assigned to each lattice position by considering each quadrant separately as the first quadrant of a normal cartesian coordinate system with the origin at the center of the core.

A given position is designated by quadrant and by X- and Y-coordinates within the quadrant; e.g., W4-2. Thus, the control rods are located in positions 2-2 and 3-1 in each quadrant and like-numbered positions are symmetric. Control rod pairs are designated by number: Number 1 in the North Quadrant, Number 2 in the East Quadrant, etc.

Fig. 11 Spert II Core Coordinate System

In describing core loadings in this report, reference will be made to the B-24/12 or BD-22/24 expanded D₂O core, etc. The letters in this nomenclature indicate the fuel assembly type, the first number indicates the number of fuel plates per assembly, and the second number indicates the number of fuel assemblies of this type in the core. The designation "expanded D₂O core" indicates that the fuel assemblies are placed on six-inch centers and separated by a three-inch water space as contrasted to their being positioned

immediately adjacent to one another as is the case for a close-packed core.

C. Fuel and Poison Content of Fuel Assemblies and Control Rods

Before proceeding to load fuel in the reactor, the uniformity of the U-235 in the fuel and the cadmium in the control and transient rods was determined. This ensured that the fuel was uniformly distributed in the plates so that no hot spots would result from localized inhomogeneities in the fuel plates and also ensured the presence and location of the poison in the control and transient rods. The investigation made use of the neutron scanning technique and equipment previously developed at the MTR⁽⁵⁾. The results indicated that the total fuel content of the various plates was the same to within 2.3 per cent, and that the distribution of poison in the control and transient rods deviated less than 4 per cent over the length of the poison sections.

D. Preparation and Filling of the System with Heavy Water

Two hundred and ninety eight drums, each containing approximately 55 gallons of heavy water, were received from the Savannah River Plant for use in the Spert II system. Each drum of the shipment was sampled with an analysis determined on composites of ten samples. D₂O purity was determined on each composite sample and a complete analysis (conductivity, pH, chlorides, turbidity, oil, and metal content) determined on four of the composites. The D₂O had a conductivity of 3.73 μ mhos, a pH of 7.02, 0.164 ppm chlorides and no metal or oil content. An average isotopic purity of 99.77 mole % was determined for the composite in the D₂O storage tank.

The contents of 263 drums less samples and heel, (130,932 lbs) were transferred to the D₂O storage tank. Weighings were made on each drum before and after transferring for an accountability record and to accurately calibrate the storage tank.

In preparation for the critical studies, it was necessary to prepare the plant for its initial operation with D₂O⁽⁶⁾. Since the primary circulating system was not to be used for these tests or the initial transient tests, the reactor vessel was isolated from the remainder of the system to minimize the system inventory and possible D₂O contamination and losses. All light water piping was isolated by removing spool pieces and installing blind flanges.

The reactor tank, clean-up piping system, outdoor heat exchanger and the D₂O transfer pump piping vacuum dried to remove any residual light water and/or vapor remaining in the system. Whenever possible, visual inspections were made to determine dryness of the various components. The piping was rinsed with 780 gal of D₂O which was then removed and stored in stainless steel drums for future use as a rinse. There was no detectable change in the isotopic purity of this D₂O after the rinse. Following this preparation, 6400 gal of heavy water were transferred to the reactor vessel from the D₂O storage tank. This provided a D₂O level of approximately 50 inches above the core.

E. Checkout of Reactor Control System

Although the control rod system had been given a detailed checkout in drydock prior to operation with light water, another checkout was performed with the control system prior to the start of the D₂O core tests. The control rod position indicators at the control room nuclear console were set at zero when the lower edge of the poison in the control rods was aligned with the lower edge of the fuel in the fuel assemblies. Each limit- and indicating-switch was set to actuate at the proper position.

F. Start-up Neutron Source

In order to facilitate the checkout of the start-up neutron instrumentation and initial critical loading, a Ra-Be neutron source with a yield of approximately 1.4×10^3 neutrons per second was placed in the reactor vessel. The source was contained in a holder approximately 1/2 in. long. For the initial criticality tests, the source was mounted in a dummy transient rod which positioned it at the approximate center of the reactor core structure.

G. Installation and Checkout of Start-up Neutron Detectors

Three neutron pulse-counting systems and two compensated ion chambers were used during the start-up of Spert II with the expanded D₂O moderated core.

Each pulse system as shown in Figure 12, consists of two B¹⁰-lined pulse chambers, two pre-amplifiers, two power supplies, a single linear

Fig. 12 Block diagram of Typical Spert II Start-up Pulse System

amplifier, and a single scaler. Switching provisions were available at the control room nuclear console for independent selection of either chamber. Prior to start-up of the reactor, each pulse-counting system was calibrated to obtain a curve of the counting rate as a function of counter voltage. Each chamber was inserted in a block of paraffin and exposed to a Po-Be neutron source. A typical curve of counting rate as a function of voltage is shown in Figure 13. Once the operating voltage had been selected for the counter, discrimination data were obtained by exposing the chambers separately to the neutron source used above and to an 80 r/hr gamma source. Typical curves are shown in Figure 14.

The pulse chambers were placed near the core in the positions shown in Figure 15. Chamber 1-A was placed in an aluminum canister and positioned on a bracket outside the flow skirt near position W3-5; Chamber 2-B was similarly placed in an aluminum canister and then inserted in a modified filler piece in position S6-1; Chamber 3-F was also in an aluminum canister and placed on a bracket outside the flow skirt near S5-3; Chamber 1-D was placed 12 inches from the mouth of the south thimble which extended from the outer vessel wall to the flow skirt; Chamber 2-E was placed 2 ft west of the south thimble against the outside of the vessel wall.

Fig. 13 Neutron Count Rate vs Counter Voltage for Typical B¹⁰-Lined Pulse Counter

Fig. 14 Neutron and Gamma Ray Discrimination Curves for Typical B¹⁰-Lined Pulse Counter

Fig. 15 Location of Start-up Neutron Chambers

Two B^{10} -lined compensated ion chambers were used during the start-up to provide log and linear neutron level information at high multiplications and for operation near criticality. The ion chambers were used uncompensated for the nuclear start-up. The linear ion chamber was placed inside the thimble as close to the flow skirt as possible in order to give the maximum indication of neutron flux. The log ion chamber was inserted in a paraffin block and hung on a bracket at the mouth of the thimble (see Figure 15). Neither of these chambers showed detectable readings when positioned near the reactor vessel with the start-up neutron source in the D_2O filled vessel and all control rods seated. The ion chambers had previously been calibrated for neutron sensitivity and gamma compensation in the MTR thermal column and gamma facility.

IV. INITIAL APPROACH TO CRITICALITY - EXPANDED D₂O CORE

The nuclear start-up of the Spert II reactor with the expanded D₂O core was begun on August 17, 1960. The first step in the program was the establishment of the clean cold critical loading with B-24 assemblies. No mockup critical experiments or exponential measurements had been made for this core.

Previous sub-critical studies with various core configurations in heavy water at Spert II led to a decision to procure BD-type assemblies for use in the reactor. The primary advantage of the BD-type assemblies over the B-type in heavy water is a reduction in the effective neutron absorption in the end boxes and lower reflector region. However, uncertainty in the delivery of BD-type assemblies resulted in the use of available B-24 assemblies in the establishment of an initial criticality with an expanded-type core. Such a procedure facilitated critical mass predictions during the later start-up with BD-22 assemblies in the same configuration.

Plots were maintained of the reciprocal counting rate as a function of fuel loading and rod position. These plots were extrapolated to predict the critical mass of the system prior to each loading step. Since these predictions changed as the loading was increased, counting rates were carefully monitored during the actual fuel insertions in order to detect the possible approach to criticality. Since the sensitivity of this method is dependent on counting statistics and since the counting rates are lowest with all rods inserted, fuel was added with one control rod withdrawn. For all cases, the withdrawn rod had a reactivity worth greater than that of the fuel assembly being added. Therefore, if the core was subcritical with a rod withdrawn, the system could always be made subcritical during the loading of a fuel assembly by scrambling the rod.

Source-level counting rates were observed with no fuel in the reactor vessel and sufficient heavy water in the vessel to provide an upper reflector of 50 inches above the top of the anticipated core. The tubular control rods and transient rod were in place. However, the filler pieces around the interior of the flow skirt and the filler pieces in the cross were removed for the start-up.

The loading sequence used for the B-24 expanded D₂O core is shown in Figure 16. The total mass of U-235 following each load is given in Table I.

The reactor became critical on August 17, 1960 with the control rods withdrawn 16.58 inches for load No. 5. A plot of inverse multiplication versus the mass of U-235 loaded is shown in Figure 17 for three of the start-up instrumentation channels. The extrapolation from Load No. 4 yielded an estimated critical mass of 2.86 kg of U-235.

TABLE I

TOTAL MASS OF U-235 IN THE CORE FOLLOWING EACH INITIAL LOADING

<u>Load No.</u>	<u>Total Mass of U-235 (kg)</u>
1	0.672
2	1.344
3	2.016
4	2.688
5	3.360

Fig. 16 Loading Sequence for Initial Criticality of the B-24/20 Expanded D₂O Core

Fig. 17 Inverse Multiplication vs Mass of U-235 Loaded in the Core with Rods Withdrawn for Three Typical Start-up Instrumentation Channels.

V. STATIC MEASUREMENTS

A. Operational Core Loading

On September 7, 1960, an operational core loading⁽⁷⁾ with BD-type fuel assemblies was obtained. The initial core loading investigated was one consisting of 20 BD-type assemblies with 22 fuel plates per assembly. The core was loaded in the same geometry as the previously discussed B-24/20 Expanded core loading. This core designated the BD-22/20 Expanded D₂O core, contained 3.08 kg of U-235 and was critical with the control rods withdrawn 19.99 inches. This may be contrasted with the B-24/20 Expanded D₂O core, which contained 3.36 kg of U-235 and was critical with the control rods withdrawn 16.58 inches.

In order for a heavy-water moderated core of this type to be useful for transient testing at reactor periods of 20 milliseconds or less at ambient temperature and atmospheric pressure, it was desirable to have an available excess reactivity of approximately seven to ten dollars. . . . The BD-22/20 Expanded D₂O core had an available excess reactivity of approximately \$2.25. To achieve the additional reactivity required for transient testing, four more assemblies were loaded in the 1-6 grid positions. This core, designated the BD-22/24 Expanded D₂O core and shown in Figure 18, contained 3.7 kg of U-235 and was critical with the con-

trol rods withdrawn 13.58 inches. The available excess reactivity was approximately \$8.10 at ambient temperature and atmospheric pressure with no filler pieces in the core or cross. This provided a core that would be satisfactory for subsequent transient testing purposes.

B. Critical Mass Determination

The critical mass of the expanded D₂O core configuration with BD-22 assemblies was determined by two separate methods. In each case the D₂O temperature was 24°C, the D₂O was 50 inches above the core, the system was at atmospheric pressure, and the critical mass was determined by plotting the value of the excess reactivity versus the mass of U-235 in the core for various configurations and extrapolating or interpolating the curve to zero excess reactivity.

○ CONTROL RODS
▨ FUEL ASSEMBLIES

Fig. 18. Operational Core Loading
Configuration of the BD-22/24
Expanded D₂O Core

The first of these determinations was made by maintaining the fuel plate configuration of the fuel assemblies constant and changing the mass of the fuel in the core and the core geometry by removal of fuel assemblies or fuel plates from the periphery of the core. This was called the "constant fuel density" case. The second determination was made by removing fuel plates uniformly from each fuel assembly in the core. This maintained the core geometry, or size, constant and is called the "constant geometry" case. The starting point for both of these experiments was the operational BD-22/24 expanded D₂O core.

For the "constant fuel density" case the core was initially loaded with a full complement of 24 assemblies with 22 fuel plates per fuel assembly. The position of the control rods with the reactor critical was established and differential rod worth data were obtained by the period method. Succeeding steps in the experiment involved first the removal of the fuel assemblies from the 1-6 grid positions, secondly the removal of six fuel plates from each of the fuel assemblies in the 5-2 positions, and the final step was the removal of two fuel plates from each of the assemblies in the 3-4 positions. For each step, the position of the control rods with the reactor critical was established and additional differential rod worth data were obtained.

With the information available from previous experiments on the differential control rod worth at various positions and that obtained from this experiment, an integral rod worth curve was constructed and the excess reactivity for each configuration was determined. The information determined in this manner is given in Table II along with the minimum radius circle that would enclose all of the fuel for each case. A plot of excess reactivity versus mass of uranium in the core is shown in Figure 19. Extrapolation of the line drawn through the data points to the zero excess reactivity line provides a reasonable indication of the critical mass for the constant fuel density case. The value obtained by this extrapolation was 2.8 kg of uranium.

TABLE II

CRITICAL MASS DETERMINATION, "CONSTANT FUEL DENSITY"

<u>Mass of U-235 (kg)</u>	<u>Control Rod Posi- tion at Critical (in.)</u>	<u>Radius (in.)</u>	<u>Excess Reactivity (\$)</u>
3.696	13.58 1/4	18 3/4	8.10
3.080	19.99	16 3/4	2.27
2.912	22.49 1/2	15 1/2	0.72
2.856	23.49	15 1/4	0.23

Fig. 19 Excess Reactivity vs Mass of U-235 for the "Constant Fuel Density" Critical Mass Determination

The second case, that of the "constant geometry" was done in the following manner: The rod worth information obtained in the first step of the "constant fuel density" case was used as the base point for this determination. The second step involved removal of four fuel plates from each assembly, and determination of the position of the control rods with the reactor critical. Differential rod worth data were also obtained at this point. Two more fuel plates were then removed so that each of the 24 fuel assemblies in the core had 16 fuel plates per assembly. This core was sub-critical but an estimate of its reactivity was made by determining a counting rate with the control rods fully withdrawn. A reactivity estimate was obtained by use of this count rate and the C_0 count rate determined during the critical loading sequence.

that for the constant fuel density case. A summary of the data obtained is given in Table III.

This information for the "constant geometry" case was handled in the same manner as

TABLE III

CRITICAL MASS DETERMINATION, "CONSTANT GEOMETRY"

<u>Mass of U-235 (kg)</u>	<u>Control Rod Position at Critical (in.)</u>	<u>Excess Reactivity (\$)</u>
3.696	13.58 1/4	8.10
3.024	18.51 1/2	3.37
2.688	Sub-Critical	-0.99

Fig. 20 Excess Reactivity vs Mass of U-235 for the "Constant Geometry" Critical Mass Determination

A plot of the excess reactivity versus mass of uranium for the "constant geometry" case is shown in Figure 20. The point at which this curve intercepts the zero reactivity line provides a reasonable estimate of the critical mass for the "constant geometry" case. The estimate of the critical mass for this case was 2.74 kg. The minimum radius that would include all the fuel is 18-3/4 in.

From this experiment it is noticed that the critical mass for the "constant fuel density" case is approximately 100 grams larger than for the "constant geometry" case. This result is reasonable since the "constant density" case results in a smaller core with a larger neutron leakage and a larger disadvantage factor within the fuel assemblies as compared to the "constant geometry" case.

C. Control Rod Worth

The reactivity worth per unit length of the control rods from the critical rod position to the fully withdrawn rod position was measured in order to determine the amount of available excess reactivity in the core, and to permit determination of reactivity effect attributable to temperature, voids, etc.

Eight tubular control rods connected in pairs and attached to the four control rod drives were used in this core. The calibration described in the following paragraphs was performed by operating the eight control rods as a bank.

The calibration of the control rods was accomplished by the period method. Several reactivity effects, i.e., fuel removal, voids, temperature, etc., were used as shims to permit calibration of the control rods over an extended range. At each position for which the control rods were calibrated, power rises with periods in the range of 140 to 20 seconds were initiated. The excess reactivity above delayed critical was determined from the period measurements by use of the inhour relationship which is independent of the prompt neutron lifetime in this period region.

The differential rod worth data obtained during the course of the following experiments are shown in Figure 21:

1. Critical Loadings and Fuel Assembly Modification Tests
2. Minimum Critical Mass Experiment
3. Fuel Assembly Reactivity Worth Experiment
4. Temperature Coefficient Experiment
5. Void Coefficient Experiment
6. ℓ/β_{eff} Experiment

Integration of the differential curve gives the integral reactivity worth of the control rods shown in Figure 22. All of the period data were obtained through use of the operational linear ion chamber and a strip chart recorder.

A statistical analysis was made of the rod worth values obtained about one average rod position during the void coefficient experiment. The maximum deviation from the average was ± 0.05 \$/in. and the standard deviation was ± 0.03 \$/in.

Fig. 21 Differential Control Rod Worth Curve

Fig. 22 Integral Control Rod Worth Curve

Fig. 23 Reactivity Worth of Fuel Assemblies

D. Fuel Assembly Reactivity Worth

Following the achievement of an operational loading, an experiment was performed to determine the reactivity worth of each individual fuel assembly in a quadrant(7). This was done to provide information useful in establishing operational safety procedures such as the reactivity effects associated with certain fuel loading or unloading sequences. The experiment also yields information on the relative importance of the reactivity effects of the individual fuel assemblies which is useful in analyzing transient test results.

The measurement was made by removing a fuel assembly from a particular position and establishing the control rod position for criticality. Then by use of the control rod worth data, the reactivity worth of the particular fuel assembly was determined.

The reactivity worth of individual fuel assemblies in the west quadrant is given in Table IV. Shown in Figure 23 is the reactivity worth of each fuel assembly in dollars along with the relative reactivity worth of each assembly normalized to the maximum which is in the 1-2 grid position.

TABLE IV

FUEL ASSEMBLY REACTIVITY WORTH

<u>Position of Fuel Assembly Removed</u>	<u>Control Rod Position at Critical (in.)</u>	<u>Excess Reactivity (\$)</u>	<u>Worth of Fuel Assembly Removed (\$)</u>
	13.58 1/4	8.10	
W1-6	14.65 3/4	6.91	-1.19
W5-2	15.10 1/4	6.45	-1.65
W3-4	15.04 1/2	6.53	-1.57
W1-4	15.16 3/4	6.39	-1.71
W3-2	15.59	5.97	-2.13
W1-2	16.27	5.31	-3.79

The insertion of a BD-22 fuel assembly loaded with dummy plates into the W1-6 position in place of a regular BD-22 assembly resulted in a reactivity loss of 1.63 dollars. Removal of the dummy assembly resulted in a reactivity gain to the system of 0.44 dollars.

E. Flux Distribution

The thermal neutron flux distribution was measured in the BD-22/24 expanded D₂O core by the use of gold foils.⁽⁸⁾ A total of 280 gold foils (5/32 inch diameter x 0.005 inches thick), located throughout the west quadrant of the core and other selected locations in the core and reflector, was irradiated. The foils within fuel assemblies were taped to the fuel plates and fuel assembly cans with mylar tape. The foils in the water spaces and reflector were taped to magnesium strips. For the vertical profiles, thirteen foils were attached at two-inch intervals over the active portion of the central plate of each fuel assembly in the west quadrant of the core. A series of foils placed at the centerline of the core was used

Fig. 24 Core Plan-Form and Foil Loading Pattern

to obtain several horizontal flux profiles. Several cadmium-covered foils were located throughout the core and reflector to provide measurements of the cadmium ratio. The estimated uncertainty in foil position was $\pm 1/8$ in. for those foils taped to fuel plates and assembly cans. The foils located in the water spaces and reflector had an estimated uncertainty in position of $\pm 1/4$ in. The core plan-form and foil loading pattern is shown in Figure 24.

The foils, which were activated at a power level of approximately 5 watts for 100 minutes, had an average counting rate of about 20,000 counts per minute. The activity of the individual foils was determined in a proportional flow counter which had been intercalibrated with a similar counter at the MTR that was calibrated in terms of absolute nvt per count. After correction of the data for radioactive decay, they were then converted to absolute nvt.

Cadmium ratios obtained from the cadmium-covered foils in conjunction with bare foil measurements were used to determine the thermal neutron flux. The cadmium ratio was 2.2 in assembly W1-2 and 2.8 in assembly W5-2. A cadmium ratio of 3.6 was measured in the center of the water channel on the north side of W5-2. In the reflector, twenty inches outside the flow skirt, a cadmium ratio of 4.4 was measured.

Vertical flux profiles shown in Figures 25, 26, and 27 for the six assemblies in the west quadrant are also applicable for the other three quadrants since the core has polar symmetry. A profile of the horizontal flux depression through an assembly was measured along the horizontal centerline of the fuel assembly in the W3-4 position and is shown in Figure 28. Shown in Figure 29 is a flux profile along the horizontal centerline of the core.

To determine the thermal nvt an average cadmium ratio of 2 was used for the 1-2 and 3-2 fuel positions. An average cadmium ratio of 2.8 was used for the 5-2, 3-4, 1-4, and 1-6 positions. The average thermal nvt was determined by summing the volume integrals of the vertical flux plots and dividing by the total volume of the fuel assemblies. Each volume integral was multiplied by the flux depression factor and by the appropriate cadmium ratio. The result of this calculation gave 1.4×10^{11} neutrons/cm² as the average thermal nvt. The average flux per watt, which is dependent on the macroscopic fission cross section and the

Fig. 25 Vertical Thermal Neutron Flux Distribution in the Center of Assembly Positions W1-4 and W5-2

Fig. 26 Vertical Thermal Neutron Flux Distribution in the Center of Assembly Positions W1-2 and W3-2

Fig. 27 Vertical Thermal Neutron Flux Distribution in the Center of Assembly Positions W3-4 and W1-6

Fig. 28 Neutron Flux Along the Horizontal Centerline in Assembly Position W3-4

Fig. 29 Neutron Flux Along the Horizontal Centerline of the Core

energy release per fission, was calculated to be 6.1×10^6 neutrons/cm²-sec-watt.

The ratio of the maximum measured flux in the core to the average over the core was 1.5. The location of the flux maximum was 14 inches from the top of the fuel at the north edge of assembly W1-2.

F. Temperature Coefficient of Reactivity

The reactivity effects associated with changes in the system temperature and the temperature coefficient of reactivity as a function of system temperature were determined over a range of initial system temperatures from 73°F. to 305°F.⁽⁷⁾

The position of the control rods for criticality was determined at approximately fifty degree increments in system temperature, from ambient (73°F) to a maximum system temperature of 305°F. The temperature of the entire core, moderator, and reflector regions was uniform and controlled by electrical immersion heaters in the reactor tank. During the course of the experiment, the system was maintained at a pressure of 260 psig to suppress boiling at the higher temperatures.

The temperature coefficient of reactivity, C_θ , and the reactivity defect due to increasing system temperature were determined from the

position of the control rods for criticality and the control rod worth data. The reactivity defect due to increasing system temperature, Figure 30, was determined to be approximately 7.35 dollars for increasing the system temperature from 73°F to 305°F. The temperature coefficient of reactivity, Figure 31, increases from -2.0 $\phi/^\circ\text{F}$ at 80°F to -4.4 $\phi/^\circ\text{F}$ at 300°F. Tables V and VI present these data in tabular form.

TABLE V

TEMPERATURE DEFECT OF REACTIVITY

<u>Temperature Range</u>	<u>Reactivity Defect</u> ($\$$)
73°F to 153°F	1.81
73°F to 205°F	3.45
73°F to 254°F	5.24
73°F to 305°F	7.35

TABLE VI

TEMPERATURE COEFFICIENT OF REACTIVITY

<u>Temperature</u> ($^\circ\text{F}$)	<u>Temperature Coefficient</u> ($\phi/^\circ\text{F}$)
113	-2.3
179	-3.2
229	-3.6
279	-4.3

The magnitude of the temperature coefficient of reactivity of the reactor resulted in a loss of all the excess reactivity available when the system temperature was increased to 325°F. Such a large defect severely limits the excess reactivity available for transient testing and essentially limits practical transient testing operation to initial system temperatures of 220°F or less.

The assumption is frequently made that the temperature coefficient of reactivity is almost entirely due to the density change of the water moderator. An investigation of the validity of this assumption for this core was made by considering the measured temperature defect to be a function of the

Fig. 30 Temperature Defect vs System Temperature

Fig. 31 Temperature Coefficient of Reactivity vs System Temperature

change in moderator density. This analysis showed that the reactivity loss is very nearly linear as a function of moderator density change and is closely approximated by a constant coefficient of $-.0.94$ reactivity change per percent uniform moderator-reflector density change, over the temperature range investigated.

G. Void Coefficient of Reactivity

The coefficients of reactivity associated with voids in the moderator-coolant channels between fuel plates in the core were determined in order to evaluate the reactivity effects of steam formation during transient tests.⁽⁸⁾

The measurements were made by use of a special fuel assembly that was sealed at the top and along the sides to form three equal volume air-tight compartments that were open at the bottom. Inlets were provided at the top of each compartment so that helium gas could be introduced to force water out of the bottom of the assembly to void it. The lower level of the void was regulated by a bubble tube that was connected in parallel with the assembly. A conceptual schematic of the apparatus is shown in Figure 32. Remotely operated valving was provided so that

Fig. 32 Conceptual Schematic of Void Apparatus

either the outer two compartments, the inner compartment, or the total assembly could be voided or flooded while the reactor was in operation. The remote operation feature allowed flooding of a voided compartment while the reactor was critical and determination of the reactivity effect of the void directly from a positive period measurement. This method allowed an accurate measurement of the relatively small reactivity effects associated with the voiding of each assembly compartment or compartments.

The void effect and coefficient of reactivity were measured for each fuel assembly position in the west quadrant. At each position, the void coefficient was measured for the outer compartments voided, the inner compartment voided, and for a totally voided assembly. The results of these measurements are presented in tabular form in Table VII and in a plan-form diagram of the core in Figure 33. No measurements were made in other core positions because of the core symmetry.

TABLE VII

REACTIVITY EFFECT AND COEFFICIENTS OF VOIDS

Assembly Position	$\Delta\rho$ For Void In:			$C_v = \Delta\rho/\Delta V$ For:		
	Center Compt. (ϕ)	Outer Compts. (ϕ)	Entire Assy. (ϕ)	Center Compt. ϕ/cm^3	Outer Compts. ϕ/cm^3	Entire Assy. ϕ/cm^3
W1-2	- 14	- 22	- 36	-0.030	-0.023	-0.025
W1-4	- 11	- 19	- 30	-0.024	-0.020	-0.021
W1-6	- 6	- 11	- 17	-0.013	-0.012	-0.012
W3-2	- 13	- 21	- 35	-0.028	-0.022	-0.025
W3-4	- 11	- 16	- 27	-0.022	-0.017	-0.019
W5-2	- 10	- 15	- 25	-0.022	-0.022	-0.017
Quadrant Total	- 65	-104	-170			
Quadrant Average				-0.023	-0.018	-0.020

The average void coefficient of reactivity, C_v , was determined by a simple averaging process of the coefficients for the assemblies in the west quadrant. This averaging yields a value for the average void coefficient of reactivity of -2.0×10^{-4} $\$/\text{cm}^3$ or -7ϕ per percent void. The data for the assemblies indicate that the sum of the reactivity effect of inner and outer voided compartments closely approximates that of a

Fig. 33 Void Coefficient of Reactivity as a Function of Core Position

totally voided assembly but the outer compartments do not give a value that is twice that of the inner compartment. It is reasonable to believe that the difference in the void effects of the inner and outer compartments is due to the flux distribution and self-shielding effects in the assemblies.

H. Statistical Measurement of β/β_{eff}

A value of the ratio of prompt neutron lifetime to effective delayed neutron fraction, or reduced prompt neutron lifetime, for the Spert II BD-22/24 expanded D_2O core was measured by observing the statistical behavior of the neutron population. This parameter, β/β_{eff} , is used to predict short period kinetic behavior of the system prior to execution of a dynamic testing program.

Several statistical techniques may be used to derive information from a multiplying medium. An analysis of the variance-to-mean ratio of counts as a function of counting time was used in this experiment. This approach is based on the measurement of the correlated fluctuations in neutron population which occur in a subcritical multiplying medium. In a counting experiment of this type, the times between the connected or correlated events, and hence the periods of the fluctuation, are determined by the prompt neutron lifetime and the delayed neutron half-lives.

The counting apparatus utilized was quite simple and for the most part consisted of components used for routine operation. Three B-10 lined pulse chambers located around the periphery of the core were used to detect the neutrons. The signals from the chambers were fed into pre-amplifiers and linear amplifiers and then to a mixer in the control center which added the signals. The output of the mixer drove a gated scaler whose gate time was precisely controlled by a 100 kc crystal oscillator. The output of the scaler was then recorded by a paper tape printer.

The measurement technique consisted of recording the number of counts detected during a fixed counting or gate time. This was repeated for about 500 gates at a constant gate time and the variance-to-mean ratio calculated for this set of data. The variance-to-mean ratio was obtained as a function of gate time by repeating the measurement for gate times ranging from 0.001 to 1.000 sec and for five subcritical reactivities varying from approximately 0.99 to 0.999.

The theory which relates the variance-to-mean statistic to fundamental system parameters was originally developed by de Hoffmann⁽⁹⁾. The relationship derived by de Hoffmann neglects the effects of delayed neutrons and thus correlates only those fluctuations in neutron population which are due to prompt neutrons. The variance-to-mean ratios calculated from the Spert II experimental data were fitted with this relationship which may be written as,

$$\begin{aligned} \frac{\text{Var}}{\text{Mean}} (\tau) &= 1 + \frac{E k_p}{(1-k_p)^2} \left[\frac{\overline{v^2} - \bar{v}}{\bar{v}} \right] \left[1 - \frac{1-e^{-\alpha\tau}}{\alpha\tau} \right], \\ &= 1 + K \left[1 - \frac{1-e^{-\alpha\tau}}{\alpha\tau} \right] \end{aligned} \quad (1)$$

where:

$$\frac{\text{Var}}{\text{Mean}} (\tau) = \text{variance-to-mean ratio} \equiv \frac{\overline{c^2} - \bar{c}^2}{\bar{c}},$$

τ = gate time,

c = counts per gate time,

K = a combination of fixed variables (constants),

E = detector efficiency,

k_p = prompt neutron reproduction factor,

$\bar{v}$ = average number of neutrons emitted per fission,

$\overline{v^2}$ = dispersion in the number of neutrons emitted per fission,

$$\alpha = \frac{1-k_p}{\ell}$$

ℓ = mean prompt neutron lifetime,

with the bar indicating a mean value.

In order to use the experimental variance-to-mean data to determine the parameter ℓ/β_{eff} , eq. (1) may be written as,

$$\frac{\text{Var}}{\text{Mean}} (\tau) - 1 = K \times (\alpha, \tau), \quad (2)$$

where the constant K is a combination of the fixed variables (constants) and X is a function of gate time, τ , and α . A non-linear least-squares fit to the data for K and α was then made using an IBM-650 program. Having obtained least-square values of α for each experimentally measured reactivity, a least-squares fit of $\alpha \equiv \frac{1-k_p}{\ell}$ vs $\frac{1-k_p}{\beta_{eff}}$ gives a straight line through the origin with slope ℓ/β_{eff} , shown in Figure 34. The value of ℓ/β_{eff} thus determined is 0.11 sec.

As indicated previously, the Spert II data were fitted assuming that delayed neutron effects could be neglected. However, recent work by Bennett(10), Tachon(11), and Albrecht(12) has shown that delayed neutron effects should be included for long prompt neutron lifetime systems. They show that in general the fitting equation should be written in the form,

$$\frac{\text{Var}}{\text{Mean}}(\tau) = 1 + \sum_{i=1}^{n+1} K_i \left[1 - \frac{1-e^{-\alpha_i \tau}}{\alpha_i \tau} \right], \quad (3)$$

where K_i is a function of the chamber efficiency, reactivity, and delayed as well as prompt neutron parameters; n is the number of delayed neutron groups; and α_i are the roots of the inhour equation. For $n = 0$, this equation reduces to the prompt approximation derived by de Hoffmann and is applicable to systems where the prompt root, $\alpha \equiv \frac{1-k_p}{\ell}$, of the in-hour equation is very much greater than the largest delayed root.

Thus, in a long-lifetime heavy-water system such as the Spert II expanded D_2O core, the neglect of delayed neutrons will introduce a systematic error into the least squares fitted value of α_1 , and tend to make the measured value of ℓ/β_{eff} larger than the true value. Because of this systematic error, no relative error was attached to the value of the reduced prompt neutron lifetime, ℓ/β_{eff} , determined by this method; and an independent determination will be made from the data obtained during the initial power excursion tests. This will allow an experimental evaluation of the error introduced by neglecting the delayed neutrons in analyzing the variance-to-mean ratio for the value of ℓ/β_{eff} of this long prompt neutron lifetime system.

Fig. 34 $(1-k_p)/\beta_{eff}$ vs $(1-k_p)/\ell$

VI. SUMMARY

The Spert BD-22/24 expanded D₂O core, composed of 3-inch square by 24-inch long, highly enriched, plate-type fuel elements spaced on six-inch centers, provides a configuration that is typical of several other heavy-water research reactor designs. Experiments with this core are intended primarily to examine the effects of a long prompt neutron lifetime on the kinetic behavior, and secondarily, to study the special features associated with heavy water systems. The static characteristics of this core provide information needed to plan, as well as to interpret, the results from kinetic experiments conducted over a wide range of environmental conditions.

A summary of the more important static characteristics is presented in tabular form in Table VIII. An operational loading of 3.7 kilograms of U-235 provides an available excess reactivity of approximately \$8.00 for the cold clean core. Because of the long prompt neutron lifetime, this amount of excess reactivity is required to perform short period kinetic tests under various system conditions. Large, negative, temperature and density coefficients were observed for uniform moderator heating, however, the smaller localized reactivity coefficients, i.e., within the fuel assemblies, are of more importance during the initial portion of power excursions. The local void coefficient was measured to be negative throughout the core, with an average value of $-0.02 \phi/\text{cm}^3$. The thermal neutron flux is relatively flat across the core and in the reflector with the measured flux depression across the fuel assemblies indicating that hot spots near assembly edges could occur during power excursions. The results of these static measurements on the BD-22/24 expanded D₂O core indicate that this system is satisfactory for the performance of a variety of kinetic experiments.

TABLE VIII

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

Core Designation	BD-22/24 Expanded D ₂ O Core
Fuel Assembly Type	Spert BD
Fuel Plates Per Fuel Assembly	22
Number of Fuel Assemblies	24
Core Configuration	Expanded (Figure 15)
Mass of U-235	
Critical Loading	2.8 kg
Operational Loading	3.7 kg
Average Thermal Flux Per Watt	$6.1 \times 10^{16} \frac{\text{Neutrons}}{\text{cm}^2\text{-sec-watt}}$
Ratio of Maximum Measured Core Flux to Average Core Flux	1.5
Temperature Coefficient of Reactivity (@ 260 psig)	
100°F	-2.2 ϕ /°F
200°F	-3.3 ϕ /°F
300°F	-4.4 ϕ /°F
Temperature Defect (@ 260 psig)	
73°F to 205°F	\$3.45
73°F to 305°F	\$7.35
Available Excess Reactivity	
73°F, Atmospheric Pressure	\$8.10
205°F, 260 psig	\$4.65
305°F, 260 psig	\$0.60
Coefficient of Reactivity Per Percent Density Change for Uniform Heating of the Moderator-Coolant-Reflector	-94 ϕ /%
Coefficient of Reactivity Per Percent Density Change of Moderator-Coolant Within the Fuel Assemblies	-7 ϕ /%
Average Void Coefficient of Reactivity	-0.02 ϕ /cm ³
Maximum Measured Local Void Coefficient of Reactivity	-0.03 ϕ /cm ³
Reduced Prompt Neutron Lifetime, ℓ/β_{eff}	0.11 sec

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