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SPERT PROJECT

October, November, December, 1964

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SPERT PROJECT
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SUMMARY

During the fourth quarter of this year, the Capsule Driver Core (CDC) was transferred from Spert I to Spert IV to initiate the experimental portion of a long-range subassembly testing program. With the transfer of the CDC to Spert IV, Spert I was deactivated as an operating reactor facility. During the nine-year operating history of Spert I, reactor safety studies have been conducted with nine cores (not including the CDC) with a wide variety of mechanical designs and dynamic characteristics. This report contains brief summaries of the various programs conducted with these cores in the Spert I facility.

With the completion of the experimental program with the close-packed, D₂O-moderated core, nuclear operations in the Spert II facility have been suspended, and the facility has been placed on a standby status.

The experimental program with the Spert III plate-type core has been completed with the performance of a series of transient tests initiated at high power levels. These tests included step- and ramp-type reactivity insertions and loss-of-flow tests initiated at power levels up to 28 MW under initial system conditions of 500°F, 2500 psig, and up to 20 ft/sec core coolant flow. Results of this series of tests were generally predictable from previous tests performed under less extreme plant conditions and from pretest calculations using the ART-04 computer code. No damage to the core resulted from any of the tests in this series.

The Spert III facility is being prepared for an experimental program with a low-enrichment oxide core to investigate the dynamic behavior of such cores under typical PWR conditions. The fuel assembly for this core contains 25, 0.466-inch-diameter, 43-inch-long fuel rods in a 5 x 5 array. During this quarter, flow measurements were made with this fuel assembly. The measurements showed that, at the design flow rate of 316 gpm (20,000 gpm core flow), the fuel assembly pressure loss was 44 psi at room temperature. This value is in good agreement with the design specification of 45 psi.

Preparations for nuclear operation of the Capsule Driver Core (CDC) in the Spert IV facility have been essentially completed. The initial program with the CDC in Spert IV will be performed to determine the transient response characteristics of the CDC fuel. In this program a CDC fuel rod (lead-rod) will be placed in the test space for a series of power excursion tests. Since the fuel rod in the test space will be subjected to energy densities about three times larger than the peak energy density in the core, the CDC fuel can be tested to destruction without endangering the core. In preparation for this testing, a program to determine a suitable capsule design has been initiated. This program includes both calculational and out-of-pile experimental investigations to determine the design requirements of such a capsule.

Also, for the CDC, a series of tests has been performed to determine if strain gauges mounted on the outside of capsules can be used to measure internal transient pressures. In these tests, exploding wires and chemical explosives were used to generate pressure pulses inside aluminum capsules. The resulting pressures were measured by hydrophones inside and outside the capsules and by foil, semiconductor, and weldable strain gauges mounted on the outside of the capsules. It was determined in these tests that the strain gauges can be used to give reasonably accurate internal pressure measurements;

the weldable gauges indicating pressures which agreed within 10 percent of those measured by the hydrophones.

As part of the PBF fuel testing program, two in-pile transient tests have been performed in the TREAT reactor facility to determine if thermal expansion of fuel pellets during transient testing could cause permanent growth of the fuel-rod clad. In these tests a PBF prototype fuel rod was subjected to peak transient energy densities of 260 cal/g, or about 125 percent of the PBF design peak energy density of 210 cal/g. Results from these tests indicated no detectable clad growth as a result of fuel pellet expansion.

The fuel for the PBF will consist of a ternary mixture of urania diluted with calcia-stabilized zirconia. An out-of-pile program has been conducted to determine the thermal constants and phase stability of such mixtures containing 60 and 70 volume percent urania. In the phase stability tests, the mixtures were held at temperatures of 1000 and 1300°C for 7 days and at 2000°C for 24 hours. On the basis of X-ray diffraction and microscopic analysis of samples from these tests, it was concluded that the phase stability of these mixtures is sufficient to permit their use in the PBF. Also, the solidus and liquidus temperatures and the thermal conductivities over the temperature range from 800 to 2200°C were determined for both mixtures.

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I. SPERT I

During the fourth quarter of this year, Spert I was deactivated as an operating reactor facility to allow construction of the Power Burst Facility (PBF) at a nearby location. In the nine-year (July 1955 to September 1964) operating history of Spert I, reactor safety studies, including step- and ramp-induced power excursion tests and stability tests, have been conducted with nine water-moderated cores with a wide variety of mechanical designs and dynamic characteristics. Also a tenth core, the Capsule Driver Core, was loaded into Spert I for a program of critical experiments and measurements of core static nuclear parameters. The following is a chronological summary of the tests which were performed with the various cores in Spert I.

A-17/28 Core (July 1955 -- June 1957). The first tests in Spert I were conducted with a highly enriched, uranium-aluminum, plate-type core, designated the A-17/28 [a] core. The purpose of the tests with this core was to determine the influence of system temperature, hydrostatic head, and moderator void effects on the kinetic behavior of the reactor, and to establish safe operating limits with regard to available excess reactivity, rate and magnitude of reactivity insertion, and minimum start-up power level for this type of core. There was a total of 885 critical determinations and 606 power excursion tests performed with this core in one year and ten months of operation. The tests included power excursions [1, 2] initiated by stepwise insertions of reactivity in the period range from approximately 10 sec to 5 msec, ramp-induced excursions [1] with ramp rates in the range from approximately 1.6 to 14 ϕ /sec and stability tests [3] in which the reactor power displayed large-amplitude, self-induced oscillations. These tests were performed under system conditions of temperature from ambient to boiling, initial power level from approximately 10^{-4} to 2×10^5 W, and with hydrostatic heads of from 2 to 9 feet above the core.

The "B" Cores (July 1957 -- June 1958). Following the tests with the A-17/28 core, a series of tests was performed with three highly enriched, uranium-aluminum, plate-type cores [4], designated the "B"-cores; viz, the B-24/32 core, the B-16/40 core, and the B-12/64 core. The purpose of the tests with these cores was to investigate the influence of the void coefficient on reactor power excursion behavior. The type-B fuel assembly when fully loaded contained 24 fuel plates; however, 20 of these plates were removable, thus allowing appreciable changes to be made in the moderator channel thickness, the void coefficient, and the number of fuel assemblies required for criticality. The average void coefficients for the three cores were $-0.93 \times 10^{-2} \phi/\text{cm}^3$ for the B-12/64 core, $-2.9 \times 10^{-2} \phi/\text{cm}^3$ for the B-16/40 core, and $-7.3 \times 10^{-2} \phi/\text{cm}^3$ for the B-24/32 core. Although the average void coefficients for all three cores were negative, the B-12/64 core did have a slightly positive coefficient in the central region of the core. With these cores, step-induced power excursion

[a] In the Spert nomenclature for plate-type cores, the letter (A) stands for the type of fuel assembly, the first number (17) indicates the number of fuel plates in a fuel assembly, and the second number indicates the number of assemblies (28) in the core.

tests [5] were performed in the period range from 10 sec to 10 msec with initial moderator temperatures from ambient to boiling.

P-18/19 Core (August 1958 -- August 1959). The first stainless steel clad core tested in Spert I was the P-18/19 core [6]; a core composed of Army Package Power Reactor (APPR) type fuel assemblies. The purpose of the tests with this core was to determine if the difference in heat transfer characteristics between aluminum and stainless steel was reflected in different reactor power excursion behavior. A total of 23 power excursion tests [7] with asymptotic periods in the range from 6 sec to 4 msec was performed with this core over an initial temperature range from ambient to boiling. It was determined from these tests that there is no significant difference in kinetic behavior between aluminum- and stainless-steel-clad, highly enriched cores. Also with the P-18/19 core, a program of pile-oscillator tests was performed which led to a significant contribution [8] to the reactor describing function concept, and a program of capsule experiments [9] was performed to aid in the understanding of moderator boiling during transient tests (for the boiling studies, 7 fuel assemblies were added to form the P-18/26 core).

BSR-II Core (October 1959 -- April 1960). A small, stainless-steel-clad, plate-type core, the Bulk Shielding Reactor-II (BSR-II), designed for use in the Bulk Shielding Facility at ORNL, was installed in Spert I to investigate the kinetic behavior of the core and to supply detailed performance characteristics of a fast-response safety system during short-period transients. Two main types of power excursion tests [10, 11] were performed with this core; those without the safety system in operation (self-shutdown tests) and those with the safety system in operation (mechanical-shutdown tests). These tests showed that the control system was more effective than the intrinsic reactivity compensating mechanisms in limiting reactor power excursions with periods longer than about 3 msec, the shortest period test performed with this core. For shorter period excursions it was determined that the intrinsic mechanisms would provide more shutdown reactivity than the safety system at the time of peak power; and in the period region where core damage could occur, no increased protection to the reactor would be provided by the safety system.

SA-592 Core (February 1961 -- October 1961). Following the tests with the BSR-II core, a low-enrichment oxide core, the SA-592 core, was loaded in the Spert I facility. The fuel rods used in this core were obtained from the Babcock and Wilcox N. S. Savannah critical assembly. The rods were six-foot-long, welded-seam, stainless steel tubes, each containing 1600 g of four percent enriched UO_2 powder compressed to 87 percent of the theoretical density. Self-limiting power excursion tests [12, 13] were performed with this core over a range of asymptotic period from 30 sec to 3.2 msec. In the short-period region with this core, it was determined that the power bursts were limited, almost entirely, by the Doppler broadening of U-238 neutron absorption resonances, thereby permitting a direct measurement of the dynamic Doppler coefficient of the core. An additional positive reactivity effect which significantly affected the power burst shape was determined to occur as a result of systematic bowing of fuel rods during power bursts. Separation of this fuel-rod-bowing effect was accomplished by the performance of two series of transient tests; one in which the rods were free to bow and one in which the bowing was mechanically restricted.

D-12/25 Core (March 1962 -- December 1962). In recognition of the need for an understanding of destructive mechanisms in highly enriched, uranium-aluminum, plate-type cores, the D-12/25 core was loaded in Spert I for a series of tests [14, 15] ultimately leading to the destruction of the core. With this core, step-induced tests initiated at ambient temperature and atmospheric pressure were performed with asymptotic periods in the range from 1.3 sec to 3.2 msec. In the 3.2-msec-period test, approximately 35 percent of the core was melted; and approximately 15 msec after the power peak, a short-rise-time pressure pulse occurred which completely demolished the core, damaged most of the associated hardware, and bulged the earth-backed reactor vessel. It was postulated that the damaging pressure pulse resulted from a steam explosion caused by the reactor water coming in contact with the molten fuel.

OC-Core (May 1963 -- May 1964). The OC-core was installed in Spert I for a program to obtain information on the cause, magnitude, and consequences of fuel-rod failure when a low-enrichment oxide core is subjected to rapid, large reactivity additions. This core was essentially identical to the SA-592 core except that a few more fuel rods were required in the OC-core for the desired excess reactivity. With this core, a series of power excursion tests, culminating with two short-period (2.2 and 1.55 msec) potentially destructive tests [16], was performed under conditions of ambient temperature and atmospheric pressure. In both of the potentially destructive tests, two fuel rods were ruptured, causing hot, finely divided fuel to be dispersed into the water, and approximately 25 percent of the fuel rods were mechanically deformed or discolored. No appreciable damage occurred to components other than the fuel rods, however; and only minor fission product releases from the ruptured fuel rods could be detected. The fuel rod failures were postulated to have resulted from high internal pressures caused by the rapid heating of water which had entered the fuel rods through defects in the cladding or end-plug closures.

Capsule Driver Core (July 1964 -- August 1964). The final core to be tested in Spert I was the Capsule Driver Core. This core consisted of 1620, six-foot-long, stainless steel fuel rods [17], each rod containing approximately 1600 g of three percent enriched UO_2 powder. A series of critical experiments and static nuclear parameter measurements [18] was made with this core in Spert I; following which the core was transferred to Spert IV for a subassembly testing program, thus ending nuclear operations in the Spert I facility.

II. SPERT III

1. SPERT III HIGH POWER TRANSIENT TEST SERIES

The experimental program with the Spert III C-19/52 plate-type core has been completed with the performance of the high power transient test series. This test series extended previous investigations of the dynamic response of the C-19/52 core to include nuclear excursions initiated from high power levels. Reactor power excursions were initiated by both stepwise and ramp reactivity insertions from initial reactor power levels up to 28 MW. In addition, partial and total loss-of-coolant flow tests were conducted from initial reactor power levels of 10 and 27 MW. All tests were conducted with initial system conditions of 500°F reactor inlet coolant temperature and 2500-psig coolant pressure; the core coolant flow rate was 20 ft/sec in all but four cases where lower flow rates (17, 10, and 5 ft/sec) were used. The duration of each transient test was limited to a period of time less than that of the primary coolant loop transport time (≈ 13 sec for a coolant flow rate of 20 ft/sec) to avoid transient temperature feedback perturbations. Table I contains a summary of all tests performed and the respective initial system conditions and reactivity insertions.

The data and conclusions presented in the following discussion are necessarily preliminary and are subject to minor changes as the analysis of the test data continues.

1.1 Transient Tests Initiated By Stepwise Reactivity Insertions

Power excursions initiated by stepwise reactivity insertions were carried out to investigate the effects of initial power level and coolant flow rate upon the kinetic behavior of the C-19/52 core.

The use of the term stepwise reactivity insertion requires qualification in the present context. For transient tests initiated from low power levels, the assumption of a step reactivity insertion can be justified since no reactivity compensation occurs prior to the completion of the insertion. During such transients, the reactor power achieves an asymptotic period which can be used to characterize the power behavior. For tests initiated at high power levels, however, the finite time required to complete the insertion (≈ 70 msec for a 1-dollar insertion) becomes a factor. For instance, for the 1-dollar transient test initiated at 28 MW (see Table I, Run Number 113) the "step" reactivity insertion was completed almost simultaneously with the occurrence of the initial power peak, and the reactor power never reached an asymptotic period. The term stepwise reactivity insertion, therefore, is used to differentiate between tests in which the reactivity is added at the fastest available rate (≈ 15 \$/sec) and tests in which the reactivity is added at slower rates.

For transients conducted from high power, the variables of the initial peak power (or the initial point of deflection) and run-out power as a function of starting power are useful in characterizing the reactor dynamic behavior. The run-out power used for comparison of the transient tests described here is defined as the power measured 1.6 sec after the start of the reactivity insertion. This particular time was selected since at this time any transient power oscillations were essentially damped out.

TABLE I
HIGH POWER TRANSIENT TEST SERIES
(Spert III C-19/52 Core)

<u>Initial Transient Test Conditions</u>					
<u>Transient Run Number</u>	<u>Reactor Power (MW)</u>	<u>Coolant Temperature (°F)</u>	<u>Coolant Pressure (psig)</u>	<u>Coolant Flow Rate (ft/sec)</u>	<u>Reactivity Insertion (\$)</u>
<u>Step Reactivity Insertions</u>					
92	0.01	501	2500	20	0.69
93	0.01	500	2500	20	1.01
94	0.1	500	2500	20	0.79
95	0.09	504	2500	20	0.98
112	0.89	503	2500	20	0.72
96	0.96	502	2500	20	0.99
100	0.98	506	2500	20	0.95
106	0.98	503	2500	20	0.99
110	0.97	504	2500	20	0.97
104	9.5	504	2500	20	0.97
113	28.0	502	2500	20	0.96
103	0.65	504	2500	17	1.01
105	0.97	505	2500	17	0.97
109	1.03	504	2500	10	0.99
114	0.65	504	2500	5	0.97
<u>Ramp Reactivity Insertions</u>					
99	0.91	503	2500	20	0.27\$ @5.3¢/sec
102	0.85	504	2500	20	0.27\$ @5.3¢/sec
101	24.5	504	2500	20	0.28\$ @5.3¢/sec
<u>Loss-of-Flow Transients</u>					
107	9.9	504	2500	20	None
108	26.7	503	2500	20	None
111	10.0	504	2500	20	None
117	27.5	507	2500	20	None

1.11 Effects of Initial Power Level. Reactivities of 0.7 and 1.0 dollars were inserted into the core in a stepwise manner over the range of initial power levels from 0.01 to 28 MW (see Table I) to investigate the effects of initial power level upon reactor response. The 0.7-dollar insertions provided initial precautionary scoping for the more severe 1-dollar transients and also tested the validity of the results of the pretest analysis.

The initial peak powers and run-out powers for the 1-dollar transient tests are shown in Figure 1 plotted as functions of their respective initial power levels. The transient tests initiated at the lower power levels of 0.01 and 0.1 MW resulted in power behavior that was essentially the same as was obtained from

Fig. 1 Initial peak power and run-out power versus initial power level for one-dollar stepwise reactivity insertions with Spert III C-19/52 core.

similar tests performed during an earlier series [19] in which the initial power levels were approximately 10 W. This indicates that over the range of initial power level from 10 W to 0.1 MW the initial peak and run-out power behavior is essentially independent of the initial power level from which a step insertion is made. For the step insertions made at initial powers above 0.1 MW, the variables of initial peak and run-out power increased rapidly in an approximately linear manner with increasing initial power. The initial peak power increased from approximately 7 to 170 MW over the range investigated; run-out powers correspondingly increased from 21 to 100 MW.

Temperatures for these transient tests were measured using thermocouples consisting of MgO-insulated, Chromel-Alumel wires encased in stainless steel sheaths which were flattened to about 7 mils at the junction and tack-welded to the fuel plate surfaces. The maximum measured fuel plate surface temperature rise, taken 1.6 sec after the start of the reactivity insertion, appears in Figure 2, plotted as a function of initial reactor power level. The trend of behavior of this temperature rise follows that described for power behavior in the preceding paragraph. Over the range of initial power used, the temperature rise increased from 64 to 119°F.

1.12 Effects of Coolant Flow Rate. The effects of coolant flow rate upon reactor response to 1-dollar step reactivity insertions from 1-MW power levels were investigated. Step reactivity insertions were made at coolant flow rates of 20, 17, 10, and 5 ft/sec. The variables of initial peak power, run-out power, and maximum measured fuel plate surface temperature rise for corresponding coolant flow rates are presented in Figure 3.

Fig. 2 Maximum fuel plate surface temperature rise after 1.6 sec versus initial power level for one-dollar stepwise reactivity insertions with Spert III C-19/52 core.

Fig. 3 Spert III C-19/52 core initial peak power, run-out power, and maximum fuel plate surface temperature rise versus core coolant flow rate for one-dollar stepwise reactivity insertions initiated at 1 MW, 500°F, and 2500 psig.

Increasing the coolant flow rate from 5 ft/sec toward 20 ft/sec resulted in approximately linear increases in the run-out power and the maximum measured fuel plate surface temperature rise. Over the span of flow rates investigated, the run-out power increased from 13 to 44 MW and the temperature rise increased from 60 to 78°F. The magnitude of the initial peak power remained essentially constant (≈ 24 MW) over the range of flow rates investigated. These results are consistent with the behavior observed during similar transient tests performed in an earlier test series where initial power levels were approximately 10 W [19].

1.2 Operational Transient Tests

The tests in this part of the series were performed to investigate the reactor response to two types of accidents that have been postulated for pressurized-water reactors operating at significant power levels: (1) control rod withdrawals, ie, ramp-type reactivity insertions; and (2) partial and total losses of coolant flow.

1.21 Ramp Reactivity Insertions (Control Rod Withdrawals). Reactor response to control rod withdrawals from appreciable power levels were investigated by means of three transient tests; two of these tests being initiated at power levels of approximately 1 MW and one at 24 MW.

The tests were started from steady-state plant conditions by inserting reactivity at a rate of approximately 0.05 dollar/sec by continuous control rod withdrawal until approximately 0.27 dollar had been inserted. Control rod motion was then stopped, and the power level was allowed to rise toward an equilibrium value. The reactor was scrammed approximately 10 seconds after the start of the reactivity insertion.

The power and temperature behavior of the ramp transient test started from an initial power level of 24 MW (Table I, Transient Run Number 101) is presented in Figure 4, plotted as a function of time after the start of the reactivity insertion. The reactor response during this test was as expected from the results of pretest calculations. The reactor power rose fairly rapidly from approximately 24 to 36 MW during the period of control rod withdrawal, then continued to increase at a slower rate until the reactor was scrammed. The maximum measured fuel plate surface temperature closely followed the power behavior throughout the test. In the tests initiated from 1 MW the reactor exhibited a similar behavior with correspondingly lower values of power and temperature.

1.22 Loss-Of-Flow Tests. The reactor response to losses-of-coolant flow while at appreciable power levels and in the absence of automatic low-flow-scram protective devices, was investigated in this part of the test series. Partial and total loss-of-coolant flow tests were performed from initial power levels of 10 and 27 MW. The total loss-of-flow tests were carried out by establishing stable reactor plant conditions at the selected power level, then shutting off the electrical power to all four main coolant pumps; the reactor was scrammed five to six seconds later. Partial loss-of-flow tests were performed by simultaneously switching off the electrical power to two coolant pumps, leaving one pump operating in each loop.

The power and fuel plate surface temperature behavior resulting from the most extreme loss-of-flow test performed, the total loss of flow from an initial power of 27 MW (see Table I, Transient Run Number 117), is presented in

Fig. 4 Spert III C-19/52 core power and maximum fuel plate surface temperature rise versus time after start of ramp; initial system conditions, 24 MW, 500°F, 2500 psig, and 20 ft/sec flow.

Figure 5 as a function of time after the coolant pumps were shut off. In this test, the reactor power level decreased rapidly for the first two seconds to approximately 66 percent of the initial power level, then decreased at a slightly slower rate for three seconds more until the control rods were scrammed. The maximum measured fuel plate surface temperature rose from 550 to 636°F over the total five-second time interval. The other loss-of-flow tests performed exhibited a similar type of behavior.

Extrapolations of power and temperature data from the above total loss-of-flow test indicate that the reactor could sustain a total loss-of-flow condition for approximately six to eight seconds before reaching the coolant saturation temperature ($\approx 668^\circ\text{F}$) and the onset of boiling. The pretest calculations indicate that even after the onset of boiling, the decrease in power due to the change in the core moderating characteristics

Fig. 5 Spert III C-19/52 core fractional reactor power and maximum fuel plate surface temperature for total loss-of-flow test; initial system conditions, 27 MW, 500°F, 2500 psig, and 20 ft/sec flow.

and the high heat transfer rates accompanying nucleate boiling would prevent reaching a departure from nucleate boiling condition (DNB) from which core damage might result.

1.3 Calculations of Transient Behavior

A major objective of the high power transient test series was to provide data for use in evaluating the effectiveness of existing analytical models in calculating transient behavior for plate-type reactor cores similar to that in the Spert III reactor plant. Prior to the performance of the test series, calculations of expected behavior for the proposed tests were performed using the ART-04 computer code [20]. These calculations were used as general guides in the performance of the test series.

A comparison of the experimental and calculated reactor response to a 1-dollar step reactivity insertion from an initial power of 1 MW and a coolant flow rate of 17 ft/sec (see Table I, Transient Run Number 105) is made in Figures 6 and 7. Power behavior appears in Figure 6 and temperature behavior in Figure 7, each plotted as a function of time after the beginning of the step reactivity insertion. The comparison shows that the experimental initial peak power was approximately 77 percent of that calculated, the experimental run-out power, taken 1.6 seconds after the initiation of the reactivity insertion, was 110 percent of the calculated value; and the rise in the maximum measured fuel plate surface temperature, after 1.6 seconds, was 130 percent of that calculated.

Fig. 6 Experimental and calculated Spert III C-19/52 core power response to a one-dollar stepwise reactivity insertion; initial system conditions, 1 MW, 500°F, 2500 psig, and 17 ft/sec flow.

Fig. 7 Experimental and calculated Spert III C-19/52 core fuel plate surface temperature response to a one-dollar stepwise reactivity insertion; initial system conditions, 1 MW, 500°F, 2500 psig, and 17 ft/sec flow.

In some cases, the initial conditions of the pretest calculations did not duplicate those used in the experiments. Therefore, an ART-04 calculation is being performed for each experiment using a duplication of the experimental initial conditions.

1.4 Conclusions

Reactor response to the transient tests initiated from conditions of high power, coolant temperature, coolant flow rate, and pressure disclosed no significant departure from the general behavior expected on the basis of pretest calculations and extrapolations of experiments performed during earlier test series conducted at less extreme plant conditions. The analytical model used for pretest calculations of transient behavior, the ART-04 computer code, was demonstrated to be effective in calculating the experimental behavior observed during the tests. This success indicates that this model can be most useful in calculating transient behavior for plate-type, pressurized water reactors similar to Spert III for cases in which experimental data are not available.

2. HYDRAULIC CHARACTERISTICS OF SPERT III TYPE-E 25-ROD FUEL ASSEMBLY

With the completion of the experimental program with the C-19/52 core, an experimental program with a slightly enriched UO_2 core, designated the Spert III E-core will be initiated. The primary objective of this program is to investigate the dynamic behavior of such cores under environmental conditions typical of operating power reactors. In preparation for this program, the hydraulic characteristics of the fuel assembly to be used in the E-core have been determined.

The type-E fuel assembly, shown in Figure 8, is composed of 25 fuel rods containing 4.8 percent enriched UO_2 fuel pellets encased in stainless steel tubes. These rods are positioned within the fuel assembly by a lower grid, a removable upper grid, and two intermediate grids. The removable upper grid permits removal, inspection, and replacement of the inner nine fuel rods. The remaining 16 fuel rods are not ordinarily removable from the assembly. The pertinent fuel rod and assembly information is given in Table II.

With a dummy type-E fuel assembly, flow tests were conducted in a low-pressure, low-temperature hydraulic test loop. The fuel assembly pressure loss was determined by monitoring the static pressure below and above the fuel assembly with a mercury manometer and a differential pressure gauge. The flow was determined by use of a sharp-edged orifice and monitored with an oil manometer. The experimental overall pressure-loss, assembly-flow relationship is shown in Figure 9. At the design flow rate of 316 gpm (20,000-gpm core flow), the assembly pressure loss was 44 psi at room temperature. This value is in good agreement with the design specification of 45 psi.

Fig. 8 Isometric and cross section view of Spert III type-E, 25-rod fuel assembly.

TABLE II

SPERT III TYPE-E,
25-ROD FUEL ASSEMBLY

Fuel	
Type	Sintered UO ₂
Enrichment (wt%)	4.8
Pellet length, short (in.)	0.511
Pellet length, long (in.)	0.766
Pellet diameter (in.)	0.420
Density (g/cm ³)	10.50
Rod	
Type	348 stainless steel
Outside diameter (in.)	0.466
Wall thickness (in.)	0.020
Fuel length (in.)	38.3
Overall length (in.)	43.4
Filler gas	He
Weight UO ₂ /rod (g)	910
Weight U-235/rod (g)	38.5
Fuel-clad gap (nominal) (in.)	0.003
Assembly	
Weight U-235/assembly (g)	962.5
Rod pitch (in.)	0.585
Overall length (in.)	52.750
Overall width (in.)	2.975
Overall thickness (in.)	2.975
Flow area (inside)/assembly (in. ²)	4.29
Flow area (3 x 3 cell)/assembly (in. ²)	4.44
Heat transfer area/assembly (in. ²)	1402
Metal water ratio (3 x 3 cell)	1.03

Fig. 9 Pressure loss versus fuel assembly flow for Spert III type-E, 25-rod fuel assembly.

III. SPERT IV

1. INTRODUCTION

Upon completion of an initial series of static tests performed with the Capsule Driver Core (CDC) in the Spert I facility last quarter [18] the CDC was dismantled for transfer to the Spert IV facility. During this quarter, the CDC support structure and associated hardware were reassembled in the north pool tank of the Spert IV facility; and preparations for nuclear operation were completed.

Nuclear operation of the CDC will be resumed in the second quarter of 1965, and a fiducial series of transient tests will be initiated. The objectives of this transient test series are to experimentally determine the kinetic behavior of the core and to establish safe operating limits for a program of subassembly testing with the CDC. In order to establish the operating limits without damaging the core, a "lead-rod" will be used; ie, since a CDC-type fuel rod placed in the test space of the core can be subjected to an energy density about three times larger than the peak energy density in the core, it is possible to investigate the response of a CDC lead-rod located in the test space without endangering the core. The lead-rod will be encapsulated for transient testing in order to ensure containment of pressure or mechanical energy generation by the lead-rod.

During this quarter, out-of-pile tests were initiated to determine a capsule design for the lead-rod transient testing.

2. CDC CAPSULE DESIGN

The CDC has been designed to subject fuel samples in the test space to short-period, high temperature excursions where failure of the fuel samples is expected. The purpose of these tests is to experimentally determine the thresholds of fuel failures and the consequences of such failures during accident conditions. The physical consequences of fuel failures are not well defined; and it is, therefore, reasonable to expect that significant transient pressure pulses may arise around the fuel rod as a consequence of several possible processes such as (a) fuel rod bursting caused by large internal pressures; (b) rapid dispersal of finely divided, hot, UO_2 fuel into the water; and (c) hot or molten clad-water reactions. Since the lead-rod tests to be performed during fiducial transient testing of the CDC will comprise the first series of capsule tests in the CDC and since little prior experience in such testing is available, stringent design requirements and an intensive out-of-pile test program have been established for the capsule to be used for the initial test series.

Among the requirements in the design of the capsule for the first series of tests are (a) physical size compatible with the test space; (b) containment of pressure or mechanical energy generation within the capsule; (c) high figure-of-merit (ratio of energy density in test fuel to maximum energy density in core); and (d) provision for instrumentation that will supply interpretable data to determine pressure and energy releases during fuel failure.

The maximum possible outside diameter of the lead-rod capsule is 3.75 inches (the inside diameter of the experiment tube which encloses the CDC test space). However, if a capsule is permanently deformed by an internal pressure pulse, it may become lodged in the experiment tube. To reduce this possibility, a maximum outside diameter of 3.25 inches has been set for the lead-rod capsule. Following potentially destructive tests performed with the Spert I Oxide Core [16], inspection of ruptured fuel rods revealed that the 0.5-inch outside diameter rods could be deformed by failure until they were up to 1-3/16 inches wide. In order to ensure that there will be no interference with the physical response of the CDC lead-rod by the capsule, a minimum inside diameter of 1.5 inches has been set for the capsule. The maximum design length for CDC capsules is about six feet.

As a design objective, it has been established that the capsule must be able to withstand up to ten percent of the available lead-rod thermal energy for a 2-msec-period test (the shortest expected reactor period possible with the CDC) in the form of mechanical energy produced by a high explosive. The mechanical energy associated with the pressure pulse and afterflow caused by the failure of two fuel rods during the 1.55-msec-period test performed with the Spert I Oxide Core was determined to be between two and three percent of the available thermal energy contained in the dispersed fuel. However, the failure of fuel rods observed during the Spert I tests is suspected to have been caused by "water-logging". Waterlogged fuel rods may rupture at smaller energy densities than fuel rods which are not waterlogged, thereby releasing less thermal energy for conversion to mechanical energy. In the Spert I tests, fuel rod ruptures occurred at about the time of peak power when the internal fuel rod temperatures were about 600°C. Many fuel rods which did not burst during these tests reached temperatures in the range of 1000 to 1400°C. If the assumption is made that the CDC lead-rod will not fail until the power burst is essentially complete, a maximum internal temperature of 2800°C or greater can be expected for a 2-msec-period test.

The lower limit of static internal pressure which the lead-rod capsule must withstand without failure has been set at about 15,000 psi.

The figure-of-merit for a given test fuel is a function of the material from which the capsule enclosing the test fuel is constructed and also of the dimensions of the capsule. Previous computer results had indicated that the figure-of-merit decreases with increasing capsule wall thickness and also that capsules constructed of aluminum allowed higher figures-of-merit than stainless steel capsules when designed for the same static bursting pressure. However, since stainless steel is more ductile than aluminum, more energy is required to burst a stainless steel capsule than an aluminum capsule. Computer studies were, therefore, undertaken to determine the optimum wall thickness for both aluminum and stainless steel capsules. Capsule inside diameters of 1.5 and 2.0 inches were used. Figure 10 is a plot of figure-of-merit versus wall thickness obtained from the calculations. As can be observed from the figure, 2-inch-ID capsules tend to yield larger figures-of-merit than 1.5-inch-ID capsules as the wall thickness is increased.

The energy of strain required to burst a capsule was computed using the formula [21]:

$$E = \epsilon_u \left(\frac{\sigma_y + \sigma_u}{2} \right) \left[\pi (r_o^2 - r_i^2) \right]$$

Fig. 10 Figure-of-merit for a CDC fuel rod as a function of wall thickness for aluminum and stainless steel capsules.

lead-rod during a 2-msec-period test. With the assumption that not more than five percent of the thermal energy released in the lead-rod will appear as strain energy in the capsule wall (the conversion of mechanical energy released in the capsule to strain energy in the capsule wall is not known) the optimum wall thickness for a 2.0-inch-ID aluminum capsule, indicated by the calculations, is 0.53 inch. The static internal bursting pressure for an aluminum capsule of these dimensions is 18,000 psi.

Figure 12 shows a plot similar to Figure 11 for a 2.0-inch-ID stainless steel capsule. Assuming that not more than ten percent of the expected thermal energy released in the lead-rod during a 2-msec-period test will appear as strain energy in the capsule wall, the optimum wall thickness for a 2.0-inch-ID stainless steel capsule indicated by the calculations is 0.17 inch. The static internal bursting pressure for a stainless steel capsule of these dimensions is 12,000 psi.

The calculations indicate that the aluminum and stainless steel capsules of optimum wall thickness each allows a figure-of-merit of 2.3 but that the stainless steel capsule can withstand almost twice as much strain energy as can the aluminum capsule. On the basis of energy containment (considered to be a more reliable criterion than static bursting

where:

$$E = \text{energy} \left(\frac{\text{in.} \cdot \text{lb}}{\text{in.}} \right)$$

$$\sigma_y = \text{yield stress (psi)}$$

$$\sigma_u = \text{ultimate stress (psi)}$$

$$\epsilon_u = \text{ultimate elongation (in./in.)}$$

$$r_o = \text{outside capsule radius (in.)}$$

$$r_i = \text{inside capsule radius (in.)}$$

As the wall thickness of a capsule is increased, both the energy required to burst the capsule and the static internal pressure which the capsule can withstand increase. Figure 11 shows a plot of wall thickness versus the energy required to burst a 2.0-inch-ID aluminum capsule and five percent of the energy expected to be released in the

Fig. 11 Optimization curves for a 2.0-inch-ID aluminum capsule.

Fig. 12 Optimization curves for a 2.0-inch-ID stainless steel capsule.

pressure), these optimization studies indicate that the stainless steel capsule is the better choice. However, the amount of mechanical energy released in a capsule which appears as strain energy in the walls is probably a function of capsule material. Therefore, out-of-pile tests will be performed to determine energy conversion efficiency as a function of capsule wall material.

An extensive out-of-pile testing program has been initiated to (a) determine experimental methods by which the mechanical energy release within a capsule can be measured; (b) determine the conversion factor from mechanical energy release within a capsule to strain energy in capsule walls for different materials and geometries; and (c) prove that the final-design lead-rod capsule will withstand a mechanical energy release equivalent to ten percent of the expected thermal energy available in the lead-rod for a 2-msec-period test.

3. CAPSULE DYNAMIC PRESSURE MEASUREMENTS

From previous CDC neutron flux distribution measurements [18], the hot spot in the lead-rod is expected to be about 16 inches above the bottom of the capsule; and since pressure disturbances may originate near the lead-rod hot spot, one or more pressure measuring devices are desired in this region. Unfortunately, the geometry of the test space and capsule prevents the use of conventional pressure sensors in this region. Since the static internal pressure within a cylinder can be calculated from the hoop strain in the cylinder wall and since no space problem is involved in attaching strain gauges to the capsule wall, experiments were performed to determine the accuracy to which dynamic internal pressures could be calculated from the response of strain gauges attached to the outer surface of the capsule wall. Experiments were also performed to determine what fraction of internal pressure can be transmitted through the capsule walls.

3.1 Methods for Producing Dynamic Pressure

Experiments were performed using two sources of dynamic pressure: (a) electrically exploded wires and (b) Primacord [a], a chemical explosive. All

[a] Trademark for the Ensign-Bickford Company detonating fuse consisting of a central explosive core containing pentaerythritol tetranitrate (PETN) and the cloth or plastic encasement of this core. The detonation products of PETN are 100 percent gaseous.

experiments were performed in a water environment. The explosion of a five-inch-long copper wire was produced by discharging a capacitor into the wire. Magnitudes of the resulting dynamic pressures were varied from 400 to 3000 psi by changing the voltage on the capacitor and also by changing the wire diameter. These pressure pulses had rise times of about $3\ \mu\text{sec}$ and durations of about $65\ \mu\text{sec}$ which did not show measurable changes over the range of pressures produced. These values may be compared with a rise time of 20 to $30\ \mu\text{sec}$ and a duration of 1 msec or less expected to result from fuel rod failure. This expectation is based on results from the Spert I Oxide Core destructive tests [16]. Durations were measured from the time the pressure started to rise until it returned to zero.

Primacord, loaded to 25 grains of PETN per lineal foot, was used to produce dynamic pressures which more closely approximated those measured during the Spert I Oxide Core destructive tests. Three-foot lengths of Primacord were exploded using a No. 6 electric blasting cap located at their centers. The Primacord was placed in a stainless steel tube in the center of the capsule, and the magnitude of the pressure resulting from the explosion of Primacord was varied from 500 to 3000 psi by changing the thickness of the stainless steel tube and by placing either water or air in the tube. Resulting pressure pulses had rise times of about $10\ \mu\text{sec}$ and durations of about $120\ \mu\text{sec}$. Again, the rise time and duration of the pressure pulses produced did not show measurable changes over the range of pressures generated.

3.2 Results of Exploding-Wire Tests

The capsule used for exploding-wire tests was 5 inches long with a 1.8-inch ID and a wall thickness of 0.1 inch. The copper wire was centered along the vertical axis of the capsule. Crystal hydrophones were used to measure pressure inside the capsule (approximately midway between the vertical axis and the capsule wall) and at various positions outside the capsule. Three types of strain gauges were mounted on the outside of the capsule near its center: weldable, semiconductor, and foil gauges. The semiconductor gauges were used, because of their known dependability, for comparison with the other types; however, they can not be used as in-core instrumentation because of possible radiation damage to the semiconductor material. All except the weldable gauges were attached to the capsule with Eastman 910 cement. Dynamic pressure inside the capsule was calculated using the strain indicated by these gauges and compared with results obtained from the hydrophones.

Internal capsule pressures calculated from the response of all three types of strain gauges used during the exploding-wire tests agreed very closely with the measured pressure inside the five-inch-long capsule. However, the foil gauges were found to fail after a few tests at about 1400-psi internal capsule pressure. The dependability of foil gauges might be increased by the use of a different type of cement in their attachment to a capsule.

A typical pressure pulse produced inside the capsule by the exploding wire, as measured by a crystal hydrophone, is shown in Figure 13. This pulse had a peak pressure of 1460 psi a rise time of $3\ \mu\text{sec}$, and a duration of $67\ \mu\text{sec}$. The strain responses of the foil, semiconductor, and weldable strain gauges to the internal capsule pressure shown in Figure 13 are shown in Figures 14, 15, and 16, respectively. As can be observed from these figures, the strain gauge responses have the same general shape as the pressure pulse. A summary of

764-A-0271

Fig. 13 Measured pressure produced by exploding wire inside 5.0-inch-long capsule.

764-A-0272

Fig. 14 Response of foil strain gauge to the pressure pulse shown in Figure 13.

764-A-0273

Fig. 15 Response of semiconductor strain gauge to the pressure pulse shown in Figure 13.

764-A-0274

Fig. 16 Response of weldable strain gauge to the pressure pulse shown in Figure 13.

the results obtained from the three strain gauges for this particular test is given in Table III. Both the weldable and semiconductor strain gauges were found to have a linear response to dynamic pressures in the range of 400 to 3000 psi, and both types of gauges failed at about 2700 μ in./in. strain in the capsule walls (3000 psi internal pressure).

Subsequent static tests have indicated that the strain gauges fail as a result of the shock associated with the dynamic strain and not as a result of installation techniques.

The pressure was measured at several positions outside the capsule using crystal hydrophones to determine the characteristics of the pressure pulses which are transmitted through the capsule walls. Pressures were also measured at identical positions when the exploding wire was not surrounded by a capsule. Results of these tests are summarized in Table IV. The presence of the capsule

TABLE III

RESPONSE OF FOIL, SEMICONDUCTOR, AND WELDABLE
STRAIN GAUGES TO DYNAMIC PRESSURE (FIGURE 13)^[a]

Type of Gauge	Maximum Strain ($\mu\text{in./in.}$)	Strain Indicated Peak Pressure (psi)	Rise Time of Strain (μsec)	Maximum Strain Rate (in./in./sec)	Duration of Strain (μsec)
Foil	1270	1400	10	120	50
Semiconductor	1310	1460	8	107	55
Weldable	1380	1530	10	66	55

[a] Hydrophone measurements; peak pressure 1460 psi, rise time 3 μsec , duration 67 μsec .

TABLE IV

CHARACTERISTICS OF PRESSURE PULSES MEASURED BOTH WITH AND
WITHOUT A CAPSULE WALL BETWEEN THE PRESSURE SOURCE AND THE HYDROPHONES

Radial Distance to Wire (inches)	Measured through Capsule Wall			Measured at a Corresponding Position from an Exploding Wire which was not Enclosed in a Capsule		
	2	6	10	2	6	10
Peak Pressure (psi)	285	135	80	595	360	290
Duration (μsec)	19	19	19	28	28	28
Rise Time (μsec)	3	3	3	3	3	3

was found to decrease the peak pressure two inches from the exploding wire by a factor of about 2 and to decrease the duration of the pulse by a factor of about 1.5. Figure 17 shows the pressure measured two inches from the exploding wire when the capsule was in place, and Figure 18 shows the pressure measured at the same position with the capsule removed.

3.3 Results of Primacord Tests

Capsules used for the Primacord tests were five feet long with a 2.375-inch ID and 0.164-inch-thick walls. In order to mock up the geometry of the CDC test space, these capsules were placed in a four-inch-OD stainless steel tube which had 0.25-inch-thick walls. Crystal hydrophones were used to measure pressure

Fig. 17 Measured pressure at 1.0 inch outside the capsule wall (2.0 inch from exploding wire).

Fig. 18 Measured pressure at 2.0 inch from exploding wire which was not encapsulated.

within the capsule, between the capsule and stainless steel tube, and outside the tube. Figure 19 indicates the positions of the hydrophones. Four types of strain gauges were mounted on the outside of the capsules: these were weldable, semiconductor, foil, and wire gauges. All except the weldable gauges were attached to the capsule with Eastman 910 cement. A weldable strain gauge was also attached to the outside of the four-inch-OD stainless steel tube which enclosed the capsules.

Initial tests using Primacord were performed with no stainless steel tubing enclosing the explosive, and pressure pulses having peaks of about 4500 psi were produced. It was found during these initial tests that the four-inch-OD stainless steel tube with 0.25-inch-thick walls decreased the transmitted pressure pulse by a factor of about 10 and also that the weldable strain gauge attached to the outside of the tube gives good indications of the pressure inside the tube when the pressure exceeds about 2000 psi. However, the 4500 psi pressure pulse deformed the 0.164-inch-thick walls of the aluminum capsule; therefore, stainless steel tubing was used to enclose the Primacord and reduce the pressure in subsequent tests.

The presence of the four-inch-OD stainless steel tube does affect the pressure measured in the annulus between the capsule and tube. During a number of tests, the tube was removed and the pressure measured immediately outside the capsule. A 60 percent reduction in peak pressure measured outside the capsule was observed in these tests.

Fig. 19 Section of test assembly showing location of Primacord, pressure-reducing steel tube, aluminum capsule, and steel experimental tube.

Figure 20 demonstrates that the capsule walls affect (to a good approximation) only the magnitude of the pressure pulse and not its shape. Figure 20a shows the output from a hydrophone located inside the capsule, and Figure 20b shows the output from a hydrophone located outside the capsule. The pressure outside the capsule between the capsule walls and the stainless steel tube was measured over a wide range of internal capsule pressures, and a logarithmic relationship was found between them. Figure 21 shows the experimental data obtained during these tests.

Fig. 20 Pressure measured by hydrophone (a) inside aluminum capsule and (b) outside of aluminum capsule.

Fig. 21 Logarithmic plot of pressure measured by hydrophones inside and outside 2.0-inch-OD aluminum capsule.

Figure 22 shows the time response of the semiconductor, wire, and weldable strain gauges to the pressure pulse shown in Figure 20a. As can be observed from Figure 22, the strain gauge responses have the same general shape as the pressure pulse. Results of the responses of the three types of strain gauges to various internal capsule pressures are summarized in Table V.

The weldable strain gauges used in these tests were pre-formed to be installed on a cylinder of approximately two-inch OD so that a minimum amount of strain was induced in the gauges during installation. Both the pre-formed weldable and the semiconductor strain gauges indicated internal pressures within 10 percent of the measured pressures in the range of 500 to 3300 psi. Both types of gauges failed for capsule strains above $2200 \mu\text{in./in.}$ (3500 psi capsule pressure).

The wire gauge indicated the capsule pressures to within 15 percent of the hydrophone values. Although not quite as accurate in measuring pressure as the other types of gauges, the wire gauge showed a greater resistance to failure at strain levels above $2000 \mu\text{in./in.}$

3.4 Conclusions

Pre-formed weldable strain gauges attached to the outside of aluminum capsules of the dimensions previously given can be used during in-pile tests to determine the internal dynamic pressure of the capsule to within 10 percent of

Fig. 22 Response of wire, weldable, and semiconductor strain gauges to pressure pulse shown in Figure 20(a).

TABLE V

RESPONSE OF CIRCUMFERENTIAL STRAIN GAUGES TO
DYNAMIC PRESSURE PRODUCED WITHIN A 6061-T6,
2.375-IN.-OD, 0.164-IN.-WALL, 5-FT-LONG ALUMINUM CAPSULE

Type of Gauge	Measured Peak Pressure ^[a] (psi)	Strain-Indicated Peak Pressure (psi)	Maximum Strain ($\mu\text{in./in.}$)	Rise Time of Pressure ^[a] (μsec)	Rise Time of Strain (μsec)	Duration ^[a] of Pressure (μsec)	Duration of Strain (μsec)
Weldable (pre-formed)	3500	3280	2060	10	12	140	150
Semiconductor	2040	2100	1300	10	12	120	140
Wire	1060	920	576	10	14	120	140

[a] Measured by hydrophone.

directly measured pressures. Therefore, it is planned to make extensive use of these gauges to determine internal capsule pressure in the CDC experimental program.

IV. POWER BURST FACILITY

1. PBF IN-PILE FUEL TESTING PROGRAM

In order to proof-test the PBF fuel rod concept, a program of in-pile transient tests is being conducted in the TREAT facility. In these tests, prototype PBF fuel rods are subjected to transient energy densities comparable to, or slightly higher than, those for which the PBF is being designed. The fuel rods being tested are composed of low-enrichment, urania fuel pellets surrounded by stabilized zirconia insulators and stainless steel cladding tubes. To date, five series of tests have been conducted in TREAT. The first four were performed primarily to determine the mechanical stability characteristics of two types of insulators which were considered for use in the PBF fuel: (a) insulators consisting of sleeves of zirconia fitted around the fuel pellets and (b) insulators consisting of a coating of zirconia sprayed directly on the fuel pellet circumference. As a result of these tests, it was concluded that the sleeved insulators are preferable for use in the PBF. These results have been previously reported [17, 18].

During this quarter, a fifth series of tests was conducted in TREAT to determine the effectiveness of annular helium gas gaps between the fuel pellets and sleeved insulators (minimum ≈ 6 mils) and between the insulators and cladding (minimum of ≈ 8 mils) in preventing clad growth as a result of fuel pellet expansion during transient tests [a]. For these tests a test fuel rod was subjected to two 130-msec-period transient tests in the TREAT reactor. The test rod was composed of 18 pellet-insulator combinations and had an active length of 18 inches and a 0.75-inch OD. Strain gauges were attached to this rod to provide a measurement of the cladding growth during the transients.

Results from these tests indicated a diametral cladding growth of approximately 1 mil which, on the basis of measured temperatures, can be attributed solely to thermal expansion of the clad. Destructive analysis of the fuel following the tests indicated that the fuel received a peak energy density of 260 cal/g, or about 125 percent of the PBF design peak energy density. Thus it is concluded that such annular gaps within the fuel rod would be adequate to prevent permanent clad growth at the PBF design temperature.

Metallographic examination of the fuel showed no evidence of melting, grain growth, fusion between pellets, or structural change. As in the previous four series of tests, both the fuel pellets and insulators were cracked and broken during these tests, but the insulators maintained an effective thermal barrier between the fuel and clad. Figure 23 shows a post-test photograph of a section of the rod used in the fifth series of tests.

[a] Recent results from the PBF fuel testing program have indicated that a fuel rod design incorporating a single gas-gap between the fuel and insulator and an interference-fit between the clad and insulator may be a better design for the PBF.

Fig. 23 Photograph of section through PBF fuel rod used in TREAT test series No. 5 showing insulating barrier between fuel and clad.

2. OUT-OF-PILE TESTS OF PBF TERNARY FUEL MIXTURE

The fuel for the PBF consists of a ternary mixture of urania, zirconia, and calcia containing 60 to 70 volume percent urania. This material is pressed into pellets and sintered to form a single-phase, face-centered-cubic solid solution. Because of a lack of published data, a program to determine the thermal characteristics of this fuel has been conducted. In this program, tests to determine the phase stability, thermal conductivity, and melting points of these mixtures have been conducted by Combustion Engineering under a subcontract with Phillips Petroleum Company.

2.1 Phase Stability Tests

The phase stability of the fuel during prolonged exposure to high temperature has been investigated for samples having 60 to 70 volume percent urania. Test samples were placed in tantalum capsules filled with helium gas and then sealed. Three test conditions were established; viz, 1000 and 1300°C constant temperature exposure for 7 days and 2000°C constant temperature exposure for 24 hours. At the conclusion of each test, the respective test sample

was examined by X-ray diffraction to determine the lattice structure, and photomicrographs of the samples were obtained and analyzed to provide further information relative to the phase stability of the samples.

2.11 X-Ray Diffraction Analysis. In preparation for the phase stability tests, the lattice parameters of the constituents in the ternary mixtures were determined by X-ray diffraction analysis. The parameters were 5.467 Å for the urania and 5.1407 Å for the stabilized zirconia, with no extraneous lines being evident for either of the starting samples.

Holding the material at elevated temperatures for extended periods of time had no significant effect on either the lattice dimensions or upon the composition of the solid solution. With the single exception of the 60 volume percent sample tested at 1300°C for 7 days, the post-test diffraction patterns were sharp and clear with no extraneous lines. The 60 volume percent 1300°C test sample did not remain as a homogeneous, single-phase material. From the diffraction patterns it appeared that the composition partially (less than five percent) broke down into two solid solutions: one slightly enriched in zirconia and the other slightly enriched in urania. However, the diffraction lines were not sufficiently strong to make an absolute determination and were somewhat masked by the primary lines. A microprobe examination is currently being made in an effort to confirm the composition of the two solutions observed.

2.12 Microscopy. Comparison of pretest and post-test photomicrographs revealed the formation of three compositions which were not originally present in the samples: (a) a lamellar phase which formed preferentially along the grain boundaries; (b) a metallic phase which formed preferentially on pore surfaces within the material; and (c) a fine precipitate within the grains normally located near the outside surfaces of the pellets. Results of the individual tests are briefly outlined below.

1000°C -- 7 days There was no change observed in either composition (60 to 70 volume percent urania) relative to the control specimens.

1300°C -- 7 days A discontinuous lamellar phase was observed which advanced toward the center of the grains from the grain boundaries and was more prevalent in the 60 volume percent urania composition. Trace amounts of a fine precipitate were observed within the grains for both samples, being more prevalent in the 70 volume percent urania composition. The precipitate was confined to a narrow band adjacent to the outer surface of the pellets. Photomicrographs of the test samples are shown in Figure 24.

2000°C -- 24 hours A fine precipitate was observed within the grains adjacent to the outer surface of the pellets for both compositions. Concentration of this precipitate was higher for the 70 volume percent urania composition which also had trace quantities of the precipitate in the central regions of the pellet. Trace amounts of a metallic phase were observed on pore surfaces in both samples. This phase tarnished rapidly upon contact with air and was completely removed by etching. The metallic phase was more prevalent in the 70 volume percent urania composition.

Fig. 24 PBF ternary fuel samples tested at 1300°C for 7 days to determine phase stability of the mixture; 500X etched.

2.13 Conclusions. When the ternary PBF fuel is held at 1300°C or above for extended periods of time three distinct changes are observed.

- (1) Formation of a lamellar phase along grain boundaries, which probably consists of urania rich and zirconia rich phases
- (2) Formation of a metallic phase on pore surfaces
- (3) Formation of a fine precipitate in a narrow band adjacent to the outer surface of the pellet.

All phases remained face-centered-cubic and no dimensional changes exceeding the limits of accuracy of the measurements were observed except for a 0.3 percent volume decrease in the 2000°C samples for which the test temperature exceeded the sintering temperature (1800°C). From the information available from both the diffraction patterns and the photomicrographs and from the observed dimensional stability of the fuel, it is concluded that the phase stability of the ternary PBF fuel is sufficient for use in the PBF over the design lifetime.

2.2 Thermal Conductivity Determination

The thermal conductivity of the ternary fuel was determined by a radial-heat-flow method. A stack of cylindrical fuel pellets was mounted on a coaxial electrical heater within a stainless steel cladding tube as shown schematically in Figure 25. Viewing ports, provided to allow temperature measurements by optical pyrometry techniques at various radial positions in the assembly, and a water-cooled jacket (not shown in the figure) completed the assembly. The electrical heater was used to centrally heat the assembly, and the thermal conductivity was evaluated from the temperature distribution as a function of time. The fuel temperature was varied from about 800 to 2200°C by control of the input to the electrical heater.

Fig. 25 Schematic of experimental arrangement for the determination of the thermal conductivity of the PBF ternary fuel mixture.

The thermal conductivities as obtained from the observed temperature distributions within the fuel are shown in Figure 26. The data display minima of about 0.0175 W/cm-°C in the vicinity of 1400 to 1700°C and maxima of about 0.035 W/cm-°C at the highest test temperatures attained (2000 to 2200°C).

2.3 Melting Point Determination

The solidus and liquidus temperatures of the ternary fuel were determined by heating a long rectangular sliver of fuel and measuring the temperatures at which certain physical changes occurred. The sliver of fuel was clamped at one end and extended into a tubular heater; the sample and heater being enclosed in a glass container to provide a protective atmosphere of helium gas. The solidus temperature was determined as the temperature at which the sample shape changed from sharp-cornered to rounded -- the liquidus as the temperature at which the entire tip of the sample became globular and

Fig. 26 Thermal conductivity versus temperature for PBF ternary fuel mixtures.

larger in diameter than the original sliver width. The temperatures were found to be:

60 Vol % UO_2	Solidus	2451°C
	Liquidus	2549°C
70 Vol % UO_2	Solidus	2451°C
	Liquidus	2576°C

V. SPERT PUBLICATIONS DURING THIRD AND FOURTH QUARTERS OF 1964

1. INTRODUCTION

This section presents a listing and short summary of the reports issued and the papers presented at national technical meetings by the Spert Staff during the third and fourth quarters of 1964.

2. AEC REPORTS

(1) IDO-16989, Self-Limiting Excursion Tests of a Highly Enriched Plate-Type, D₂O-Moderated Reactor Part III: Effect of System Temperature, R. C. Schmitt and J. E. Grund (issued July 1964).

Self-limiting power excursion tests have been performed to investigate the effect of initial system temperature on the dynamic response of the Spert II Expanded Core. This D₂O-moderated core was comprised of 3-inch-square by 24-inch-long, highly enriched, uranium-aluminum, plate-type fuel assemblies spaced on six-inch centers. The power excursions were initiated by step reactivity insertions with the system at various initial temperatures, near atmospheric pressure, with no forced coolant flow, and both with and without a moderator expansion space (gas dome) in the reactor vessel. The reactivity insertions ranged from about 50 cents to 3.7 dollars and resulted in initial asymptotic reactor periods from about 5 sec to 37 msec.

These tests demonstrated that increasing the initial system temperature toward the system saturation temperature permits steam formation to occur earlier than in comparative ambient temperature or pressurized tests. The increased dependence on steam formation for shutdown was reflected in a reduction in peak powers, energy releases, and fuel plate surface temperature rises. Also demonstrated was an increased margin of safety to the reactor. With the system at higher initial temperatures, larger reactivity additions could be safely withstood without damage to the core. Whereas earlier work has indicated that the onset of core damage occurred for ambient temperature tests with initial periods of about 70 msec, tests with the system temperature at 100°C and having asymptotic periods as short as 37 msec were successfully conducted in this test series without observable core damage.

(2) IDO-16990, Self-Limiting Excursion Tests of the Spert II Highly Enriched, Plate-Type, D₂O-Moderated Reactor Part IV: Data Summary Report, V. W. Goldsberry (issued September 1964).

This report, the fourth part of a four part series, is a documentation of experimental data from three series of tests performed in the Spert II reactor with D₂O moderator and highly enriched fuel. The tests consisted of stepwise insertions of excess reactivity under varying conditions ranging from ambient temperatures to $\approx 100^{\circ}\text{C}$ and static pressure from atmospheric to ≈ 375 psig. The data obtained from these tests are presented in tabular form and as plots

of the time behavior of reactor power, fuelplate surface temperature, pressure, compensated reactivity, and energy. Summary data figures displaying measured test data as functions of asymptotic reactor period also are included.

(3) IDO-16997, Heavy Water Handling Experience in the Spert II Reactor Facility, C. R. Montgomery (issued July 1964).

The Spert II reactor is a pressurized-water research reactor which was constructed to provide a facility for conducting reactor kinetic behavior and safety studies. The facility provides for the alternate use of both light water and heavy water as coolant-moderators, and also can accommodate frequent changes in core types and configurations. This report describes the particular design features of the facility which provide for the conservation of heavy water and the handling techniques developed in order to minimize losses and maintain isotopic purity of the heavy water inventory.

(4) IDO-17000, Reactor Power Excursion Tests in the Spert IV Facility, J. G. Crocker and L. A. Stephan (issued August 1964).

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

(5) IDO-17006, Diffusion in the Stabilized Zirconia-Urania System, D. R. Mash and K. K. Ikeuye, Reese S. Kern, Editor (issued November 1964).

Diffusion in the stabilized zirconia-uranium system has been investigated by the "moving" boundary (metallographic) technique. The objective of the investigation was to measure penetration of zirconia cladding by uranium at elevated temperatures.

Natural UO_2 cored pellets were coated by the plasma-spray technique. Uniform coatings of about 89 percent of theoretical density were applied using carefully controlled process conditions. Diffusion annealing treatments were carried out in the 2000 to 2450°C range for eight-hour periods in helium. Diffusion zone characteristics, observed metallographically, provided evidence of relatively complicated interactions taking place in this ternary system at temperatures close to the melting point of the coating. The ZrO_2 - UO_2 interface region after diffusion included a void zone, a ternary solid solution zone, the "moving" boundary, and a high density zirconia-rich zone. The long term influence of each of these zones on the diffusion process, as well as a quantitative understanding of rate controlling factors associated with each zone, was beyond the scope of the current investigation.

Based on this investigation, it can be concluded that diffusion in the stabilized ZrO_2 - UO_2 system is strongly temperature dependent; however, even at temperatures close to the melting point of zirconia, uranium would not completely penetrate a 0.050-inch coating after eight hours at temperature.

The investigation described in this report was performed by Advanced Materials and Processes, 1890 Embarcadero Road, Palo Alto, California, under Subcontract No. C-251 B to Phillips Petroleum Company.

(6) IDO-17011, Transient Tests of the Fully Enriched, Stainless Steel, Plate-Type, P Core in the Spert I Reactor: Data Summary Report, A. P. Wing (issued December 1964).

This report presents the data obtained from the transient test series performed on the Spert I, stainless steel, plate-type P core. The tests consisted of step insertions of reactivity under varying conditions ranging from ambient temperature to $\approx 97^\circ\text{C}$ and resulted in asymptotic reactor periods ranging from 6.0 sec to 4.0 msec. The experimental data obtained from each test are presented in both tabular and graphical form. The graphs include time-history plots of reactor power, fuel plate surface temperature, and pressure. Time-history composite graphs displaying fuel plate surface temperature as a function of reactor core position are included. Correlated graphs displaying test data as functions of asymptotic reactor period also are included.

(7) IDO-17028, Experimental Results of Potentially Destructive Reactivity Additions to an Oxide Core, J. E. Grund, Editor (Issued December 1964).

The results of two short-period, potentially destructive transient tests performed with a low-enrichment, UO_2 -powder, stainless steel clad fuel rod core are presented. The tests were initiated by 2.6- and 3.3-dollar reactivity insertions. In both tests, two fuel rods ruptured and dispersed UO_2 into the water. Approximately 25 percent of the fuel rods also were mechanically deformed or discolored. No appreciable damage to the core structure occurred as a result of the excursions, and only minor fission product release from the ruptured fuel rods was determined. The course of the excursions was predictable from experiment and theory until the fuel rods ruptured. The rupture of the fuel rods provided additional shutdown to that available from Doppler effects and reduced the expected energy release of the excursions. The cause of the fuel rod ruptures is suspected to have been from "waterlogging". Details of instrumentation for the tests and analysis of the experimental results are presented in Appendixes.

3. PAPERS PRESENTED AT NATIONAL TECHNICAL MEETINGS

The following papers were presented at the winter meeting of the American Nuclear Society in San Francisco, California, November 30-December 3, 1964.

- (1) "The Influence of System Parameters on the Dynamic Response of a D₂O-Moderated Reactor", R. L. Johnson and J. A. Norberg, Trans. Am. Nuc. Soc., 7, No. 2, (1964) p 500.

This paper presented a summary of the results obtained from self-limiting power excursion tests performed in the Spert II reactor with the close-packed, D₂O-moderated, -reflected, and -cooled core. The tests were initiated by step insertions of reactivity, with various combinations of initial system conditions: pressure (0 to 360 psig), temperature (70 to 195°F), and forced coolant flow (0 to 14 fps). The information presented appears for the most part in two previously published quarterly reports [17, 22].

- (2) "An Experimental Investigation into the Threshold of Departure from Nucleate Boiling (DNB) in a D₂O-Moderated Reactor", J. A. Norberg and R. L. Johnson, Trans. Am. Nuc. Soc., 7, No. 2, (1964), p 504.

This paper described an experimental investigation conducted in the Spert II facility with the close-packed, D₂O core to determine the threshold power level for the departure from nucleate boiling at initial system conditions of 200-psig pressure, 80°F inlet temperature, and 7-fps coolant velocity through the core. Step- and ramp-type reactivity insertions resulting in reactor periods ranging from approximately 400 to 100 msec were used to study the effect of power history on the threshold of DNB. Two previously published quarterly reports [17, 22] summarized the major portion of the information presented.

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