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QUARTERLY TECHNICAL REPORT
SPERT PROJECT
OCTOBER, NOVEMBER, DECEMBER 1967

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QUARTERLY TECHNICAL REPORT
SPERT PROJECT
OCTOBER, NOVEMBER, DECEMBER 1967

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<u>Quarter</u>	<u>1960</u>	<u>Number</u>
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<u>Quarter</u>	<u>1961</u>	<u>Number</u>
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2		IDO-17207
3		IDO-17228
4		IDO-17245

<u>Quarter</u>	<u>1967</u>	<u>Number</u>
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4		IDO-17279

SUMMARY

SPERT III -- OXIDE CORE KINETICS PROGRAM

Reactor kinetic behavior is being studied with the Spert III E-core, a slightly enriched UO_2 -fueled core, under environmental conditions typical of pressurized water reactors. The objectives of this test program are to acquire experimental data on the kinetic behavior of UO_2 -fueled reactors, and to demonstrate the capability of calculational models to predict such behavior.

The two low-initial-power series comprising the first phase of the E-core test program were completed during the first and second quarters of 1967. Transient testing with the E-core continued this quarter (October-December) from initial power levels of about 1 MW and 20 MW. The initial system conditions were 500°F, 1500 psi, and a coolant flow velocity of 14 fps. The purpose of these elevated-initial-power tests was to obtain (a) improved neutron detector calibration constants, and (b) experimental data for the determination of the transient behavior of the core with initial reactor power levels up to 20 MW.

All previously published values of the magnitude of reactor power and energy obtained during transient testing with the E-core have been based on neutron detector calibration constants that were calculated using flux wire activation data. One standard deviation in absolute reactor power using these constants is estimated to be $\approx 15\%$. In order to improve the accuracy of the constants, heat balance experiments were performed this quarter for comparison with flux wire data. As a result of these experiments, the standard deviation in absolute power for all tests performed at 500°F initial temperature was reduced to $\pm 10\%$. Analysis of the revised calibration constants indicates that all previous constants calculated from flux wire data at an initial system temperature of 500°F should be increased by 10% for all initial power levels up to 20 MW.

Abnormal operation was experienced during the performance of the third transient performed from an initial power level of 20 MW. The control rods dropped into the core more slowly than had been expected during the scram following completion of the test. The primary system coolant was found to contain gaseous fission products, and the available reactivity of the core at ambient temperature increased 0.18\$ compared with the conditions prior to this test series. Investigations into these anomalies thus far indicate that:

- (1) The longer-than-normal control rod drop time was associated with power operation, is reversible, and is not affecting the reactor at this time.
- (2) Gaseous fission products are getting into the primary coolant through small holes in the clad of one or more fuel rods; there were no particulates in the coolant.
- (3) An observed change in critical position that occurred prior to the start of the elevated-initial-power tests is attributed to $\approx 1\%$ difference in the reproducibility of the temperature experiment resulting from a wiring change. Investigations are continuing to determine why the critical positions following the final 20 MW test were, unpredictably, equivalent to those before the wiring change.

The system will be opened and selected core components inspected next quarter in order to further investigate the possible causes and effects of the observed anomalies.

SPERT IV CAPSULE DRIVER CORE -- SUBASSEMBLY

TEST PROGRAM

The Subassembly Test Program now under way in the Spert IV Capsule Driver Core (CDC) is designed to investigate the accident conditions that would be expected to exist in operating nuclear reactors. Small samples are irradiated in a controlled environment within a test space centrally located in the core to produce heating rates, power distributions, and radiation levels typical of a reactor accident.

The initial series of the CDC program of fuel testing continued this quarter with performance tests on encapsulated F-type fuel rods. The F-type fuel rods consist of 4.8%-enriched, pelletized, UO_2 fuel clad with 0.466-inch-OD Type 304 stainless steel. Both waterlogged and nonwaterlogged rods were tested with the objective of establishing the approximate energy deposition required to damage or cause failure of the rods and to obtain scoping data on axial cladding growth and bowing of the rods when subjected to repeated tests at energy densities in the range from 200 to 250 cal/g of UO_2 .

Waterlogged rods were subjected to single power bursts ranging in energy deposition from 153 to 283 cal/g of UO_2 . Diametrical growth resulting from internal pressure generated during the power transient was shown to increase with increased energy. The fuel rod exposed to the highest energy deposition (283 cal/g) ruptured at an energy of 250 cal/g, expelling about 75% of the fuel.

The particle size distribution of the expelled fragmented fuel was determined by screening and weighing the residue. The surface areas of the fragments were calculated from the results of the screening. The surface-to-volume ratio of the expelled fuel was a factor of about five less than that of powder fuel from previously tested waterlogged CDC powder-fuel rods. This large heat transfer area for the powder fuel is believed to be the reason the powder-fuel rods failed at lower energy depositions (40 cal/g vice 250 cal/g of UO_2) and produced higher capsule pressures (10,000 psi vice 1300 psi) than did the pellet-fuel rods. The fact that the powder rods were clad with cold-worked Type 304 stainless steel whereas the pellet rods were clad with annealed Type 304 stainless steel may also have contributed, but only to a minor degree, to the different failure characteristics.

Scoping tests were initiated this quarter on dry (nonwaterlogged) F-type rods to obtain data on the behavior of these rods when subjected to repeated nondestructive tests at energies in the 200 to 250 cal/g range. Cumulative measurements of permanent axial cladding growth, rod bowing, and cladding deterioration were the main objectives of the tests. Progressive cladding growth and rod bowing were found to be the main effects of transients in these energies. Although the cladding was significantly discolored, there was no evidence of cladding failure.

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I. SPERT III -- OXIDE CORE KINETICS PROGRAM (R. K. McCardell, D. I. Herborn, W. A. Eisenbarth)

1. INTRODUCTION

The Spert III E-core, a slightly enriched UO_2 core, is being used in the Oxide Core Kinetics Program to experimentally study reactor kinetic behavior under environmental conditions representative of those of commercial pressurized-water-power reactors. The two initial test series comprising the first phase of the E-core test program have been reported [1,2]. These test series consisted of low-initial-power, step-reactivity-induced power excursion tests with initial system temperature from ambient to 500°F, pressures from atmospheric to 1500 psi, coolant flow rates from 0 to 21 fps, and reactivity insertions up to 1.2\$. The data obtained allow comparisons of power excursions initiated from ambient-temperature (representative of a cold-startup accident) and elevated-temperature (representative of a hot-startup accident) conditions. The experimental data have also provided a basis for verification of the Spert-developed PARET computer code, a program for the analysis of reactor transients, as an excellent analytical tool for predicting the kinetic response of a UO_2 -fueled reactor over a broad range of initial environmental conditions.

The elevated-initial-power testing phase of the Spert III E-core experimental program was initiated this quarter (October-December 1967). The objectives of this phase of testing were to obtain improved neutron detector calibration constants for reactor power levels up to 20 MW, and to obtain experimental data for the determination of the transient behavior of the core with initial reactor power levels up to 20 MW. Initial system conditions were 500°F temperature, 1500 psi, 14 fps coolant flow rate, and power levels of about 1 and 20 MW.

During the third transient performed from an initial power level of 20 MW the control rods dropped into the core more slowly than had been expected during the scram following completion of the test. The primary system coolant was subsequently found to contain gaseous fission products, and it was determined that at ambient temperature the core had about 0.18\$ more reactivity available than it had prior to the elevated power tests. A number of possible causes of these three unexpected results are being evaluated in order to permit a realistic determination of possible limitations for continued testing with the core.

2. ELEVATED INITIAL POWER TESTS -- NEUTRON DETECTOR CALIBRATIONS

All previously published values of the magnitude of reactor power and energy obtained during transient testing of the E-core have been based on neutron-detector calibration constants that were calculated using flux wire activation data [1,2,3,4]. On standard deviation in the calibration constants obtained from these data is estimated to be $\pm 13\%$. This uncertainty, when coupled with an estimated $\pm 7\%$ precision in the reactor power data, leads to

a standard deviation in absolute reactor power of $\pm 15\%$. In order to improve the accuracy of the neutron detector calibration constants, heat balance experiments were performed during high-power testing in Spert III. One standard deviation in the constants was thereby reduced to $\pm 6\%$ for all tests performed at 500°F initial system temperature. Thus, the uncertainty in absolute power for all tests performed at 500°F initial system temperature was reduced to $\pm 10\%$.

Reactor power determinations have been performed using four basic techniques: (a) flux wire irradiation, (b) low-power, primary system heat balance (temperature coastdown), (c) high-power, primary system heat balance (core flow- ΔT) and (d) high-power, secondary system heat balance (heat exchanger heat balance). Analysis of these techniques included evaluations of the uncertainties associated with each technique. The objective of the uncertainty analysis is to point out the sources of uncertainty in order to qualify the results of each technique under the conditions of the Spert III calibrations.

During the high-power, secondary system heat balance experiments, malfunctions of experimental apparatus prevented obtaining usable experimental data. The basic method, uncertainties, and test procedure for each of the remaining three calibration techniques are discussed in the following sections.

2.1 Flux Wire Activation Experiments

Thin cobalt wires were activated in Spert III to determine both reactor power and spatial flux distributions. The technique was based on known neutron absorption cross sections and decay radiation properties of cobalt. After irradiation, small segments of the wires were scanned with a radiation detector and corrections were made for the decay time since irradiation. The corrected data were then determined using the activation time and the appropriate fission cross sections.

For previous neutron detector constants calculated at Spert using flux wire activation data, it has been assumed that fissioning occurred only in U-235 for thermal neutron energies. This is a reasonable assumption for thermal reactors; however, considerable epithermal fissioning occurs in the E-core. Therefore, corrections to account for epithermal fissioning have been made in the equation previously used to calculate reactor power from flux wire activation data.

The counting rate obtained with a gamma counter for a cobalt wire used to determine the neutron flux associated with the volume element V_j in the core is given by

$$C_j = KN_o \lambda e^{-\lambda t'} \int_t \int_u \sigma_a(V_j, u) \phi(V_j, u, t) du dt \quad (1)$$

where it has been assumed that the atomic number density of cobalt, N_o , does not change appreciably with time; and

K = a constant taking into account the counter efficiency and geometry

λ = the decay constant for cobalt

$\sigma_a(V_j, u)$ = the microscopic absorption cross section for cobalt as a function of position in the core and lethargy

$\phi(V_j, u, t)$ = the neutron flux in the core during activation as a function of position, lethargy, and time of activation, t

t' = time between activation and counting.

Assuming that the neutron flux is constant during activation, that σ_a is not position dependent, and if the multigroup diffusion approximation is used, Equation (1) becomes

$$C_j = KN_o \lambda e^{-\lambda t'} t \sum_{i=1}^4 \sigma_{ai} \phi_i(V_j) \quad (2)$$

when four lethargy groups are used. If the flux shape is not lethargy dependent, then $\phi_i(V_j) = n_i \phi_{th}(V_j)$; viz, the flux in any lethargy group, i , is a constant times the thermal flux. From Equation (2), the thermal flux in a given volume element of the core during activation is

$$\phi_{th}(V_j) = \frac{C_j}{KN_o \lambda e^{-\lambda t'} t \sum_{i=1}^4 \sigma_{ai} n_i}.$$

The average thermal flux in the core during activation is

$$\bar{\phi}_{th} = \left(\frac{1}{KN_o \lambda e^{-\lambda t'} t \sum_{i=1}^4 \sigma_{ai} n_i} \right) \frac{\int C_j dV_j}{\int dV_j}$$

or

$$\bar{\phi}_{th} = \left(\frac{1}{KN_o \lambda e^{-\lambda t'} t \sum_{i=1}^4 \sigma_{ai} n_i} \right) \frac{\sum_{j=1}^k C_j V_j}{V_T} \quad (3)$$

where

k = the number of wires activated

V_T = the volume of the reactor core.

Again, if the multigroup diffusion approximation is used, the microscopic fission cross sections of the core nuclides are assumed to be independent of

position, and if the flux shape is assumed to be independent of lethargy, then the reactor power during activation is given by

$$P = K_1 \phi_{th} V_T N' \sum_{i=1}^4 \sigma_{fi} n_i \quad (4)$$

where

K_1 = a constant ($\frac{\text{watt-sec}}{\text{fission}}$)

N' = the number density of the fission nuclides in the core

σ_{fi} = homogenized microscopic fission cross section for the i th lethargy group.

Substituting ϕ_{th} from Equation (3) into Equation (4) yields

$$P = \left(\frac{K_1 \sum_{j=1}^k C_j V_j}{K \lambda e^{-\lambda t'_t}} \right) \sum_{i=1}^4 \left(\frac{N' \sigma_{fi}}{N_o \sigma_{ai}} \right) \quad (5)$$

The equation previously used to calculate reactor power from flux wire activation data from thermal reactors was

$$P = \left(\frac{K_1 \sum_{j=1}^k C_j V_j}{K \lambda e^{-\lambda t'_t}} \right) \Sigma_1^f$$

where Σ_1^f is the macroscopic fission cross section for U-235.

The test procedure for preliminary power calibration using flux wire activations consisted of (a) installing cobalt wires in fuel assemblies, (b) irradiating the wires for about 20 minutes at a low reactor power, (c) removing the wires from the core, and (d) gamma scanning the wires. The calibrations were performed for reactor conditions of 70°F system temperature and atmospheric pressure with no coolant flow, 400°F and 1725 psi with 23 fps flow, and 500°F and 1700 psi with 23 fps flow. The volume of the assemblies and the microscopic fission and absorption cross section which were obtained from the computer codes PHROG[a] and TOTEM[b], were then used in Equation (5) to calculate reactor power.

[a] The PHROG computer code, a code for calculating fast diffusion constants, is a revised version (unpublished) of GAM[5].

[b] TOTEM, a code for computing thermal diffusion constants, is a combination of TOPIC[6] and TEMPEST[7] computer programs.

Several inherent problems exist in the use of the flux wire irradiation technique for power calibrations. The location of the flux wire in the core, for example, was known only within about ± 0.05 cm, and flux gradients were large near the surface of fuel rods. Therefore, the ratio of the flux density at the wire surface to the flux density at an average point in the fuel rod was known only to within $\pm 5\%$. The nuclear constants that were applied to the wire irradiation data (data which nominally have a statistical counting uncertainty of about 5%) have an estimated uncertainty of about $\pm 2\%$. The resulting estimated fractional standard deviation in a single power calibration obtained from flux wire irradiation was estimated to be $\pm 13\%$.

Calibration constants calculated from flux wire data for neutron detectors, designated GE-46, GE-51, and Op Lin, and for the various initial system temperatures are listed in Table I. Since heat balance experiments were performed at an initial system temperature of 500°F, more accurate neutron detector constants were obtained for this system temperature and are discussed in Sections 2.2, 2.3, and 2.4. However, the neutron detector constants listed in Table I for system temperatures of ambient, 250°, and 400°F are the only data obtained at these temperatures.

TABLE I
NEUTRON DETECTOR CALIBRATION CONSTANTS OBTAINED BY
FLUX WIRE ACTIVATION

System Temperature (F°)	Detector Constant (MW/ μ amp)		
	GE-46	GE-51	Op Lin
Ambient	5.5	4.8	0.16
250°	4.5	4.0	0.13
400°	3.4	3.0	0.097
500°	2.5	2.2	0.070

2.2 Low-Power Primary System Heat Balance Experiments

The low-power primary system heat balance (temperature coastdown) technique for determining reactor power was performed at reactor power levels of about 1 MW. Heat balances were performed for two initially quasi-steady-state system conditions, one of which included reactor power operation.

In the first quasi-steady-state system condition, the system was operated such that the power produced by the reactor, S_n , plus the power generated in the coolant by the pumps, S_p , was approximately equal to the power extracted from the primary coolant by the heat exchangers, S_{HX} , plus the power lost from the coolant by conduction, S_c , and radiation, S_R . For this situation, a heat balance was written

$$\sum_i S_i - MC_p \dot{T}_1 = 0 \quad (6)$$

where the S_i includes all power sources and losses, MC_p is the effective energy storage of the primary system, and $\dot{T}_1$ is the time rate of the change of the primary coolant temperature.

In the second quasi-steady-state system condition, the reactor power was reduced to zero so that the system heat balance became

$$\sum_i S_i - MC_p \dot{T}_2 = 0. \quad (7)$$

The first term of Equation (7) is identical to that of Equation (6) with the exception that S_n is now zero.

Subtraction of Equation (7) from Equation (6) yields a suitable formula for reactor power:

$$MC_p (\dot{T}_1 - \dot{T}_2) = S_n.$$

In this equation the magnitude of S_n was evaluated by using the calculated value of the effective primary system energy storage and the experimentally determined difference in the cooling rate with and without reactor power.

The Spert III primary system was heated to 500°F with the primary coolant pumps and held at this temperature for several hours to establish thermal equilibrium in preparation for the calibrations. An indicated steady-state nuclear power level of approximately 1 MW and a steady-state temperature were then established and maintained for several minutes without any system perturbations. The reactor was then scrammed and the system permitted to cool slowly for several minutes, again without perturbations. The primary coolant temperature was continuously recorded throughout the test, and the coolant temperature change was determined from these recordings. The results are listed in Table II, together with the reactor power indicated by the neutron detectors. Neutron detector constants used to obtain the powers listed in Table II were those calculated using the flux wire activation technique.

The problems inherent in the low-power, temperature-coastdown experiments included (a) difficulty in maintaining long-term steady-state conditions (coolant temperature changes of 10°F were expected), (b) determination of the volume of the water in the primary system to within 2 to 3%, and (c) ability to determine the heat storage of the primary vessel and piping closer than $\pm 20\%$. On the basis of these considerations, the fractional standard deviation due to the various sources of uncertainty in a single, low-power, temperature-coastdown experiment was estimated to be between ± 7 and $\pm 14\%$.

2.3 High-Power Primary System Heat Balance Experiments

The high-power primary system heat balance (core flow- ΔT) technique was based on analysis of primary coolant flow through the core and the temperature increase as the coolant passed through the core. Reactor power was determined from the measured coolant mass flow rate, temperature increase, and heat capacity of the coolant water.

TABLE II
PRIMARY SYSTEM HEAT BALANCE POWER
CALIBRATION RESULTS

<u>No.</u>	<u>Heat Balance (MW)</u>	<u>Op Lin Detector (MW)</u>	<u>GE-46 Detector (MW)</u>	<u>GE-51 Detector (MW)</u>
1 ^[a]	1.1	1.1	1.1	1.2
2 ^[a]	1.2	1.1	1.1	1.2
3 ^[b]	21	19	19	19
4 ^[b]	20	18	18	19
5 ^[b]	21	17	18	18
6 ^[b]	21	17	18	19
7 ^[b]	21	18	18	18
8 ^[b]	21	17	17	18

[a] Flow coastdown technique.

[b] Core flow ΔT technique.

In the steady-state condition the power produced by the reactor appeared as an increase in the coolant temperature as the coolant passed through the core. Reactor power, S_n , was obtained directly from the formula

$$S_n = MC_p \Delta T$$

where M is the measured coolant mass flow rate, C_p is the coolant specific heat capacity, and ΔT is the average coolant temperature increase across the core.

Several core flow- ΔT experiments were performed at a system temperature of 500°F. The results are tabulated in Table II together with the results from the flow-coastdown experiments. During each experiment the reactor was operated at a steady power level of approximately 20 MW for ≈ 20 minutes. Primary coolant temperature measurements were made at the core inlet by a thermocouple located in the bottom tee and at the core outlet by four thermocouples located above the core in the primary vessel. Neutron detector and primary coolant flow information was continuously recorded during power operation of the reactor.

The principal sources of uncertainty were (a) complete mixing of the primary coolant at any particular thermocouple location at the core outlet or at the heat exchanger outlets could not be assured and (b) the uncertainty in the flowmeter calibration was about $\pm 8\%$. The fractional standard deviation for the core flow- ΔT power calibration technique for a single test is estimated to be between ± 15 and $\pm 30\%$.

2.4 Analysis and Conclusions

Analysis of the calibration constants obtained for the Spert III neutron detectors at an initial system temperature of 500°F for 1 MW and 20 MW power levels indicates that previous calibration constants calculated from flux wire activation data should be increased by 10%. These calibration constants can be used for all power levels up to 20 MW at a system temperature of 500°F.

Operation of the reactor at different power levels could have influenced the detector constants in two ways:

- (1) The detectors may be sensitive to the difference in control rod position between low-power and 20 MW. The difference in control rod position was about four inches, and the detectors were all located over nine feet from the edge of the core and at least three feet above the core center line. Therefore, the detector constants would be expected to be smaller for high-power levels than for low-power levels (due to a decreased leakage path to the detectors) if these constants were dependent on control rod position. However, the experimental data for the low-power coastdown and core flow- ΔT measurements indicate that the detector constants are larger for high-power levels than for low-power levels, which is contrary to what would be expected if the chamber constants were control rod position dependent.
- (2) Increased flux leakage at high-power levels (due to decreased moderator density) would be expected to yield smaller detector constants for higher power levels. However, diffusion theory calculations indicated that the flux spectrum in the core (a good index of moderator density and other changes) did not change appreciably between low- and high-power conditions at 500°F system temperature.

It was concluded that the flux wire activation experiment yielded detector constants that were slightly low. Therefore, the results of the low-power coastdown, core flow- ΔT , and flux wire irradiation measurements were used in a weighted averaging technique to obtain new detector constants. Detector constants obtained in this manner are listed in Table III together with their uncertainties, which are the standard deviation of the experimental results about the weighted average.

Comparison of the above data with the data in Table I indicates a 10% increase over the previous 500°F initial system temperature calibration constants with a calculated standard deviation of $\pm 6\%$. Including the precision of $\pm 7\%$ in reactor power leads to an uncertainty of $\pm 10\%$ in absolute reactor power for tests performed from an initial system temperature of 500°F.

TABLE III

NEUTRON DETECTOR
CALIBRATION CONSTANTS
FOR 500°F SYSTEM TEMPERATURE

<u>Neutron Detector</u>	<u>Constant (MW/μamp)</u>
Op Lin	0.078 $\pm$ 0.005
GE-46	2.7 $\pm$ 0.2
GE-51	2.4 $\pm$ 0.2

3. ELEVATED INITIAL POWER TESTS -- KINETIC
BEHAVIOR OF THE E-CORE

Five test transients from an initial power level of $\approx$ 1 MW and three transients from an initial power level of $\approx$ 20 MW were performed this quarter. The series was interrupted at the end of the third 20 MW initial-power transient to permit investigation of unusual operating behavior that was observed during and following the final test. The kinetic behavior of the E-core during these tests and the results of investigations thus far into the anomalous behavior of the core are discussed in the following sections.

3.1 Test Results

The peak reactor power values obtained during the elevated-initial-power tests are listed in Table IV and plotted in Figures 1 and 2 as a function of the reactivity insertion.

TABLE IV

RESULTS FROM ELEVATED-INITIAL-POWER TESTS

<u>Test No.</u>	<u>Initial Power (MW)</u>	<u>Initial Temperature (°F)</u>	<u>Reactivity Insertion (\$)</u>	<u>Peak Power (MW)</u>
T-79	1.1	513	0.86	13
T-80	1.3	506	1.08	120
T-81	0.90	504	1.17	330
T-82	1.2	505	1.29	880
T-83	1.1	504	1.25	620
T-84	19	505	0.46	39
T-85	19	503	0.87	130
T-86	19	502	1.17	610

BR-0942

Fig. 1 Peak power as a function of reactivity insertion for transients initiated from a power level of 1 MW.

BR-0943

Fig. 2 Peak power as a function of reactivity insertion for transients initiated from a power level of 20 MW.

Because there were no asymptotic reactor periods for the high-initial-power transients [a], it was not possible to determine the initial reactivity insertions from the reactor kinetic equations. The reactivity insertions were instead determined as a function of the reactivity worths of the control rods. The reactivity insertion for each test was obtained by integrating the differential control rod worth curve and correcting the results for initial reactor power and coolant inlet temperature variations between the time that the final critical measurements were made and the transient was initiated. The differential control rod worth curve was based upon experimental data from the static and low-initial-power transients tests [1,2,3].

A technique independent of the control-rod worth was developed to determine the experimental reactivity insertion for high-power transients. As a part of this technique, the SPORT [8] computer program, a system for processing reactor transient data on the IBM-7040, is used to derive the net system reactivity as a function of time from the experimental power history of the reactor. To this net system reactivity is added the total compensated reactivity calculated by PARET [9], a program for the analysis of reactor transients from the experimentally determined data. This method yields the time-dependent reactivity insertion. Curves showing the calculated peak power versus reactivity insertion are compared with the experimental results in Figures 1 and 2. The horizontal error bars represent an uncertainty of one standard deviation ($\approx 4\%$ uncertainty) in the reactivity insertion. Also shown in the figures are the data points for the reactivity insertion determined by the SPORT-PARET method described above.

The PARET-calculated peak power curves yield values that are about 10 to 20% lower than the experimental data. For the 1 MW initial power transients, the calculated peak power values are in agreement with the experimental results well within the uncertainties in the reactivity insertion determined from the control rod worth. For the 20 MW initial power transients, however, the calculated peak power values are not quite within the experimental uncertainties. Possible reasons for this discrepancy are being investigated.

3.2 Abnormal Operating Behavior

On November 30, 1967, following performance of tests T-86, the third of three tests initiated from a power level of 20 MW, it was observed that (a) the control rod drop times were longer than expected, (b) fission products were in the water, and (c) the system had an apparent criticality shift. Investigations

[a] During a test excursion an asymptotic reactor period is attained during the power rise only if there are no reactivity changes taking place in the reactor during the power rise. An asymptotic period thus implies that all the reactivity causing the perturbation is inserted and that there is no appreciable reactivity feedback taking place. For low-initial-power transients and fast reactivity insertions there is very little energy deposition during the early part of the transient; therefore, there is a considerable time after reactivity insertion when reactivity feedback is negligible and the reactor power therefore attains an asymptotic period. For reactivity perturbations from high initial reactor power, however, considerable energy is released immediately and appreciable reactivity feedback occurs soon after the initiation of the transient. Thus system reactivity is never constant and an asymptotic period is not reached.

were performed to identify and establish the significance of these anomalies. A synopsis of the investigations into, and evaluation of, each of the anomalies is presented in the following discussion.

3.21 Scram Time. During each transient, the control rod drop times were recorded and subsequently compared with predicted values based on previous rod drop information obtained under similar system conditions. Previous experience with the Spert III system indicated that rod drop times are generally predictable to within about $\pm 20\%$. During test T-86, however, the observed drop time on one rod was about 80% longer than predicted.

Calculations were performed to identify any possible interference between the control rods and control rod guide tubes during elevated-initial-power operation of the facility. The calculations show that combined differential thermal expansion and bowing of the control rods could reduce the clearance between the rods and guides only from 0.025 ± 0.010 inch to 0.0185 ± 0.010 inch. However, the design clearance may not exist, and, if the actual clearances are smaller, the slight growth at high power and during excursions could cause increased rubbing of the control rod rub pads against the guides with a resultant increase in drop time.

As part of the study to establish the cause of the change in control rod insertion time, several reactor scrams have been performed at low-power conditions since the unexpected behavior observed during test T-86. In every case, the scram times were normal. It was therefore concluded that the phenomenon of the long drop time during test T-86 was associated with power operation, is reversible, and is not affecting the reactor at this time.

3.22 Fission Products. Following each excursion, water samples were drawn for gross count rate determinations and fission-product analyses. For tests prior to test T-86, analysis had shown no fission products in the water. An analysis performed immediately following test T-86 showed that an amount of I-131 equal to 1.5% of the inventory of one average fuel rod was present in the primary coolant, but that neither particulate fission products nor fuel was in the water. All postulated causes of the existence of fission products were ruled out except for the possibility of a leaky fuel rod. The absence of particulate products indicates that gross fuel rod rupture did not occur and that the fission products are getting into the water through small holes in the clad of one or more fuel rods.

3.23 Criticality Shifts. Cold critical measurements were performed on December 6 in an operation designed to further investigate the condition of the core following the high-power transient tests. The measurements showed a shift of about 0.12 inch in the control rod bank position at critical (representing increase of about 0.18\$ in available reactivity). Because of this apparent criticality shift, every conceivable possible cause was considered, and all previous criticality data were reviewed and reevaluated. In the course of investigating the apparent reactivity change, criticality checks were performed to detect any change in the critical rod position. The primary coolant system was drained and refilled in order to determine if any substance or temperature stratification in the coolant was causing the shift. No change in criticality occurred during these investigations.

In order to determine just where or when the reactivity gain occurred, the criticality data just prior to each of the elevated-initial-power tests (T-77

through T-86), at an initial system temperature of 500°F, were examined for anomalies. This check revealed that all criticals were within $\approx \pm 0.04\%$. The criticality data from tests T-55 through T-68 (low-power, 500°F tests performed during March and April 1967 [1,2]) were also reevaluated, since the data from these tests had been observed to differ by approximately 0.18% or 5°F (1% of the system temperature) from the criticality data for tests T-77 through T-85.

The observed change in critical position between T-68 and T-77 has been attributed to $\approx 1\%$ difference in the reproducibility of the temperature measurement. This difference is believed to have been the result of wiring changes made after test T-68 to correct noise and drift problems in the temperature measurement system. The assumption that the differences in critical position between T-68 and T-77 were caused by differences in absolute temperature measurement is consistent with the fact that no reactivity change was observed during the ambient temperature criticals prior to T-77. The change in the critical position following T-86 to the position equivalent to critical positions prior to the wiring change is an anomaly, however. The change may have resulted from temperature effects, although there is no conclusive evidence to justify this explanation.

In an effort to further explore the causes of the apparent criticality shifts, the system will be opened and selected core components inspected during the next quarter.

II. SPERT IV CAPSULE DRIVER CORE -- SUBASSEMBLY TEST PROGRAM (L. A. Stephan)

1. INTRODUCTION

The basic objective of the Subassembly Test Program, currently under way in the Spert IV Capsule Driver Core (CDC), is to obtain experimental information on reactor fuel behavior under transient overpower conditions. The CDC provides a high-intensity neutron pulse source for irradiating test fuels at conditions typical of those to be expected in severe reactivity accidents. Activities with the CDC prior to this quarter have included tests to establish the operational capability of the core, subassembly experiments with encapsulated CDC fuel rods, instrumentation tests to develop reliable techniques for the measurement of variables of interest within the capsule, and fuel performance tests on encapsulated F-type fuel rods. Details relating to these experiments are presented in previously published quarterly reports [1-4,10-12].

Fuel performance tests initiated last quarter on encapsulated F-type fuel rods were continued. The F-type fuel rods consist of 4.8%-enriched pelletized, UO_2 fuel clad with 0.466-inch-OD Type 304 annealed stainless steel. Data relative to the physical and nuclear characteristics of the F-type fuel are summarized in the July-September 1967 report [4]. Tests on F-type fuel rods represent the initial efforts in the CDC program of fuel testing, the primary objectives of which are: (a) to determine the energy deposition thresholds of fuel damage and failure, the modes of fuel failure, and the consequences of fuel failure; and (b) to correlate the experimental results in terms of fuel design parameters, burnup and environmental conditions. The specific objectives of the tests performed this quarter were to establish the approximate energy depositions required to damage or cause failure of waterlogged F-type rods, and to obtain scoping data on axial cladding growth and bowing of F-type rods when subjected to repeated tests at energy densities in the 200 to 250 cal/g of UO_2 range.

2. FUEL PERFORMANCE TESTS ON WATERLOGGED, F-TYPE FUEL RODS

Five waterlogged, F-type fuel rods were subjected to single power bursts at reactor periods of 6.8, 4.9, 4.2, 3.5, and 3.2 msec, corresponding to fuel energies of about 153, 181, 223, 257, and 283 cal/g of UO_2 , respectively. The entire void space in the rods was filled with water prior to testing. All rods experienced diametrical growth, with the maximum growth occurring in the region of the highest neutron flux. The magnitude of the growth increased with the energy deposition. Radiographs of the fuel rods which did not fail (energies up to 257 cal/g of UO_2) showed that the fuel in the high-flux region was shattered considerably more than for comparable tests on dry fuel rods.

The rod exposed to 283 cal/g of UO_2 ruptured 13 msec after peak power, expelling about 75% of the fuel from a 2-1/2-inch longitudinal cladding break.

The energy deposition at the time of rupture was ≈ 250 cal/g of UO_2 . The maximum pressure measured at the bottom of the capsule was 1300 psi. Post-test internal examination of the ruptured rod and the rod exposed to 257 cal/g of UO_2 showed that no metal-water reaction had taken place on the inner surfaces of the cladding. Figures 3 and 4 are photographs of the fuel rods following the 257 and 283 cal/g tests, respectively. Figure 5 is a sketch of the ruptured rod showing the location of the rupture and the fraction of the fuel remaining in the rod. Preliminary test results are presented in Table V.

The fuel from the rod that ruptured during the 283 cal/g of UO_2 test was fragmented into particles ranging from a slightly chipped pellet down to micron-sized pieces. The aggregate was screened and weighed to determine the particle size distribution. Photographs of the collections of particles, and the screen size, sieve opening, weight of collected particles, and the percent of the total weight represented by each group of particles are shown in Figure 6. These data show that a major part of the fuel residue was broken into relatively small particles. For example, 53.4 wt% of the fuel was in the range of 14 to 55 mils and 21.5 wt% was less than 14 mils in diameter.

The surface areas of the fragments were calculated from the results of the screening. The particle surface area for each collection and the total surface area are shown in Table VI. Calculations were made based on two methods of representing the particles constituting each collection. In both methods, the particles are assumed to be spherical. In method 1, the particles collected on a particular screen are represented by a single diameter, chosen as the mean diameter between the sieve opening of the collecting screen and the sieve opening of the next larger screen. In method 2, all particles constituting a particular collection are assumed to be of different diameters with the increments of diameter constant throughout the particular collection of particles. The diameters and surface areas for each collection were represented as follows:

d_o = sieve opening of collecting screen

d_n = sieve opening of next larger screen

D = increment of diameter

$d_1 = d_o + D; d_2 = d_o + 2 D; \dots d_i = d_o + iD; \dots$

Surface Area =

$$\frac{\pi}{4} \sum_{i=0}^n d_i^2 = \frac{\pi}{4} \sum_{i=0}^n (d_o + iD)^2.$$

Dividing the calculated surface areas by the volume of expelled fuel gives surface-to-volume ratios of $310 \text{ in.}^2/\text{in.}^3$ (method 1) and $242 \text{ in.}^2/\text{in.}^3$ (method 2). These values are about 20 to 25 times higher than corresponding values for the integral (intact) fuel. The surface-to-volume ratio of the expelled pellet fuel is significantly less than that of the powder fuel from previously tested waterlogged CDC powder-fuel rods [10,13]. The powder fuel had a surface-to-volume ratio of about $1500 \text{ in.}^2/\text{in.}^3$, based on the average fuel particle diameter

Fig. 3 Waterlogged F-type fuel rod following Test 217 (257 cal/g of UO_2).

Fig. 4 Waterlogged F-type fuel rod that ruptured during Test 218 (283 cal/g of UO_2).

Fig. 5 Location of fuel in rod that ruptured during Test 218 (283 cal/g of UO_2).

TABLE V

PRELIMINARY RESULTS OF TESTS ON WATERLOGGED
F-TYPE FUEL RODS

Test No.	Period (msec)	Maximum Energy (cal/g UO ₂)	Post-Test Observations
202	6.8	153	Rod was bowed slightly. There was one small discolored area ($\approx 1/2$ cm ²). At 12 inches from the bottom, cladding diameter increased 20 mils; at 13.5 inches, diameter increased 18 mils; at 22 inches, diameter increased 15 mils; at 26 inches, diameter increased 2 mils; there was no diameter change at 33 inches. No break in the cladding was detected.
205	4.9	181	Rod was bowed slightly. There were several brown spots of discoloration on the cladding in the region from 10 to 14 inches from the bottom of the rod. At 12 inches, cladding diameter increased 31 mils; at 13.5 inches, diameter increased 49 mils; at 22 inches, diameter increased 24 mils; at 26 inches, diameter increased 15 mils; at 33 inches, diameter increased 10 mils. There was a slight bulge in the region 13.5 inches from the bottom of the rod. No break in the cladding was detected.
206	4.2	223	No detectable bowing was observed. There were numerous dark spots of discoloration in the region from 10 to 16 inches from the bottom of the rod. At 12 inches, cladding diameter increased 29 mils; at 13.5 inches, diameter increased 24 mils; at 22 inches, diameter increased 25 mils; at 26 inches, diameter increased 19 mils; at 33 inches, diameter increased 10 mils. No break in the cladding was detected.
217	3.7	257	Rod was slightly bowed. The rod was considerably discolored in the region of highest flux. At 9.5 inches, cladding diameter increased 67 mils; at 10 inches, diameter increased 70 mils; at 12 inches, diameter increased 55 mils; at 13.5 inches, diameter increased 53 mils; at 22 inches, diameter increased 45 mils; at 33 inches, diameter increased 67 mils. No break in the cladding was detected.

TABLE V

(Contd.)

Test No.	Period (msec)	Maximum Energy (cal/g UO_2)	Post-Test Observations
218	3.2	283	Rod ruptured in form of a 2.5-inch slit at about 15.5 inches from the bottom of the rod. The rod was not discolored. At 9 inches, cladding diameter increased 69 mils; at 10 inches, diameter increased 70 mils; at 12 inches, diameter increased 62 mils; at 13.5 inches, diameter increased 56 mils; at 15.5 inches (bottom of rupture), diameter increased 74 mils; at 18.5 inches (top of rupture), diameter increased 79 mils; at 22 inches, diameter increased 57 mils; at 33 inches, diameter increased 21 mils. About 75% of the fuel was expelled from the rod.

of about 100 microns. This large heat transfer area for the powder fuel is believed to be the reason the powder-fuel rods failed at lower energy depositions and produced higher pressures than the pellet-fuel rods. Powder-fuel rods failed at about 40 cal/g of UO_2 and produced capsule pressures to about 10,000 psi, whereas the pellet-fuel rod failed at about 250 cal/g and produced a maximum capsule pressure of 1300 psi. A difference in cladding heat treatment could also have contributed to the failure characteristics. The pellet-fuel rods were clad with annealed Type 304 stainless steel; whereas the powder-fuel rods were clad with 35% cold-worked Type 304 stainless steel. The difference in cladding cold-work, however, is believed to be a secondary effect. Thus, the information obtained to date indicates that significant differences in the failure threshold and consequences exist between waterlogged pellet and powder fuels.

COLLECTING SCREEN (TYLER SCALE)	SIEVE OPENING (INCH)	WEIGHT OF FUEL COLLECTED (g)	PERCENT OF TOTAL WEIGHT
2.5	0.312	53.97	8.7
3.5	0.223	17.94	2.9
12	0.0555	82.66	13.4
42	0.0139	329.10	53.4
170	0.0035	119.30	19.4
PAN	--	12.93	2.1

Fig. 6 Particle distribution of fuel from rod that ruptured during Test 218 (283 cal/g of UO_2).

TABLE VI

PARTICLE SURFACE AREA CALCULATIONS -- TEST 218

Collecting Screen (Tyler Scale)	Volume (in. ³)	Method 1 Area (in. ²)	Method 2 Area (in. ²)
2.5	0.327	5.344	4.65
3.5	0.109	2.44	2.39
12	0.500	21.6	16
42	1.992	344.3	286
170	0.7221	498.2	411
Pan	0.081	284.5	182.2
TOTAL SURFACE AREA		1156.384 in. ²	902.24 in. ²
SURFACE-TO-VOLUME RATIO		310 in. ² /in. ³	242 in. ² /in. ³
<u>Initial (Pretest) Fuel Conditions</u>			
Fuel Area/Rod	=	64.08 in. ²	
75% Area	=	48.06 in. ²	
Fuel Volume/Rod	=	4.986 in. ³	
75% Volume	=	3.739 in. ³	
Surface-to-Volume Ratio	=	12.8 in. ² /in. ³	

3. FUEL PERFORMANCE TESTS ON DRY (NONWATERLOGGED) F-TYPE FUEL RODS

Tests were initiated this quarter on encapsulated, nonwaterlogged, F-type fuel rods to obtain scoping data on the behavior of these rods when subjected to repeated nondestructive tests at energies in the 200 to 250 cal/g of UO₂ range. Of primary interest were cumulative measurements of permanent axial cladding growth, rod bowing, and cladding deterioration. Tests were performed on each of three rods as discussed below.

The first rod was subjected to five transients, each having a period of ≈ 3.6 msec and corresponding test fuel energy of ≈ 250 cal/g of UO₂. Intermediate spacers were located at 13 and 26 inches from the bottom of the rod. Inspection after each test showed evidence of a slight bow after the first test. The bow progressively increased to about 3/4-inch off-center following the fifth transient. The maximum measured cladding surface temperature rise during

the first test was 1090°C and decreased slightly on each succeeding test to about 1030°C on the final test.

The second rod was subjected to four transients having periods of 3.8 msec and corresponding fuel energies of 225 cal/g of UO₂. Intermediate spacers were located 13 and 26 inches from the bottom of the rod. No bowing was detected following the first test; however, bowing occurred on succeeding tests, reaching a maximum of about 1/2-inch following the fourth test. The maximum measured cladding surface temperature rise was 1110°C which occurred on the first test. Surface temperatures were lower on succeeding tests, the lowest maximum of 790°C rise occurring on the third transient. The four tests resulted in a permanent axial growth of 750 mils.

The third rod was subjected to five transients having reactor periods of 4.3 msec and corresponding fuel energies of $\approx$ 200 cal/g of UO₂. Intermediate spacers were located 13 and 16 inches from the bottom of the rod. A slight bow was observed after the first test. The bow increased progressively throughout the series. The maximum measured cladding surface temperature rise was 861°C which occurred during the first test. Surface temperatures were lower on succeeding tests, the lowest maximum being 720°C. The five tests resulted in a permanent axial growth of 633 mils.

4. CONCLUSIONS

The results of tests on waterlogged F-type fuel rods subjected to single transients with energy densities from 153 to 283 cal/g of UO₂ revealed that diametrical cladding growth resulting from internal pressure generated during the transient increases with increased energy deposition. The maximum growth and discoloration occur in the region of the highest neutron flux.

The rod exposed to 283 cal/g of UO₂ ruptured, expelling about 75% of the fuel. The energy deposition at the time of rupture was $\approx$ 250 cal/g of UO₂. Measured capsule pressure was 1300 psi. The surface-to-volume ratio of the expelled pellet fuel was a factor of $\approx$ 5 less than that of powder fuel from previously tested CDC rods. The large heat transfer of powder fuel is believed to be the principal reason the powder-fuel rods failed at a lower energy deposition (40 cal/g of UO₂) and produced higher capsule pressures (10,000 psi) than did the pellet-fuel rods.

The scoping tests performed on dry F-type rods indicate that progressive axial clad growth and rod bowing are the primary consequences of sequential testing in the 200 to 250 cal/g of UO₂ energy-density range. There was no indication of cladding failure although the cladding did become badly discolored.

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