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SPERT PROJECT

April, May, June 1959

S. G. Forbes, Manager

Edited by J. C. Haire

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I. SUMMARY

SPERT I - The characteristics of the boiling process and its relation to moderator expulsion in Spert I have been investigated in a series of capsule type experiments. A fuel-bearing capsule, instrumented to provide pressure, volume, and temperature data during transient power excursions, was placed in a high flux region of the Spert I P core. Provision was made to control the boiling point in the capsule by varying the static pressure. The results of these experiments indicate that transient boiling plays a major part in moderator expulsion in the short period region thus substantiating the results of previous experiments.

Five step-induced transients initiated from boiling indicate that the kinetic behavior of the stainless steel clad P-18/19 core is dependent on initial temperature in a manner similar to that of previously tested aluminum clad Spert cores.

Reactivity oscillator techniques have been used in the P-18/19 core to determine the phase and magnitude of the reactivity-to-power transfer function from 0.01 to 18.4 cps at low power and at temperatures below boiling.

Criticality data on relatively simple lattices, both rod-free and containing a single poison rod, have been obtained from a series of clean critical experiments performed on a number of light water-moderated and -reflected slab configurations of Spert III fuel elements. The experiments were conducted in a specially instrumented tank located adjacent to the Spert I shield tank. Reactor control was accomplished remotely by variation of the water height in the experiment tank. The experiments provide a means of investigating the accuracy of the stainless steel constants and the control rod theory which led to discrepancies in previous calculations.

Changes in water height during the critical water height experiment were measured to ± 0.0013 inches by means of a simple remote-indicating system designed and built for this purpose. The system utilized a float to detect water level, transmitted the information through a synchro system, and provided direct readout from a synchro-driven dial indicator.

SPERT III - The operational loading has been determined to be 44 fuel assemblies and 8 control rods. Experiments have revealed that this loading provides about \$2.50 excess at 2500 psi and 650°C and a total rod worth above ambient critical of about \$20.50. Measurements have been made of the pressure and temperature coefficients of reactivity and the temperature defect.

Apparatus for non-nuclear engineering tests was installed and the tests were initiated to obtain data on core hydraulics, system energy balance and general plant equipment. Several difficulties were encountered with plant equipment during the hot critical experiments and subsequent engineering tests. These included primary system leaks, galling of core components and electrical component failures.

A primary coolant sampling station has been added.

Experimental apparatus for void experiments has been designed and fabrication started.

DATA-REDUCTION AND INTERPRETATION - A review of previous investigations of boiling and its relation to shutdown effects has been made in pursuance of a photographic investigation of shutdown mechanisms. A photographic technique was developed for conducting tests in a fuel-bearing, water-filled test cell in Spert I. Results of the investigation, which included high-speed motion picture of boiling and measurements of fuel plate surface temperatures within the cell, indicate nucleate rather than film boiling occurs and that water boiling plays a significant role in initial reactor self-shutdown in the short period region.

A machine program using computed reactivity as input data has been used to obtain power solutions to the kinetics equations. Comparison of the computed and experimental power burst shapes has shown the reactivity data and power data to be consistent and in this way has essentially verified the independent program for reactivity calculations. Previous studies have revealed a dip in reactivity following the transient power burst. The present comparison has established that this dip could be a consequence of small errors in experimentally observed power burst shapes which are used as input data.

ENGINEERING - Corrosion and galling tendencies have been investigated in materials of construction for threaded components used in high temperature demineralized water service.

The Spert II primary piping system is complete. Calibrations, check-out, and final tests are in preparation. Work continues on control system wiring, wiring schedules and schematic drawings. The initial core has been accepted.

II. SPERT I

A. The Mechanism of Moderator Expulsion in Spert I Shutdown (S. G. Forbes and P. French)

Experiments have been performed to investigate the characteristics of the boiling process in relation to moderator expulsion in the Spert I reactor. Previously reported preliminary experiments⁽¹⁾ indicated that radiolytic gas is not a significant contributor to reactor self-shutdown for periods shorter than 2 sec. They also indicated that in the presence of fuel the expected moderator displacement occurs in a manner consistent with the rapid establishment of boiling. From these experiments it was concluded that a heat source is necessary to achieve substantial moderator displacement, with the most likely expulsion mechanism being transient boiling.

The preliminary experiments demonstrated the need for accurate measurements of the contribution of boiling effects to shutdown and necessitated modification of the experimental apparatus for the series of tests reported here.

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Fig. 1 - Capsule for Measurements of Pressure, Volume and Temperature During Transient Expulsion of Moderator from a Simulated Core Channel

For these tests an aluminum-clad fuel plate was rolled into a cylinder approximately 1/2 in. in diameter and 24 in. long and placed in the capsule shown in Fig. 1. The capsule had provisions for measuring dynamic pressures and fuel plate surface temperatures at five stations along the length of the fuel, and also had volume transducers of small effective mass on each end.

The use of two volume transducers made the capsule effectively "open" at both ends, simulating the hydrodynamic conditions of the reactor fuel assemblies. In these experiments, provision was also made to control the boiling point in the capsule by controlling the static pressure. The capsule was placed in a high flux region of the stainless steel clad P core and, as was expected, the temperatures observed in the capsule were somewhat higher than elsewhere in the core.

The first group of experiments performed with this capsule consisted of three step transients, each having an initial period of about 30 msec. These experiments showed the onset of pressure to coincide closely with the attainment of boiling temperatures with no apparent delay. Fig. 2 shows the behavior of the observed temperature and pressure at the fuel cylinder centerline. Static pressures of 0, 10, and 100 psig were applied, respectively, in the three tests. The boiling points for the 0 and 10 psig cases are indicated on the temperature trace, which is representative for all three transients. The boiling point for the 100 psig case was 170°C and is not shown since it was not attained during the transient. The observed decrease in dynamic pressures as the boiling point is raised confirms the earlier indication that boiling provided propulsive pressures for moderator expulsion.

A second group of shorter period transient tests was performed with the same apparatus and applied static pressures of 0, 50, 100, and 200 psig. These tests had initial periods ranging from 8 to 10 msec. Representative behavior of temperature and dynamic pressure is shown in Figs. 3, 4, and 5. In each case the boiling temperature and saturation pressure corresponding to the fuel temperature are shown for comparison.

In the unpressurized case, shown in Fig. 3, the dynamic pressure may be seen to follow the saturation line quite closely up to a few psi and then, to depart sharply from the saturation line and remain considerably below it. The erratic behavior shown before the pressure drops to zero is peculiar to this unpressurized run, and was not seen with higher static pressures.

Figs. 4 and 5 show the same information for applied static pressures of 100 and 200 psig, respectively, and are typical of the pressurized experiments. The pressure rises smoothly from the time boiling temperature is attained, always being considerably below the saturation pressure.

The calculated volume changes which would result if the pressure were effective in expelling moderator agree with the observed volume changes within a factor of two for all conditions of initial static pressure. The comparison for the 100 psig static pressure run is shown in Fig. 6.

Fig. 2 - Dynamic Pressure Variation With Static Pressure for 30 msec Step Transients

Fig. 3 - Capsule Pressure and Temperature vs Time from Peak Power ($\tau = 8.6$ msec)

Fig. 4 - Capsule Pressure and Temperature vs Time from Peak Power ($\tau = 10$ msec)

Fig. 5 - Capsule Pressure and Temperature vs Time from Peak Power ($\tau = 10$ msec)

Fig. 6 - Moderator Expulsion vs
Time from Peak Power for
10 msec Step Transients

Fig. 6 also shows the effect of pressurization on moderator expulsion. The lower, slowly varying parts of the curves represent the results of thermal expansion. Boiling points are indicated by the closed circles on the curves. In all cases, rapid moderator expulsion begins within a few milliseconds after boiling temperature is attained.

The extreme abruptness of this boiling shutdown effect should cause this type of shutdown to be relatively insensitive to the reactor void coefficient. This observation is in agreement with the empirical observation that non-linear shutdown coefficients were required in the "empirical model" to produce agreement with experimental observations.

In summary, the earlier experiments would appear to definitely eliminate radiolytic gas production as a significant contributor to self-shutdown in Spert-type reactors, at least for initial periods shorter than 2 seconds.

The experiments reported here show that transient boiling plays the major part in moderator expulsion in the short period region. Important characteristics which have been found include the coincidence of pressure rise with the attainment of boiling temperatures, and the consequent rapid moderator expulsion following within a few milliseconds. The nature of the boiling is not yet known; that is, whether it is of a nucleate or film nature. The results of initial experiments to determine the nature of boiling are presented on page 27.

B. Boiling Transients, P-18/19 Core (R. W. Garner)

A series of five, two-foot head, step transient tests with initial reactor temperature near boiling ($\sim 95^{\circ}\text{C}$) has been performed on the stainless steel clad P-18/19 core.

It is concluded from the data, presented in Table 1, that the transient behavior of the stainless steel clad core at boiling differs from that at ambient in the same manner as previously tested aluminum clad Spert cores.^(2,3)

TABLE 1
BOILING TRANSIENT DATA P-18/19 CORE

CEC Record Number	Period (msec)	Peak Power* (Mw)	Temp. at Peak Power ($^{\circ}\text{C}$)	Maximum Temp. ($^{\circ}\text{C}$)	Maximum Pressure (psig)
10223	76	2	101	102	0
10230	23	15	107	113	0
10233	11	89	116	126	6
10235	5	390	174	233	21
10241	4	560	190	266	27

* A +15% increase in neutron chamber sensitivity in going from ambient to boiling, approximated by other cores, has been used in obtaining these data, since the neutron chambers were not otherwise calibrated for boiling reactor temperatures.

C. Reactivity Oscillator Program (A. A. Wasserman)

Reactivity oscillator measurements are being conducted in the P-18/19 core in Spert I using the apparatus described previously.⁽⁴⁾ The phase and magnitude of the reactivity-to-power transfer function from 0.01 - 18.4 cps have been determined at 500 - 1000 watts and at ambient temperature using reactivity amplitudes from about 2.3% to 11%. Measurements have been made also at 75 - 100 kw and about 50°C , using auxiliary cooling of the reactor water. No feedback effects were observed for this condition. The data are still being analyzed. Experiments are in progress at 500 kw and about 70°C . During runs at 75 kw and above, various fuel plate temperatures and strains, water flow-rates and temperatures, and pressures are being monitored.

D. Critical Water Height Experiments with Spert III Stainless Steel Assemblies in Slab Geometry (A. H. Spano)

1. Introduction

Clean critical experiments have been conducted on a number of light water-moderated and -reflected slab configurations of Spert III type 1S fuel elements. The objective of these experiments has been to obtain criticality data on relatively simple lattices which either are rod-free or contain a single poison rod. These data may be used to verify separately the reactor constants and the control rod analysis used in a previous four-group multi-region calculation for the operational Spert III core.^(5,6)

In the earlier analysis of the stainless steel core, carried out in cooperation with the Central Physics and Mathematics group at Bettis, flux-weighted, two-dimensional radial diffusion solutions were combined with a one-dimensional axial solution for the eigenvalue and flux distributions over the core. The results of these extensive calculations, performed for a variety of reactor conditions, showed the calculated k_{eff} to be generally about 3% greater than the corresponding experimental value. However, application of similar analytical methods to Zr-H₂O systems yield eigenvalue results which are in the neighborhood of 1% of criticality. This discrepancy in calculational results between stainless steel and zirconium reactor systems suggests the use of either incorrect stainless steel constants or an inadequate control rod theory. These possibilities were investigated by means of the unrodded and single-rodded experiments. Analysis of the experimental results is in progress.

Simplification in the experimental design allowed for flexibility and ease of assembly of core configurations. For clearness in the interpretation of data, reactor control was provided entirely by variation of the water height. No control rods were used for control purposes. Thus, rod-free cores were uniformly packed arrays of fuel elements only, while rodded cores contained a single, fully inserted poison rod surrounded by fuel. All experiments were conducted in an auxiliary vessel situated adjacent to the Spert I reactor shield tank, with reactor control exercised remotely one-half mile from the experimental site.

2. Experimental Design

a. Tank and Control System (T. Quigley). Cores were assembled in a 4 x 6 x 3.5 ft high, open rectangular tank, which was located next to the Spert I shield tank for water dumping purposes. Control of critical assemblies was effected by variation of the water height, using a remotely controlled system for filling and draining.

The water inlet, supplied by a 15 gpm positive displacement pump, provided two fill rates by means of an electric-pneumatic valve placed in parallel with a small open by-pass line. With the pump on and pneumatic valve open, a fill rate of approximately 1 in. of water height per minute was obtained. With the valve closed, the fill rate was reduced by a factor of 7. Use of the slow fill rate provided more precise control of the water level in the region of the critical water height.

The drain system consisted of a slow and fast drain, and a scram water dump. For the slow drain, a 1.5 in. flexible hose was connected to the bottom of the tank and attached to a Spert I control rod drive to form a movable U-shaped overflow tube. The 20 in. maximum travel of the rod drive limited the usable range of the overflow tube from the top of the tank (approximately the top of the core) down to a level corresponding to 16 in. above the bottom of the core. This overflow tube had the safety feature of effectively preventing the water level in the tank from rising above a given height. The fast drain consisted of a 2-in. pipe leading from the bottom of the experiment tank to two 1.5-in. solenoid valves located at the bottom of the Spert I shield tank. For reactor safety, a fast acting, positive water dump was provided by a 3-in. flexible hose suspended from its center by an electromagnet to form an inverted U-shaped tube, one end of which was fitted to an outlet near the bottom of the tank and the other (weighted) end pointed downward into the shield tank. This latter arrangement yielded a maximum dumping rate in excess of 1 in. of water height per second. Finally, an additional emergency reactor shutdown mechanism consisted of a 3-gal capacity, boron solution injection system.

b. Instrumentation (R. W. Garner and J. Siegwarth). Seven instrumentation channels were used to transmit information on neutron levels, water height and water temperature.

For neutron data, two boron-lined pulse counters were used primarily for subcritical multiplication measurements and two boron-lined, gamma-compensated ion chambers provided linear and log power traces for criticality and reactor period determinations. Neutron detectors were placed in watertight metal tubes, which were positioned 6 in. or more from the sides of the core in order to minimize perturbations by voids in the radial reflector region.

The water level was monitored in two ways. For operational purposes, a rough visual indication having a reading accuracy of about 1/8 in. was provided by a television camera viewing a yardstick mounted on the wall of the experimental tank. For experimental measurements, a simple serviceable water level indicator was constructed utilizing a water float system; this is described in detail on page 16. The differential water height accuracy obtained with this instrument was better than 0.01 in.

Water temperature was measured by a four-junction thermopile located in the reflector. All critical experiments were performed in the temperature range between 55 and 60°F, with the temperature variation during the course of any experiment limited to less than 1°F. A small stirrer was operated intermittently to insure uniform bulk water temperature throughout the core.

c. Description of Fuel Assembly and Control Rod (A. H. Spano). Cores were built up with Spert III type 1S fuel elements spaced on 3-in. centers. This was identical to the Spert III lattice spacing. Positioning of the elements at the bottom was provided by a reinforced 19 x 3/4 x 1/2-in. thick aluminum-grid plate having a 5 x 10 array of holes for the fuel element end-boxes. The grid supported the fuel elements approximately 1/2-in. above the floor of the tank. At the top of the core, aluminum angles, rigidly clamped to the sides of the tank and to each other, formed a rectangular

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Fig. 7 - Spert III, Type 1S
Fuel Element

59-4415

Fig. 8 - Spert III Control Rod
and Guide Tube

TABLE 2

TYPE 1S FUEL ELEMENT PARAMETERS

Nominal Active Length of Fuel Plate	36 in.
Full Length of Fuel Plate	37 in.
Nominal Active Width of Fuel Plate	1.045 in.
Full Width of Fuel Plate	1.445 in.
Meat Thickness	0.020 in.
Plate Thickness	0.030 in.
SS Clad Thickness	0.005 in.
Water Channel Thickness	0.128 in.
Number of Fuel Plates	38
Overall Width of Element	2.950 in.
Overall Thickness of Element	2.950 in.
Uranium Enrichment	93.2% U-235
Meat Composition:	
U-235	15.4 Wt. %
UO ₂	18.8 Wt. %
Type 304 SS	81.2 Wt. %
Total U-235 per Fuel Element	638 g
Cell Size	3 x 3 in.
Metal-to-Water Ratio per Cell	0.298

perimeter of appropriate slab dimensions surrounding the elements. To maintain the correct internal spacing, aluminum shims were placed between elements near the top end-boxes.

The type 1S fuel element consists of 38 fuel plates, side plates, and a center reinforcing plate. Some construction details of the fuel element are illustrated in Fig. 7, and Table 2 presents the pertinent fuel element parameters.

Construction details of the control rod are shown in Fig. 8. The stationary poison rod used in the experiments is an exact duplicate of the Spert III control rod poison section and consists of a water-filled, box-shaped cylinder of enriched (93.2% B¹⁰) boron alloyed with stainless steel. This is surrounded by a slightly larger, box-shaped stainless rod guide tube.

3. Experimental Procedure (H. Whitener)

In accordance with normal Spert procedure, reactor control was accomplished remotely. In addition, for these water height experiments personnel were excluded from the reactor site whenever water was present in the tank. The approach to critical for each core configuration was made by raising the water level in initial increments of 5 in. With a neutron source present in the slab the neutron multiplication was measured following each addition of water and a prediction of the critical level obtained from a plot of the inverse multiplication vs water height. With increasing multiplication the water level increments were decreased appropriately. Near criticality the source was removed and the water level slowly adjusted to the exact critical height.

Following the determination of the critical height, h_c , two measurements of the incremental water-height reactivity worth, $d\rho/dh$, were made. Incremental changes, Δh , in water height varied from about 0.2 to 0.6 in., yielding reactor periods from several hundred seconds down to about 50 sec. Values for $d\rho/dh$ were then computed and associated with mean water heights, given by $\bar{h} = h_c + \Delta h/2$.

4. Experimental Results (A. H. Spano)

Fuel elements were arranged in essentially slab arrays of 2, 3, 4, or 5 rows, either rod-free or with a single, fully inserted control rod. Orientation of the elements was such that the fuel plates were parallel to the row axis, that is, the long dimension of the slab, except for a single comparison case in which the plates were oriented normal to the long dimension.

Two-Row Configuration: A 2-row, 9-column (2 x 9) slab was found to be subcritical with the water height at the top of the core, at 36 in. Further, no definite increase in neutron multiplication was observed for this long slab as the water height was raised to the top of the core, indicating that the infinitely long 2-row slab is probably subcritical.

Three-Row Configuration: In order to minimize the error in the determination of the axial buckling, several three-row cases were studied to obtain a critical water height near the top of the core. The results

for the rod-free and rodded slabs are shown in Table 3. In column 2 of this table, core configurations are indicated by the array of circles for elements and crosses for control rods.

TABLE 3

THREE-ROW SLAB RESULTS

Slab Type	Loading Pattern	Critical Level* (h_c , in.)	Incremental Water Worth (cents per inch at $\bar{h}$.)
rod-free	<pre> ooooo oooo oooo </pre>	Subcritical	
rod-free	<pre> oooo ooooo oooo </pre>	34.04	15.5 at 34.30 in., 15.0 at 34.46 in.
rod-free	<pre> ooooo ooooo oooo </pre>	27.75	29.3 at 27.90 in., 28.7 at 28.03 in.
rod-free	<pre> ooooo ooooo ooooo </pre>	23.49	47.2 at 23.61 in.
rod-free**	<pre> ooooo ooooo ooooo </pre>	24.55	39.0 at 24.72 in.
rod-free***	<pre> oooo oooo oooo oooo oooo oooo </pre>	28.33	28.9 at 28.51 in., 28.3 at 28.65 in.
rodded	<pre> oooooooo oooooxoooo ooooo </pre>	Subcritical	
rodded	<pre> oooooooo oooooxoo ooooo </pre>	31.41	20.9 at 31.62 in., 20.6 at 31.86 in.
rodded	<pre> oooooooo uuuuuuuu ooooo </pre>	28.12	29.3 at 28.24 in., 29.2 at 28.41 in.
rodded	<pre> oooooooo oooooxoooo ooooo </pre>	23.64	45.1 at 23.76 in.
rodded	<pre> oooooooo oooooxoo ooooo </pre>	31.26	21.9 at 31.52 in., 21.6 at 31.72 in.

* Water height above nominal bottom of active core.

** Slab case in which fuel plates were oriented normal to the row axis.

*** Rod-free split core case of two 3 x 4 slabs separated by a 3-in. water gap.

Four-Row Configuration: A 19 fuel element, 4 x 5 rodded array with the control rod located at (3,3) was critical at a water height of 34.26 in. Measurements of incremental water worth yielded values of 14.4 cents per in. at $\bar{h} = 34.68$ in., and 13.0 cents per in. at $\bar{h} = 34.98$ in.

Five-Row Configuration: A cylindrical configuration was approximated by 20 fuel elements in a 5 x 5 rodded array with the corners missing and the control rod in the center. Criticality was obtained at a water height of 29.37 in., and the incremental worth was measured to be 28.9 cents per in. at 29.58 in., and 27.2 cents per in. at 29.72 in. Bare and cadmium-covered gold foils were activated in the core and reflector regions to obtain radial and vertical flux distributions. The gold foils were 0.156 in. in diameter and 0.005 in. thick. Under these conditions, the critical water level increased slightly to 29.56 in., while the incremental worth decreased to 25.9 and 24.3 cents per in. at the respective mean water heights of 29.69 and 29.77 in.

In Fig. 9 a quarter section of the cylindrical core is illustrated with the positions indicated for the flux traverses. The horizontal flux distribution is shown in Fig. 10. The small peaks in the radial thermal flux distributions are seen to occur at the fuel-free regions of the center reinforcing plates and in the water channels between assemblies. Fig. 11 shows the vertical flux distribution.

The incremental water height worth for rodded and rod-free three-row slabs has been plotted as a function of water height and is shown in Fig. 12.

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Fig. 9 - Quarter Section Through 5-Row Core Showing Location of Horizontal and Vertical Flux Traverses

Fig. 10 - Horizontal Flux Distribution Through the 5-Row Core

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Fig. 11 - Vertical Flux Distribution Through the 5-Row Core

59-4420

Fig. 12 - Reactivity Worth of Water Height vs Water Height for the 3-Row Slab Cores

The points fit a fairly smooth curve, which in the neighborhood of the top of the core might be expected to drop off sharply. The integral of this curve is shown as the dashed curve. The excess reactivity for each slab case is obtained from the integral curve.

Fig. 13 shows a plot of the water height as a function of $(d\rho/dh)^{-1/3}$. Except for the two points near the top of the core, where changes in reflector savings occur due to the proximity of the end-box reflector region, a linear fit is indicated. Thus, for water heights in the range from 23 to 33 in., application may be made of the linear relationship

$$h_c = (\pi^2 M^2)^{1/3} (d\rho/dh)^{-1/3} - S, \quad (1)$$

where S is the total axial reflector savings and M^2 , for two-group theory, is given simply by

$$M^2 = \frac{\tau}{1 + \tau B^2} + \frac{L^2}{1 + L^2 B^2} \quad (2)$$

In Eq. (2) a small correction term, which takes into account the difference in ages between prompt and delayed neutrons, has been neglected. A least squares analysis of the data yields a value of 3.7 in. for the reflector savings, and from the slope in Eq. (1), a value of $477 \beta_{\text{eff}}$ in.² for M^2 . Further analysis of these results is presently underway.

Fig. 13 - Water Height vs Inverse
Cube Root of $d\rho/dh$

5. Precision Water Level Monitor (K. E. Krauter)

A simple remote-indicating system for measuring changes in water height was designed and constructed for use in the critical water height experiments. The system, shown pictorially in Fig. 14, consisted basically of a water float unit, a synchro-generator and a synchro-driven dial indicator.

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Fig. 14 - Illustration of Water Level Monitor System

The water height sensing unit was a float made from a weighted film can 12 in. in diameter. Vertical motion of the float was transmitted to the synchro-generator by means of a 0.008 in. diameter, stranded stainless steel wire attached to the float and passing over a pulley mounted on the synchro-generator shaft. A counterweight was attached to the free end of the wire and, together with the weighted film can, provided tension to eliminate wire slippage over the pulley, improve float stability and reduce non-vertical motion of the float. The effect of water turbulence in the reactor tank on the water level measurement was minimized by placing the float in a perforated 2-1/2 ft diameter can. The 0.9510 in. diameter precision pulley mounted on the synchro shaft provided 1.2 degrees rotation of the synchro for a change in water height of 0.010 in. To reduce pulley-eccentricity errors, final machining of the pulley was made after it was fixed to the synchro-generator shaft. Two type-IF, 110 volt, 60 cps Bendix synchros were used to relay the water level information 3000 ft from the reactor to the control center. Readout of this information, which was reproducible to ± 0.002 in., was obtained from a dial attached to the synchro motor shaft. A fine wire was wrapped around the synchro shaft and a weighted pointer was attached to the free end to indicate the number of turns made by the synchro and to reduce backlash effects.

A fixed reference level for the float in the 2-1/2 ft diameter can was given by the measured distance from the top surface of the reactor grid plate to the lowest flow-restricting hole in the can. With the water dumped from the reactor tank, the water in the can was drained until the water level reached the lowest hole. Care was taken to avoid disturbance of the water surface in the can prior to taking a reference water level reading.

Various measured and calculated errors associated with the water level indicator system are presented in Table 4. The measured errors were determined using a 36 in. machinist scale and a 5 in. micrometer-caliper having a least count of 0.001 in.

TABLE 4

WATER HEIGHT ERRORS

Measured Errors

System Calibration ($1.2^{\circ} = 0.01$ in.)	+ 0.0013 in./in. *
Repeatability	+ 0.002 in.
Hysteresis	< 0.001 in.
Synchro	< 0.0004 in. **

Calculated Errors

Change in Pulley Diameter	0.003 in./0.001 in. *
Misalignment of Pulley and Shaft	0.003 in./0.001 in. **
Temperature Expansion of Pulley	0.0001 in./ $^{\circ}\text{C}$ *
Temperature Expansion of Cable and Synchro Support	0.0001 in./ $^{\circ}\text{C}$ *
Change in Cable Diameter	0.0005 in./0.001 in. *
Change in Float Weight	0.045 in./oz

* Accumulative error per inch of water height

** Repeated error per revolution of synchro

The system calibration error noted means that the $1.2^{\circ} = 0.01$ in. relation is inexact and should be $120^{\circ} = 1.0013$ in. Repeatability error gives the imprecision in readings from successive measurements of the same visual point of references. Hysteresis error is the upper limit to error of water height measurement arising from imperfect response to synchro-dial indicator to a change in the synchro-generator. Synchro error is the maximum amplitude of deviation of the actual synchro response curve from a linear response curve.

The system proved to be practicable and provided sufficiently precise and accurate data for the critical water height experiments.

III. SPERT III

A. Spert III Reactor Experiments (C. R. Toole)

High temperature studies were initiated on March 30 to determine the operational loading and various core parameters for Spert III. (The operational loading is defined as one which has about 2% excess reactivity available at 650°F and 2500 psi and is subcritical by 5 to 10 per cent at ambient temperature with the rods inserted.) During the first quarter of 1959, the fuel loading had been increased from the minimum critical loading of the C-19/27 core* to the estimated operational loading, C-19/48 core(7), shown in Fig. 15. Tests on the C-19/48 core were concluded after finding that the maximum temperature for which criticality could be achieved was about 640°F. All tests on the C-19/48 core were conducted with the system at 2500 psi.

Since the operational loading requirements were not fulfilled by the C-19/48 core, the loading was increased to a C-19/52 core, also shown in Fig. 15, and an additional series of hot critical tests was undertaken. Criticality was achieved at 30.25 in. with the reactor temperature at 655°F and the pressure at 2500 psi. Interpolation of the critical position versus temperature data predicts that the critical position at 650°F would be 27.5 in. Incremental rod worth measurements indicate an available excess reactivity at 650°F of about \$2.50. Fig. 16 is a plot of the rod worth at critical, from period measurements, versus critical position. Integration of this curve gives a total rod worth above ambient critical of about \$20.50. The C-19/52 core thus satisfies the excess reactivity requirements as the operational loading for Spert III.

From measurements of the critical position at various pressures, the pressure coefficient of reactivity was determined to be about +0.014¢/psi at 80°F and about +0.019¢/psi at 420°F.

The temperature coefficient of reactivity was obtained as a function of temperature by the following procedure. With the system temperature held as near constant as possible, the ganged-rod critical position was determined. By raising the control rods slightly, enough reactivity was introduced into the system to put the reactor power on about a 60 sec period. When the power had increased sufficiently to yield a good period measurement, one control rod was inserted until the reactor was slightly

* The notation L-N₁/N₂ core is consistent with that used for all cores tested in Spert reactors. The letter L denotes the type of fuel assembly used; the number N₁ denotes the number of full-width fuel plates per assembly; and N₂ the number of fuel assemblies in the core. In Spert III cores, N₂ is the number of core lattice positions occupied by both fuel assemblies and fuel-poison control rods. Hence C-19/27 designates a loading of C type fuel assemblies having 19 full-width plates (actually 38 half-width plates) per assembly, and a total of 19 fuel elements plus 8 control rods in the reactor, i.e., 27 core positions occupied.

Fig. 15 - Spert III Lattice

Fig. 16 - Control Rod Worth at Critical vs Critical Position, Spert III

subcritical. At this time three rods were level and the selected rod was an inch or two lower. The heat exchanger valves were then adjusted to allow the system temperature to increase at a slow, but relatively constant, rate. While the temperature was rising, the selected control rod was adjusted to keep the reactor close to critical. When the excess reactivity which had been added previously was almost compensated by the rise in temperature, the selected rod was raised to its initial supercritical position, i.e., level with the other control rods. The power was then allowed to rise slowly until it reached a maximum and began to decrease. At the time of maximum power the reactor was assumed to be critical. By correlating the power data with the temperature readings taken throughout the procedure, the temperature difference between the two critical measurements was determined. The previously mentioned period measurement was used to determine the reactivity compensated by the temperature increase and, hence, the temperature coefficient. The rod worth as a function of critical position was also derived from the period measurement. If the temperature varied between the time of the critical measurement and the time of the period measurement, a first-order correction was applied, using an approximate temperature coefficient. The approximate temperature coefficient was estimated by ignoring any temperature change between the initial critical determination and the period measurement.

The temperature coefficient for the C-19/52 core has been determined for various temperatures at 2500 psi and, over a limited range, at 1000 psi. The data are presented in Fig. 17 and show that a reduction in pressure from 2500 to 1000 psi is accompanied by a 25% increase in the temperature coefficient. At 650°F and 2500 psi the temperature coefficient is about $-16\%/^{\circ}\text{F}$. In Fig. 18 temperature defect at various temperatures is shown. The total defect from 80° to 650°F is about \$18.00.

Fig. 17 - Temperature Coefficient
vs Temperature, Spert III

Fig. 18 - Temperature Defect
vs Temperature, Spert III

Completion of this series of tests concluded the nuclear operation of Spert III for this quarter. The final two months of the quarter were devoted to the preparation for, and initiation of, engineering studies of the Spert III plant.

B. Engineering Tests (J. A. Norberg)

1. Introduction

Non-nuclear engineering tests were initiated on the Spert III reactor system following the hot critical experiments. These tests have been designed to obtain information on the hydraulic and thermal characteristics of the system for use in the nuclear experimental program.

Of primary importance are data on core hydraulics and the system energy balance. Other engineering information, such as coolant flow versus valve position, heat exchanger pressure drop versus flow and system expansion, is also being obtained.

Since the Spert III experimental program will consist of investigations over a wide range of pressure, temperature and flow conditions, an independent investigation of each parameter is being made during the engineering tests to determine its effect on system behavior. This is being accomplished by a series of flow tests conducted at selected temperatures during each of three constant pressure runs at 650, 1250 and 2500 psig.

2. Core Hydraulics

Pitot tubes, static pressure tubes, and a turbine-type Potter flowmeter have been placed in the core to obtain information on coolant distribution. Pitot tubes are also installed in the water annuli between the thermal shields. Location of the instrumentation is indicated in Fig. 19.

It should be noted that lattice positions not occupied by the C-19/52 core were loaded with orificed filler pieces, as was the core periphery region.

3. Energy Balance

Knowledge of the energy relationships of the process system is desirable in the calibration of nuclear power instrumentation. The energy balance of the system is being determined by three methods: (1) steady-state heat balance, (2) system heating rate, and (3) system cooling rate.

4. General Plant Data

Information on the operation of general plant equipment is being obtained. These data include the pressure drop versus flow

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Fig. 19 - Spert III Lattice Showing Instrumentation Locations for Engineering Tests

characteristics of the heat exchangers and reactor vessel. Flow control valve position versus flow is being recorded and the system primary loop expansion is being measured.

5. Operating Procedure

The system is pressurized to the desired pressure of the run, i.e., 650, 1250, or 2500 psi, and the temperature stabilized at an equilibrium condition of 250, 350, 450, 550, 600, or 650°F. (Maximum temperature for each run is determined by the system pressure.) The coolant flow rate is adjusted to several different values and data relating pressure drop, flow distribution, steady-state heat balance, valve positions, and system expansion are obtained for each flow condition. Heating rate data are obtained as the system is heated to the next higher temperature point. Cooling rates are being recorded at each of the above noted temperature points during the system cooling cycle.

6. Results

Plant operational difficulties, discussed below, have caused considerable delay in the test program; consequently, data have been obtained on only the 650 and 1250 psi runs. Tests are continuing at 2500 psi and analysis of the data is in progress.

C. Operational Problems

1. Mechanical (S. W. Gleave)

Several plant equipment difficulties have been encountered during the hot critical experiments and engineering tests. The following summary outlines the major problems that occurred.

a. Primary System Leakage. Several leaks of varying degree developed in the Spert III primary coolant system. The channel covers on the East and West loop heat exchangers continued to leak intermittently with an apparent strong-dependence on the mode of the exchanger operation. The trouble appears to be due to unequal temperature gradients across the channel cover, causing expansion and/or contraction in the gasket area which results in leakage. In one instance during plant operational conditions of 2500 psig and 600°F, the channel cover of the West heat exchanger was hot-butted to stop a leak. The corrective action was short-lived, however, since leakage reoccurred at a temperature of 645°F.

The 16-in. by-pass valve in the East heat exchanger developed a major leak through the packing assembly, resulting in a plant shutdown. The leak occurred while opening the valve with plant conditions at 1250 psig and 550°F. The exact cause of the failure has not been determined; however, examination of the valve revealed several significant factors. The valve stem was scored, the gland follower was bent, the gland eye-bolts had failed under tensile stress, and the bonnet bushing, lantern ring and gland were frozen to the valve stem. Apparently, one or more of the above-noted parts bound-up on the valve stem as the valve was being opened, and

the force from the motor operator plus the system pressure was great enough to break the gland eye-bolts and allow the packing assembly to follow the valve stem out of the valve body, resulting in the leakage.

The 4 in. valve in the top pressurizer-reactor connecting line again failed to function properly.⁽⁸⁾ The resulting flow of steam from the pressurizer steam dome into the reactor vessel necessitated plant shutdown. The valve was to be replaced in the line by a plugged section of 4 in. pipe. However, when the valve was cut free, stresses in the piping were relieved and the two ends of the pipe moved out of alignment by about 3 in. Consequently, the ends of the pipe were separately capped and no attempt was made to realign the pipe.

Other leakage difficulties, attributed to installation rather than components, were encountered with fittings. Two 1/8-in. Conax fittings bringing instrument leads from the reactor vessel failed through their seal gland and were repaired by replacement of the gland. A 1/2-in. compression fitting parted and required reinstallation. Several seal welds on various fittings and valves developed minor leaks requiring field repair.

b. Miscellaneous Problems. Galling between stainless steel parts was again encountered during fuel assembly changes. The lower end-boxes of five dummy fuel assemblies were modified and placed in the reactor during the non-nuclear engineering tests. Evaluation of the effect of the modification will be made at the completion of the tests.

During single-loop operation of the primary system, a flow dependent rhythmic hammering was observed. The phenomenon, apparently due to slamming of the check valves in the idle loop, was remedied by closing the control valve in the idle loop.

Considerable delay in nuclear operations resulted from slow pressure buildup of the control rod air pressure. Inspection of the control rod air compressor revealed sand-like material in the cylinders and valves. Two cartridge-type filters were therefore installed in the compressor suction line.

A compressor on the 5-ton air conditioner for the transient instrumentation room failed after four hours of operation, requiring installation of a new compressor unit by the manufacturer.

2. Electrical Equipment (D. T. Jones)

a. Process System Power. When operation of the Spert III process system at high pressure was attempted the flow control valves failed to function properly. Investigation revealed two difficulties. The flow controllers were hunting, thereby causing frequent starting of some of the control valve motors and consequent overheating. This resulted in tripping the overheat relays in the motor starters, rendering these valves inoperative. Properly rated circuit breakers were installed in the branch circuits serving other valve motors. The overheat relays in the motor starters were of a type which can be made to reset manually

or automatically, as desired. Accordingly, they have all been changed from manual reset to automatic reset so that should future overheating occur, it will not completely incapacitate the system even in the absence of operating personnel.

b. 480 Volt Wiring Problems. Design of the control rod drive system for Spert III was not frozen until construction of the facility was well advanced, so that installation of electrical cables to the control rods had to conform to existing structure. It had been intended to run neoprene-jacketed, 19-conductor cable from the relay racks to a junction box at the reactor, and then to use special heat and moisture resistant wiring from the junction box to the drives. However, the completed control system design included nearly 400 wires carrying 120 and 480 volt, 60 cycle power to the drives. This would have required 21 neoprene cables and a rather large junction box, which would have created a space problem. The problem was obviated by making up less bulky cables using hookup wire and heavy plastic tubing and by running the cables directly from the relay racks to 100-pin connectors in the drives, thus eliminating the junction box entirely. A supporting structure was provided for the cables at the drives. The structure folds into the floor grating, thereby providing storage for the loose cables when they are disconnected for removal of the reactor vessel head.

Specifications written for the 16-gauge hookup wire used in these cables, in a duplicate set for the dry dock, and in internal connections in the relay rack were in conformance with those previously used in obtaining an 18 gauge wire which has performed satisfactorily in similar circuits. The 16 gauge wire was not obtained from the manufacturer of the 18 gauge wire and although it met specifications it did have a thinner lacquer coating on the cellulose acetate yarn insulation. The breakdown voltage of the dry insulation was measured to be 2000 volts to ground or 4000 volts wire to wire.

The wire was installed and gave satisfactory service until April 14, 1959. On that date, during operation of the reactor at high pressure and temperature, two drive motor circuit wires in the relay rack developed simultaneous wire-to-panel faults, creating a 480 volt short-circuit, tripping the feeder breaker and scrambling the control rods. The system automatically switched to standby power and initiated fast-run-down of the rod drives. The short circuit burned wires and lowered voltages causing chattering of contactors carrying the extreme load, so that many contacts were destroyed or badly damaged. The demands made on the control system during nuclear operation are such that protective circuit breakers were specifically omitted from the control system. In case of emergency, the drive motors will continue to run, in an attempt to shut the reactor down, until some external action is taken, i.e., manual shutoff or component failure.

After the damage was repaired, examination of the relay rack wiring during running of the drive motors disclosed audible and visible arcing through the cellulose acetate insulation from wires to the mounting panel and within the 100-pin Amphenol connectors used for switchover between reactor and dry dock. These connectors are rated for 600 volts rms.

Because the entire control system is ungrounded, no short circuits developed. A short circuit in an ungrounded system requires a wire-to-wire fault or two simultaneous ground faults on different wires.

The arcing was observed to occur most frequently upon starting of the drive motors, a little less frequently upon stopping the motors, and occasionally while the motors were running at normal speed.

Inductive surges from motor windings are usually to be expected only on circuit interruption. As these drive motors are wound for three speeds, starting current in this case may conceivably induce surges in unused windings. But it is inconceivable that squirrel-cage induction motors could produce high voltage surges while running.

Of possible significance was the observation of abnormally high noise levels on the intercom system between the reactor and the control center. This was on the same day arcing not attributable to starting or stopping transients was observed.

A mockup of a drive motor circuit using a spare motor and gear unit and spare contactors, and using capacitors to simulate the capacitance of the cable wiring, was constructed and operated in the control center shop. Extensive tests failed to produce a single arc, even though spark gaps placed in the circuit were narrowed to the minimum gap detectable with the naked eye.

A check of circuit-to-ground leakage of wiring from relay rack to reactor and dry dock showed consistent values of about one-millionth mho per wire. No faulty circuits were detected.

These circumstances have been presented at length because they constitute a serious problem which is not yet fully understood. Numerous possibilities for recurrent difficulty may exist since 100,000 ft of hookup wire was used. To summarize briefly, arcing was observed in circuits in which a particular type of 16 gauge wire was used. The wire met the specifications but had a thinner lacquer coating on the cellulose acetate insulation than previously used 18 gauge wire having identical specifications as to insulation and voltage limits but supplied by a different manufacturer. Attempts to reproduce and measure the voltage surges producing the arc phenomenon were unsuccessful.

Two remedial measures have been taken which seem to have solved the problem empirically. Each phase of the 480 volt input has been grounded to the panel through three, series-connected, 60 watt, 120 volt light bulbs to allow possible high voltage static charges to bleed off, and a 4 microfarad capacitor has been connected between each motor winding lead and ground at the relay rack to reduce inductive surges from the motors.

D. Engineering Design (R. E. Heffner)*

1. Primary Coolant Water Sampling Station

A sampling station was added to the primary coolant system to obtain coolant samples for chemical analysis. Because of the inherent danger from water under the maximum operational conditions of the primary system, personnel protection has been provided by enclosing the station in a sheet metal housing with a "Plexiglass" viewing window.

2. Apparatus for Void Experiments

Design has been essentially completed and fabrication started on experimental apparatus for void coefficient measurements in the Spert III core. The equipment consists of a pump, manifold system, and 208 small diameter tubes for insertion between fuel plates. Through a combination of valves in the manifold system, various configurations of the tubes can be voided, either by opening to atmosphere and flashing the enclosed water to steam at high temperature pressure operation or by simply evacuating water from the tubes at low reactor water temperatures.

* Division Engineering Branch

IV. DATA REDUCTION AND INTERPRETATION

A. Photographic Investigations of Reactor Shutdown Mechanisms (R. W. Miller)

1. Review of Previous Studies

The type of shutdown mechanism which manifests itself at any time during reactor operation and the magnitude of its effect depend upon reactor design and upon the power history. It is convenient to correlate the observations from the mechanism studies with the primary variables measured during the transient tests, i.e., the initial asymptotic periods, the power levels reached, the fuel plate surface temperatures at the time of peak reactor power and the maximum fuel plate surface temperature, and also with the observed transient power burst shapes. It has been found that for those transients which are of relatively low power and which do not exhibit fuel plate temperatures above the saturation point of water, analysis based upon known mechanisms leads to shutdown models in agreement with experimental results.(9) Transients in this region apparently have simple shutdown, mostly due to thermal expansion effects. However, for those transients which exceed saturation temperature, this simple analysis is inadequate and further experimental observations are required to determine the type and effect of the mechanisms which exist. Water boiling, which can produce large negative reactivity effects, is expected to be important in short period transients. Radiolytic gas may also be important but experimental evidence is needed to determine which, if either, of these mechanisms is responsible for initial shutdown during and just prior to the time of peak power of a transient. It is possible that although saturation temperatures are reached prior to peak power, a state of superheat may delay onset of boiling and leave initial shutdown to be explained by other mechanisms. Delays of this type have been demonstrated by the transient boiling studies of Rosenthal.(10)

To investigate the importance of boiling shutdown, an experiment was conducted recently in which the Spert I A core was coated with several layers of thermo-setting plastic.(11) If boiling were important to initial shutdown, the time required for nuclear heat to pass through this layer of plastic should produce a significant effect upon the power levels ultimately reached in a transient. Results of this experiment were ambiguous, but indicated that the boiling process probably was not important to initial shutdown. Following this, a well-controlled experiment was conducted using small test cells with and without fuel.(1) These experiments indicated that voids were specifically needed for shutdown and were probably not due to radiolytic gas. A further refinement of these experiments (see p. 3) demonstrated that pressure and volume changes occurred simultaneously with a rise of plate temperature through the boiling point. Within experimental error no delays were observed regardless of initial period or pressure. This is strong evidence for boiling and leads to disbelief of inferences from the insulated core experiment.

2. Photographic Study

As a visual assistance to the study of initial shutdown, high-speed photography has recently been incorporated in in-pile experiments. Based upon the assumption that boiling is the phenomenon to be investigated, a mock-up of the experiment was first produced in the laboratory, primarily to develop techniques to simulate the rapid heating rates of the reactor. Boiling mercury was allowed to fall against the back face of a thin (0.005 inch) stainless steel diaphragm. Boiling of stagnant water on the front face of the diaphragm was then photographed at approximately 6000 pictures per second with a Fastax camera.

Following this work a water-filled test cell, containing a small piece of fuel plate with attached thermocouple, was constructed for use in the reactor. The cell was placed in the high flux reflector region of the Spert I P core. Boiling within the cell was photographed through a periscope. During each transient the thermocouple output and reactor power were recorded on an oscillograph. A synchronizing signal provided a correlation between the photographic and oscillographic records.

Motion pictures showing the onset of boiling were obtained on four short period transients. Three of the transients had initial asymptotic periods of approximately 10 msec; the fourth, approximately 7 msec. At the time when boiling was first observed on the 10 msec tests, the indicated temperature rise rate was between 5.7 and 6.7×10^3 °C/second. In all cases, the onset of boiling was characterized by nucleate bubbles. Less than 5 msec after boiling began in the 10 msec transients, the bubble population had increased to the extent that interaction between bubbles resulted in the formation of large composite steam bubbles covering about 30% of the observed area. The indicated maximum superheat was about 40°C. Immediately after the peak temperature was reached, the temperature dropped rapidly and the composite bubbles collapsed, leaving only a few small film regions which appeared to vibrate until the temperature fell below boiling.

Behavior in the 7 msec transient was markedly different. The 7 msec transient had an indicated temperature rise rate at boiling of 8.5×10^3 °C per second. As the temperature rose above boiling, the initial nucleation coalesced as before into larger composites and these again into still larger multiply-connected regions resembling "fingers", to the extent of covering 50% of the observed area. An indicated maximum superheat of 90°C was obtained, the peak of which was sustained for about 10 msec. During this plateau period the regions of steam were highly mobile, changing rapidly in shape but not in apparent volume. With falling temperature, these regions decreased in volume and mobility to thin stationary regions of steam covering less than 10% of the observed area. Finally, as the temperature fell below 100°C, all steam regions disappeared leaving only a few small bubbles attached to the fuel plate surface.

In these developmental experiments, it was found that the incidence of boiling observed photographically did not agree with the recorded fuel plate temperature readings, which were 7 to 18°C below the boiling point. These discrepancies have tentatively been attributed to poor response

time of the thermocouple, which consists of 0.010 in. chromel-alumel wire stud-welded on the stainless steel fuel plate. (These observations on thermocouple response may apply to some other experimental situations but do not apply to the earlier Spert experiments in which transient temperature data were obtained from aluminum cores using thermocouples peened into the fuel plate cladding.) Work is under way to improve the thermocouple response for future experiments.

In this experiment the temperature and power histories of the test fuel should not be construed as representative of the over-all reactor behavior. The reactor core had been rearranged to accommodate the test cell in a position favorable from the standpoint of flux peaking. For this reason, although accurate timing may be put upon the photographic record of boiling, the reactor behavior should not be interpreted in terms of the observations on this plate. However, the photographic boiling record may be correlated directly with temperature history of the test fuel and by this method extended to similar temperature data obtained at any point in the core, other factors remaining constant.

The results obtained in this experiment indicate with little doubt that water boiling is a primary agent in initial reactor self-shutdown whenever temperatures permit.

B. Verification of Excess Reactivity Calculational Program (R. J. Wagner)

The computer program to obtain the excess reactivity k_{ex} , during a reactor transient from the experimental power data⁽¹²⁾ has been verified by an independent program based on the inverse problem, namely, solution of the kinetic equations for the power using the computed reactivity as input. In the analysis of reactor shutdown behavior, calculation of k_{ex} is of importance because of its general relation to the reactor period and power burst shape.⁽¹³⁾ In addition, the post-burst dip found in the computed k_{ex} curves⁽¹⁴⁾ was investigated to determine its sensitivity to certain features of the input data.

1. Description and Results

Comparison of the experimental power history with the power history calculated from the inverse program was the means adopted for verification of the k_{ex} program. The k_{ex} program operated on the original power data and yielded $k_{ex}(t)$. These computed values of $k_{ex}(t)$ were then inserted into the reactor kinetics equations in the inverse program to provide a numerical calculation of the power.

An IBM-650 library code⁽¹⁵⁾ for solving simultaneous differential equations was used to solve the reactor kinetics equations. This code evaluates systems of the form

$$\dot{Y}_i(t) = \sum_{j=1}^n A_{ij}(t, Y) Y_j(t) ; \quad i = 1, 2, \dots, n$$

the solution being advanced through successive fixed intervals, h , by a 4th order Simpson method of Runge-Kutta. In the present application to reactor transients, additional programming was provided to evaluate the non-constant matrix coefficients, which are dependent on $k_{ex}(t)$. The values of $k_{ex}(t)$ were entered as tabular data at variable spaced values of time, with intermediate values obtained by linear interpolation.

Fig. 20 shows the results of the verification for a sub-cooled 9.8 msec transient. Curve A is the experimental power trace. From these data $k_{ex}(t)$ was computed by the k_{ex} program. This is shown as Curve B. Curve C is the power obtained from the inverse program using Curve B as input. Excellent agreement is seen between power plots A and C.

To examine the sensitivity of the computed power to the dip obtained in the k_{ex} curve, B, the post-burst dip in k_{ex} was smoothed out and the smoothed curve, D, used as input data. This resulted in the power trace shown by Curve E. Very little difference in the power traces is observed from the elimination of the dip. To investigate the possibility of this similarity arising from numerical methods used in the solution of the kinetics equations, the two computed power curves, C and E, were themselves processed by the k_{ex} program. The results of this computation were identical with the Curves B and D in Fig. 20. These identical results have been derived through a complete circle of computation, from $k_{ex}(t)$ to power to $k_{ex}(t)$ using two independent computer programs, and the agreement obtained indicates that the numerical approximations to the kinetics equations utilized in each of these programs are sufficiently accurate for this application.

Fig. 20 - Power and k_{ex} as Functions of Time for a 9.8 msec Ambient Transient

A more rigorous test of the computer program is afforded by calculations for boiling transients since the post-burst dip in k_{ex} has been observed⁽¹⁴⁾ to be much more pronounced on transients initiated from boiling temperatures.

Fig. 21 illustrates the results obtained for a 5.8 msec boiling transient. Curve A is the original experimental power trace. Curve B is the derived $k_{ex}(t)$ and Curve C is the computed power using Curve B as input data. Good agreement is obtained between the power plots. The slight difference of about 2% in the initial asymptotic periods of the power curves arises from the use of a graph of $k_{ex}(t)$ to read off the input values of k_{ex} . The error in asymptotic period causes a slightly higher peak power in the computed power curve and a correspondingly higher energy release. The latter, in turn, leads to the power discrepancy found at the end of the burst as a result of the creation of a larger number of delayed neutron precursors. Curves F and G in Fig. 21 are the computed power curves obtained from the input curves, D and E, respectively.

Fig. 21 - Power and k_{ex} as Functions of Time for a 5.8 msec Boiling Transient

2. Conclusions

Since it is possible to reproduce the power data by using the k_{ex} data as input to an independent computer program, it is concluded that the k_{ex} program does not impose any limitations or introduce any extraneous behavior which would invalidate the results. In this way the program has been verified to produce reactivity values consistent with

the input power data. The fact that the k_{ex} obtained from the computed power is identical with the k_{ex} derived from the original experimental power is further evidence that the two computer programs are consistent. Since the k_{ex} program has been shown to be correct, it would follow that for the experimental power data analyzed, the dip in the k_{ex} curve is real.

However, based on postulated shutdown mechanisms, there is no ready explanation for this dip. In order to study the sensitivity of the dip to the power data, the dip in the k_{ex} curves was eliminated by replacing the dip by a straight line, and the power traces resulting from these variations in k_{ex} were obtained by the inverse program. Since the k_{ex} and the inverse programs are consistent, these power traces can be said to be the power traces necessary to eliminate the dip in the k_{ex} curve. Curves C and E of Fig. 20 illustrate the small changes in power data needed to remove the dip for an ambient 9.8 msec period transient. More information can be obtained from the boiling 5.8 msec period transient, in which the magnitude of the dip in the k_{ex} curve is much larger. For this transient, two variations in k_{ex} were made, and the resulting power traces, Curves F and G, show significant deviation from the power curve, C, derived from the unaltered k_{ex} . (The power curves derived from the altered k_{ex} curves for this transient are compared with the power curve obtained from the original k_{ex} data rather than the experimental power due to the difference in slope discussed earlier.) These two variations in k_{ex} can be considered as upper and lower limits on the behavior of k_{ex} for this region of the burst. Any suspected error in the recorded power which approaches the difference between the recorded power and the power obtained from the altered k_{ex} curves necessarily leads to doubt as to the physical existence of the dip. An experiment has been performed which shows that the recorded power could possibly be a factor of 3 or 4 times too high at this region of the burst due to expulsion of moderator from the entire core.⁽¹⁶⁾ Photographic evidence indicates that this is possible for a short period boiling transient. Thus, the conclusion is drawn that no significance can be given to the values of k_{ex} approximately a period beyond peak power. It is also evident that no attempt should be made to fit shutdown phenomena to the dip.

It appears that the effect of void formation on power detection is a plausible explanation for the appearance of this anomaly in the reactivity behavior. In a qualitative manner, this effect explains such features of the dip as the magnitude of the dip being greater for boiling transients than sub-cooled transients, and the magnitude of the dip increasing for decreasing reactor period. Besides the void effect, all the other known errors in power detection tend to make the recorded power higher than the actual power during the shutdown portion of the burst. The other recognized sources of error are the frequency response of the recording system and the recording of gamma build-up in uncompensated neutron detection chambers. While it can not be definitely proved that these effects are the explanation for the appearance of the dip in the k_{ex} behavior, it is fairly certain that Curve G in Fig. 21 is a power curve more nearly approximating the actual reactor power, and thus the k_{ex} curve shown by Curve E is a better representation of the actual k_{ex} curve.

The purpose of this discussion has been only to verify that the k_{ex} computer program gives a valid solution of the kinetics equations for k_{ex} and to study reasons for the post-burst dip behavior. As indicated in the reference(12), statements about the actual k_{ex} behavior when the reactor is greatly shutdown must be qualified by considerations of the limitations of the one-group lumped parameter kinetics equations which have no provision for time dependence of the parameters.

V. ENGINEERING

A. Investigation of Materials for Threaded Fasteners (R. E. Heffner)

A test program has been initiated to evaluate various materials of construction for threaded components used in high temperature demineralized water service. The test materials have been fabricated into identical bolt specimens and threaded into a Type 304 stainless steel plate. The specimens are to be repeatedly autoclaved to 670°F in de-ionized water with periodic inspection for corrosion and susceptibility to galling. The bolt materials being evaluated are the following:

- AISI Type 431 Stainless Steel, annealed
- AISI Type 431 Stainless Steel, heat treated to Rc30
- 17-4 PH Stainless Steel, condition A
- 14-4 PH Stainless Steel, condition H-900
- A-286 Stainless Steel, heat treated to Rc34
- A-286 Stainless Steel, heat treated to Rc30
- Commercial Carbon Steel, "Electrolized"
- AISI Type 304 Stainless Steel, "Electrolized"

B. Spert II

1. Status of Construction (C. R. Montgomery)

Installation of the primary piping system is complete, and the system has been satisfactorily hydro-tested at 750 psi. Work continues on the installation of D₂O piping and miscellaneous small piping.

Calibration of process instrumentation and checkout of process control electrical circuits is now in progress.

2. Mechanical Design (R. E. Heffner)

The fit of the core components and the operation of the locking mechanisms of the core internal structure were satisfactorily checked using core filler pieces having the same external dimensions as the fuel assemblies.

3. Control System Wiring (D. T. Jones)

IBM-tabulated wiring schedules have been revised to include modifications in design of control system wiring made during installation. "As-built" schematic drawings are being prepared to facilitate checkout and maintenance of the control system.

4. Fuel Procurement (T. R. Wilson)

The initial core for the Spert II reactor, consisting of 68 Type B fuel assemblies, 30 control rod fuel sections and 750 removable fuel plates has been completed by the fabricator and delivered to Phillips Petroleum Company. All items have been inspected and approved.

VI. REFERENCES

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