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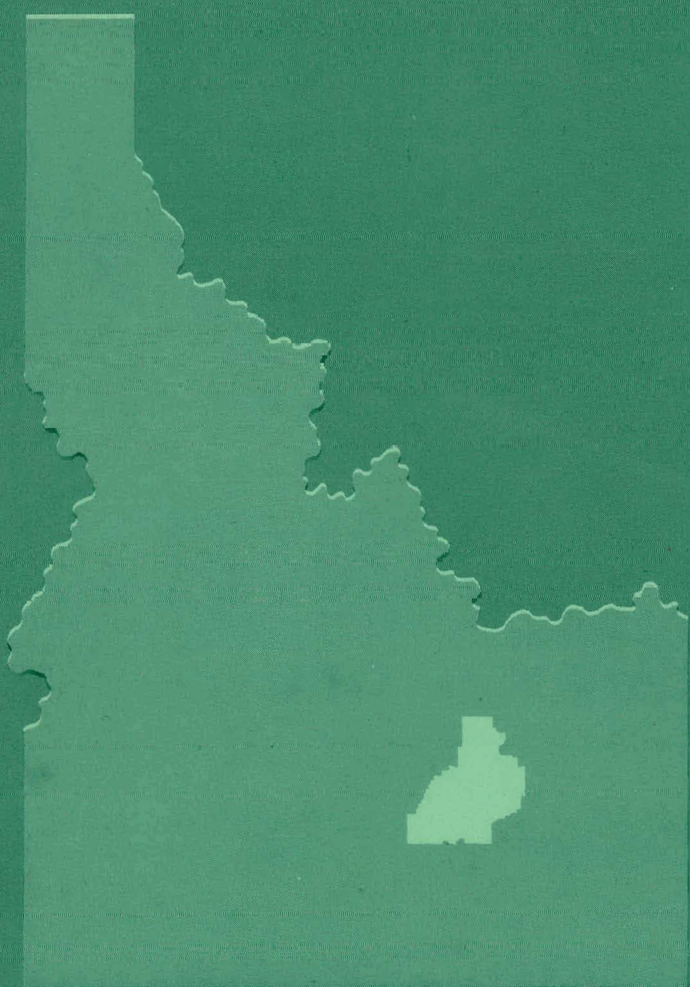
SPERT PROJECT
October, November, December, 1960

F. Schroeder, Ed.



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NATIONAL REACTOR TESTING STATION
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SPECIAL POWER EXCURSION REACTOR TESTS

QUARTERLY TECHNICAL REPORT
SPERT PROJECT
OCTOBER, NOVEMBER, DECEMBER, 1960

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Atomic Energy Division
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I. SUMMARY

SPERT I - The current series of tests for the investigation of reactivity-compensating mechanisms in plate-type, highly enriched reactors has now been completed. During this quarter, tests were conducted in the in-pile test cell with a set of aluminum-clad fuel plates which were coated with insulating plastic in order to investigate changes in transient boiling due to the presence of the thin layer of plastic adjacent to the fuel plate surface. Tests were also performed with aluminum-clad fuel plates having a 60% greater fuel density (0.061 g U^{235}/cm^2) than previously tested plates. In both series of tests, motion pictures of transient bubble formation were obtained in addition to the measurements of moderator displacement, fuel plate surface temperatures and transient pressures. Reduction and analysis of data obtained from these tests and from previous capsule tests are in progress.

As a preliminary part of the forthcoming kinetic experiments with a low-enrichment oxide core in Spert I, in-pile tests have been performed to investigate the possibility of damage to the long thermal-time-constant fuel rods when subjected to short-period power excursions, and also to test the effectiveness of an instrumentation technique for measurement of UO_2 fuel temperatures within the rod. A fuel rod, instrumented with ten pairs of internal and cladding-surface thermocouples, ruptured during the exponentially rising portion of an 8.2 msec-period excursion. Subsequent tests of a fuel rod with no cladding penetrations produced no discernible damage or distortions of the rod for periods as short as 7.2 msec. These tests support the postulation that the rupture of the first rod was related to the cladding penetrations made for installation of the internal thermocouples.

SPERT II - A knowledge of the effective dynamic value of the ratio of prompt neutron lifetime, ℓ, to the effective delayed neutron fraction, β_{eff}, is required for analysis of the short-period transient behavior of reactors. As a part of a continuing program of comparing the measurements of ℓ/β_{eff} by various static techniques with measurements based on short-period power excursion data, ℓ/β_{eff} has been determined for a D_2O-moderated core in Spert II by analysis of the statistical behavior of the neutron population in the subcritical system. A value of 0.11 sec was obtained for an expanded configuration of fuel assemblies in D_2O. This value will be compared with that which will be obtained from the power-excursion tests with this core.

Measurements have also been made of the neutron flux and the void coefficient of reactivity as functions of position in this core. These measurements will be of value in the analysis of the forthcoming kinetic tests.

SPERT III - Additional data have now been reduced from the previously reported investigations of the effects of system pressure and coolant flow rate on room temperature power excursions in Spert III. Previously reported results are briefly reviewed in this report and additional data are presented. For 10 msec-period excursions at 2500 psi system pressure, the power maximum is about 30% higher than is observed at atmospheric pressure. The fuel plate surface temperature traces for this test show indication of the rapid decrease in heat transfer rates after the plate surface reaches the critical point for water. The addition of 18 ft/sec coolant flow velocity for similar short-period excursions produces essentially no change in the peak power or temperatures reached, but the temperature behavior indicates that good heat transfer is maintained in the region of the critical point for the flow tests. Detailed examination and correlation of these test data are continuing.

Rippling and bowing of the stainless-steel-clad Spert III fuel plates has been observed following short-period power excursion tests. In some cases the combined bowing and rippling, which is attributed to the large thermal gradients present in the plates during the excursions, is sufficient to close more than half the width of the normal water channels. Blisters have also been observed on the outside fuel plates of three fuel assemblies. Preliminary metallurgical examination of the blistered plates has revealed the presence of a few intergranular cracks, but none of these appears to have penetrated the cladding and they may have been formed as a consequence of the blistering. The cause of the blistering has not yet been determined, although there appears to be some correlation between fuel segregation at the fuel-clad interface and the location of the blisters.

ENGINEERING - Channel flow distributions have been obtained for a Spert III fuel assembly in the ETR hydraulic test facility. For a total assembly flow rate of 400 gpm, corresponding to a total reactor flow of 20,000 gpm in Spert III, the standard deviation from the mean for channel flow rates is 2.1%, and the maximum deviation from the mean is 19.6%.

II. SPERT I

A. Mechanism Studies

The experimental in-pile capsule program[1] for the study of the kinetics of water moderator expulsion from the regions surrounding fuel plates during the course of a power excursion, has been completed with tests conducted on two additional sets of capsule fuel plates.

The first of these additional sets of plates was similar to the initial set of seven aluminum-clad fuel plates used in the capsule[1]. In addition, the entire surface area of the fuel plates was covered with a thin coating of insulating plastic. The plastic used was lithcote, LC-34, having a diffusivity approximately 30% less than that of water and, consequently, an insulating property somewhat better than that of water. The thickness of the coating was approximately a few thousandths of an inch thick. The purpose of these insulated fuel plate tests was to investigate changes in transient boiling effected by the replacement of the few-mil-thick layer of water adjacent to the fuel plate surface (in which region large temperature gradients are developed) by a non-boiling plastic having heat conduction properties not very different from that of water. Five power excursion tests were performed at 0 psig static capsule pressure, with initial reactor periods in the range between 65 and 10 msec. Motion pictures of transient bubble formation were obtained during these tests.

The final capsule fuel loading consisted of a set of three aluminum-clad fuel plates, similar to those used in the earlier tests except for a 60% greater fuel density (0.061 g U^{235}/cm^2). Three groups of power excursions were performed at reactor periods of approximately 19, 14 and 9 msec. Within each group of tests the capsule static pressure was varied from 0 psig to a maximum of about 2400 psig. Motion pictures of transient boiling were obtained in addition to the measurements of piston displacement and velocity, fuel plate surface temperature, and transient pressure in the capsule. Reduction and analysis of the data obtained from these tests and from the previous capsule tests is in progress.

B. Power Excursion Tests of a Low Enrichment UO_2 Fuel Rod

As a preliminary part of the forthcoming experimental program on a Spert I oxide core comprised of 4%-enriched UO_2 fuel rods[2], in-pile power excursion tests of two such rods were conducted at Spert I using the highly enriched, water-moderated, plate-type P-core as a neutron source. These tests were performed to obtain experimental data relating to (a) the likelihood of cladding damage when such long thermal-time-constant fuel rods are subjected to the short-period power excursions planned for the oxide core experimental program, and (b) the effectiveness of the instrumentation technique used for measuring the UO_2 fuel temperature within the rod.

The fuel rod used in these experiments is a welded-seam stainless-steel tube (6 ft long, 0.500 in. OD and 0.028 in. wall thickness) containing 1600 g of 4%-enriched UO_2 powder compressed to an effective density of 9.45 g/cm^3. During the course of a power excursion, the relatively low heat transfer properties of this fuel rod may be expected to give rise to large temperature gradients and cladding stresses, leading to the possibility that the cladding may fail if the power transient is sufficiently severe. In addition, the probability of cladding failure may be expected to be further enhanced if penetrations of the cladding are made for the installation of thermocouples for monitoring the internal temperature of the fuel rod. The fuel rod tests that were performed consisted of a series of power excursions on a fuel rod instrumented with internal thermocouples and of a similar series on a fuel rod containing no internal thermocouples.

The fuel rod in the first test series was instrumented with ten pairs of internal and cladding-surface thermocouples, spaced one inch apart along the length of the fuel rod. The thermocouples used were 5 mil chromel-alumel. The internal thermocouples were inserted in holes drilled diametrically through the fuel rod, with the junctions at the center of the fuel. The holes in the steel cladding were covered with a water-proofing epoxy. The fuel rod was placed in an open-top, water-filled aluminum container in a centrally located flux trap of the P-core, created by removing one of the P-core fuel assemblies. The internally instrumented fuel rod was subjected to a series of ten self-limiting power excursions in which the reactor period was varied from 30 sec to 8.2 msec.

During the exponentially rising portion of the 8.2 msec excursion, a 12-in. split in the stainless steel cladding occurred along the length of the fuel rod, approximately following the line of holes previously drilled through the rod for the installation of thermocouples. In this test, the maximum power density ($\approx$ 50 kw/cm^3) obtained in the fuel rod and the maximum temperature gradient ($\approx$ 550°C/cm) between the center of fuel rod and cladding surface were greater than for any of the previous tests. As a result of the cladding rupture, approximately 700 g of the UO_2 powder was dispersed throughout the reactor vessel water.

Following decontamination of the reactor vessel, a series of twelve self-limiting power excursions was run on a fuel rod with intact cladding. In this series the reactor period varied from 57 to about 7.2 msec. No discernible damage or distortion to the fuel rod occurred as a result of these transients, supporting the postulation that rupture of the first fuel rod was related to the cladding penetrations made for installation of the internal thermocouples. It appears possible that failure of the water-proofing epoxy seals, perhaps as a result of radiation damage, permitted water seepage into the fuel, resulting in a steam explosion during the power excursion.

The results of these limited tests suggest that for reactor power excursions with periods as short as 7 msec the probability of fuel rod rupture is small for fuel rods with intact cladding, but that fuel rod rupture may be likely for a rod with cladding penetrations of the type tested which is left immersed in water following a power excursion. Pending further analysis of the data, no recommendations can be made regarding the application of internally instrumented fuel rods in the Spert I oxide core tests.

III. SPERT II

A. Measurement of ℓ/β_{eff}

A knowledge of the effective value under transient conditions of the ratio of prompt neutron lifetime ℓ, to the effective delayed neutron fraction β_{eff}, is required for prediction of the short-period kinetic behavior of a reactor. This quantity can be obtained directly for the Spert reactors from analysis of super-prompt-critical power excursion data. Several static methods are frequently used for determination of ℓ/β_{eff} and these methods have been found to yield results which are in good agreement with the dynamic measurement for the light-water-moderated Spert I reactors[3]. An additional method, based on analysis of the statistical behavior of the neutron population measured by a counting-type experiment, has been employed as a part of the static test program for the Spert II BD-22/24 core in D_2O, shown in Fig. 1, so that the value thus obtained can be compared with the dynamic measurements which will be obtained as a part of the kinetic experiments. The method employed utilizes the relationship originally derived by deHoffman[4] which relates the experimentally measured variance-to-mean ratio of counts as a function of counting time to system parameters, one of which is the prompt neutron lifetime. Since this method exploits the theoretical prediction that the neutron population fluctuations are themselves an integral property of the system, measurements of fluctuations at a single point may be utilized to yield integral properties of the system.

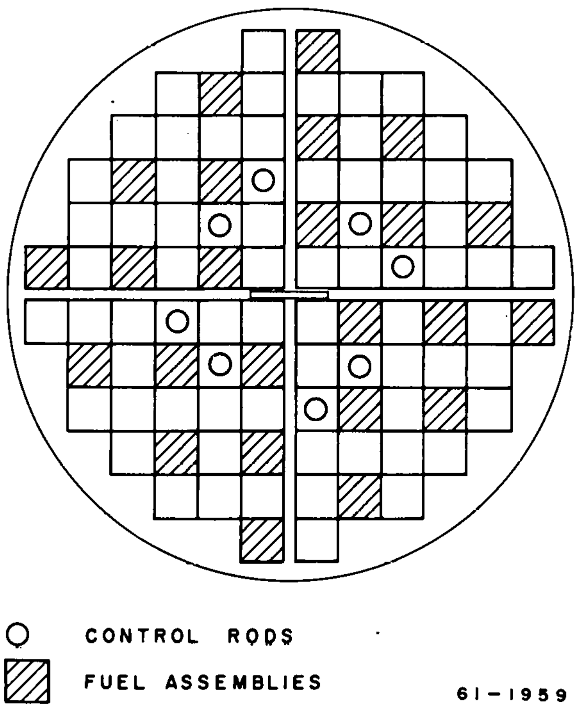

Fig. 1 - Spert II BD-22/24 Core Configuration

The counting apparatus consisted of three B^{10}-lined pulse chambers located around the periphery of the core, associated pre-amps, linear amplifiers, mixer, and a crystal-oscillator-controlled gated scaler. The frequency response of the system was approximately 100 kc which was more than adequate since the highest count rates observed were less than 2 kc per channel. The noise level in the measuring system was low enough to have no effect on the experimental data. This was checked by observing that a random, or Poisson, counting distribution was obtained from a Ra-Be source for several gate times.

The measurement technique consisted of counting the number of neutrons detected during a fixed counting or gate time. This was repeated for many gates and the number of counts obtained at the fixed gate time was analyzed for the ratio of variance-to-mean. Data on the behavior of the variance-to-mean ratio as a function of gate time was obtained by repeating the measurements for gate times, ranging from 0.001 to 1.000 sec. The effect of gate time was studied at five subcritical reactivities, varying from approximately 0.99 to 0.999. The experimental data were correlated in terms of the theory in order to evaluate ℓ/β_{eff}.

The variance-to-mean ratios obtained from the experimental data were fitted with the theoretical relation:[5]

$$\frac{V_c^{\,T}}{M_c} = 1 + \frac{E\,k_p}{\left(\Delta\,k_p\right)^2}\left[\frac{\overline{\nu^2} - \bar{\nu}}{\bar{\nu}}\right]\left[1 - \frac{\left(1 - e^{-aT}\right)}{aT}\right] \qquad (1)$$

where:

$\frac{V_c^{\,T}}{M_c}$ = variance-to-mean ratio of c = $\frac{\overline{c^2} - \bar{c}^2}{\bar{c}}$

c = counts per gate time

E = detector efficiency

k_p = prompt neutron reproduction factor

$\Delta\,k_p$ = $1 - k_p$

$\bar{\nu}$ = average number of neutrons emitted per fission in U^{235} = 2.47

$\overline{\nu^2}$ = dispersion in the number of neutrons emitted per fission in U^{235} = 7.32

a = $\frac{\Delta\,k_p}{\ell}$

ℓ = mean prompt neutron lifetime

T = gate time

and the bar indicates mean value.

This equation may be written in the form

$$W = \frac{V_c^{\,T}}{M_c} - 1 = Z\,X(a,T). \qquad (2)$$

The constant Z is a combination of the fixed variables (constants) and X is a function of gate time and "a". A nonlinear least-squares fit for Z and "a" was then made using an IBM-650 program. The values of k_p were calculated from rod worth data taken prior to the experiment.

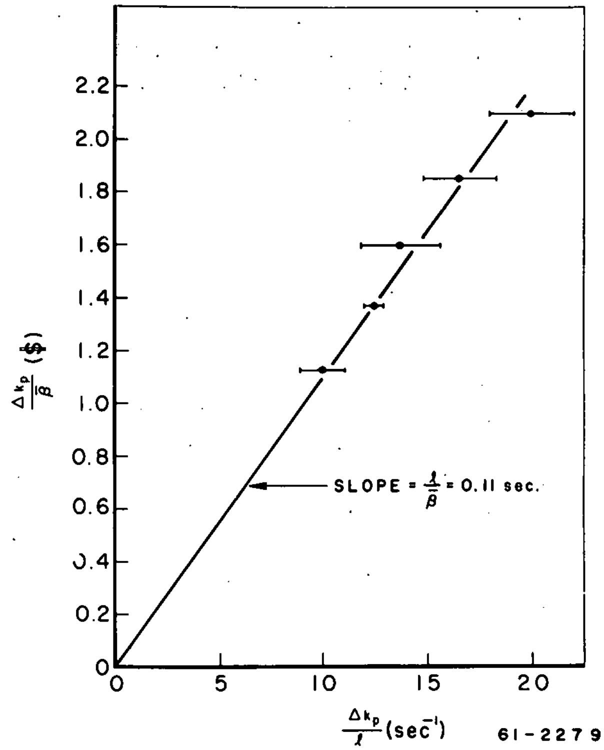

Fig. 2 - $\Delta k_p/\beta_{eff}$ vs $\Delta k_p/\ell$ for Spert II BD-22/24 Core

Having obtained values of "a" for each multiplication, a least-squares fit of "a" $\equiv \Delta k_p/\ell$ vs $\Delta k_p/\beta_{eff}$ gives a straight line through the origin with slope ℓ/β_{eff}, as shown in Fig. 2. The value of ℓ/β_{eff} thus determined is 0.11 sec, for the Spert II BD-22/24 expanded D_2O core. Analysis of the short-period power excursion data to be obtained for this core will yield a dynamic measurement of ℓ/β_{eff} which can be compared with the above value.

B. Measurement of Neutron Flux Distribution

In order to provide information for future use in the analysis of kinetic experiments, the thermal neutron flux distribution has been measured in the D_2O-moderated BD-22/24 core shown in Fig. 1. The flux map was obtained for one quadrant of the core by the activation of 250 gold foils. Fig. 3 shows the relative neutron flux profile

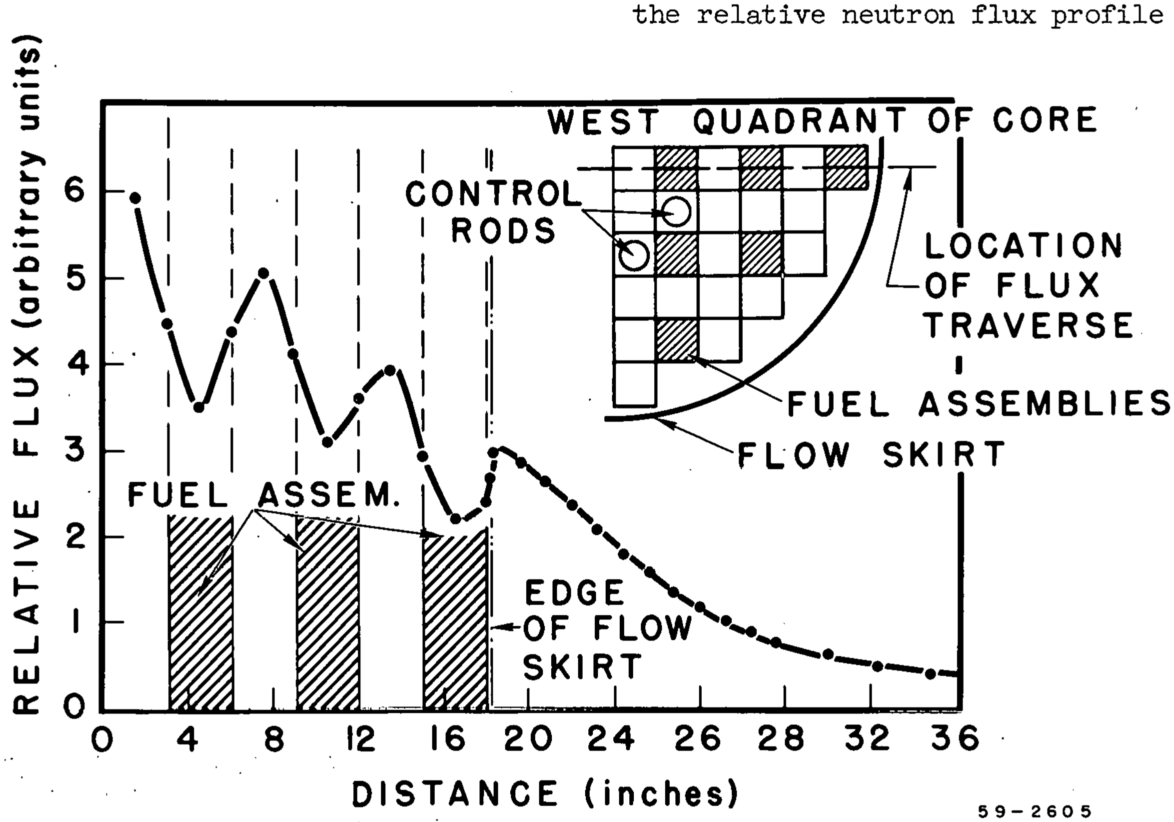

Fig. 3 - Spert II Horizontal Flux Profile

for a horizontal traverse through one row of fuel assemblies, as indicated in the figure. As would be expected, substantial flux peaking occurs in the water gaps between the fuel assemblies of this core configuration. A detailed plot of the flux distribution within a fuel assembly is shown in Fig. 4.

C. Measurement of Void Coefficient

Analysis of the self-limiting power excursion data to be obtained for the D_2O-moderated Spert II BD-22/24 core, shown in Fig. 1, will require some knowledge of the local importance of steam voids. Preliminary measurements have recently been made of the reactivity worth of voids within fuel assemblies in this core. One fuel assembly was divided into three gas-tight sections of equal volumes. Each section included the full vertical height of the assembly. Using helium gas to simulate voids, the reactivity worths of voiding the central third of the assembly, the outer two-thirds, and the complete assembly were determined for each lattice position of one quadrant of the core. The reactivity changes were determined by period measurements. The results of these measurements are indicated in Fig. 5. Within the limits of experimental error, the values obtained from the fully voided assembly agree with the average void worth for the partially voided assembly.

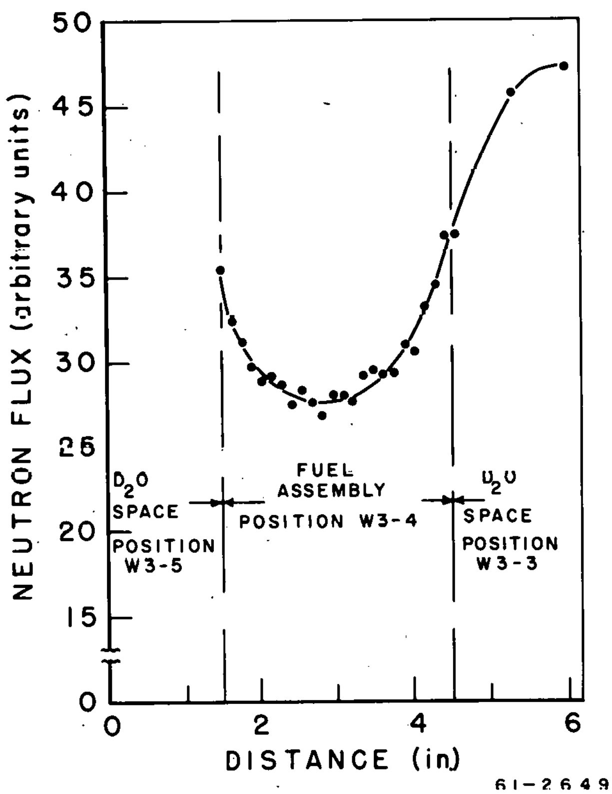

Fig. 4 - Spert II Horizontal Flux Profile within a Fuel Assembly

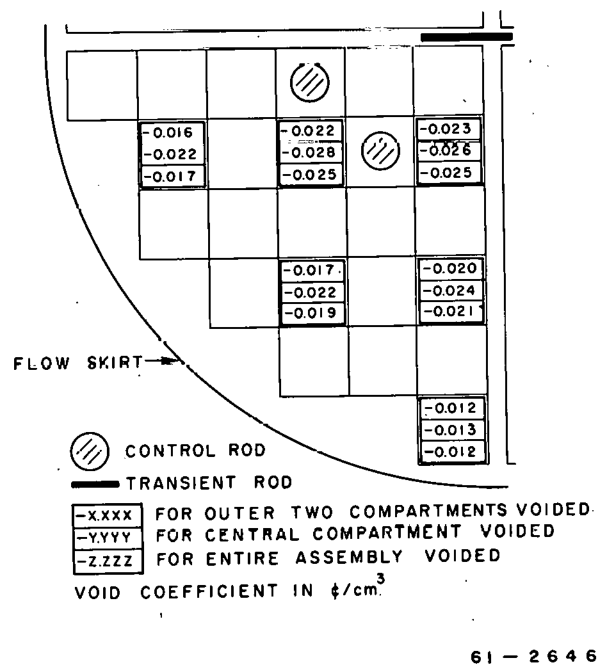

Fig. 5 - Void Coefficient of Reactivity as a Function of Core Position for Spert II BD-22/24 Core

IV. SPERT III

A. The Effect of System Pressure on Room Temperature Power Excursions

One of the conclusions which has been drawn from previous experimental investigations in Spert I for water-moderated, highly enriched systems operating at atmospheric pressure is that, for short-period power excursions, the formation of steam voids is an important contributor to the reactivity compensation which limits the power burst. An increase in the system operating pressure will delay or suppress entirely the onset of moderator boiling during the initial power burst for an excursion initiated at room temperature. Investigations of the effects of system pressure on room temperature power excursions in Spert III have been reported in a previous quarterly report(6). Since an additional test series has recently been performed and the preliminary data have been somewhat refined, a brief review of the effect of system pressure can be presented.

The tests which have been performed investigated the response of the reactor to step-wise insertions of reactivity from an initial power level of about 10 w, at room temperature (~ 30°C), with no forced coolant flow, for a range of initial asymptotic periods from 10 sec to 11 msec. The initial system pressure was varied from 0 to 2500 psig. The tests discussed here were performed with the primary coolant system liquid-full.

For power excursions with initial asymptotic periods longer than about 40 msec, where boiling does not occur prior to the power maximum at atmospheric pressure, elevation of the system pressure has no effect on the power burst behavior, as would be expected. For shorter period tests, the effect of elevated system pressure was to increase the maximum power of the burst by about 30% with most of the change occurring in the first 50 psi of pressure increase. The time of occurrence of the power peak was slightly delayed and the burst was slightly broadened.

Fig. 6 shows some of the data for tests with initial asymptotic periods of about 18 msec, corresponding to a reactivity step of about $1.12. The temperature of the hottest measured fuel plate surfaces at the time of the power maximum, $\Theta(t_m)$, is plotted as a function of the system pressure. Data are shown from the two different test series which were performed using different sets of fuel assembly thermocouples. The saturation temperature is also shown as a function of pressure. It is seen that $\Theta(t_m)$ increases by about 20% with the first 100 psi pressure increase but that, for pressures above 200 psia, $\Theta(t_m)$ is less than the saturation temperature and is approximately constant with further pressure increase. The rise of temperature to about 200°C is apparently sufficient to limit the excursion by nonboiling expansion, and increasing the pressure further does not increase either the peak power or the temperature at the time of peak power.

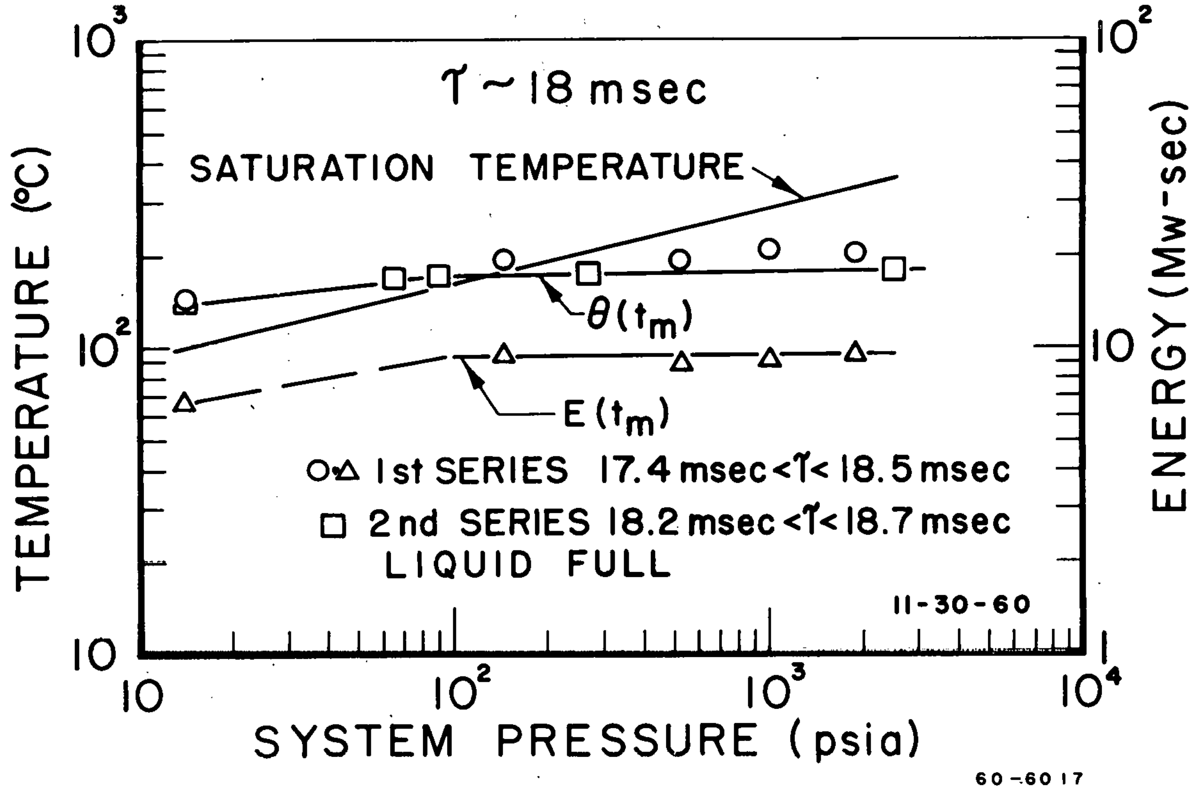

Fig. 6 - $\Theta(t_m)$ and $E(t_m)$ vs System Pressure for 18-msec Tests

The nuclear energy release at the time of the power maximum, $E(t_m)$, is also shown as a function of system pressure in Fig. 6, for the first series of data.

Fig. 7 shows the peak temperature reached by the hottest measured fuel plate surface for the test series with approximately 18 msec initial periods. For system pressures above 2000 psi it appears that boiling does not take place at any time during the test, even after the power rise has been halted.

Fig. 8 shows the $\Theta(t_m)$ data for excursions with periods of about 11 msec, corresponding to reactivity insertions of about \$1.25. For these shorter period tests, the fuel plate surfaces do reach saturation temperature at the time of the power maximum until the system pressure exceeds about 2000 psia, but boiling does not occur before the power peak at a pressure of 2500 psi. As for the longer period tests, the largest change in peak power and hence in the energy release $E(t_m)$ occurs in the first 100 psi pressure increase. The energy released up to the time of power peak is approximately 18 Mw-sec for pressures above 200 psi.

The maximum measured fuel plate surface temperatures for the 11-msec tests are shown in Fig. 9. Although the maximum temperature increases more slowly with pressure than does the saturation temperature, Θ_{max} is still above boiling at 2000 psi. The rapid increase in the maximum temperatures reached for pressures above 2000 psi is reproducible and

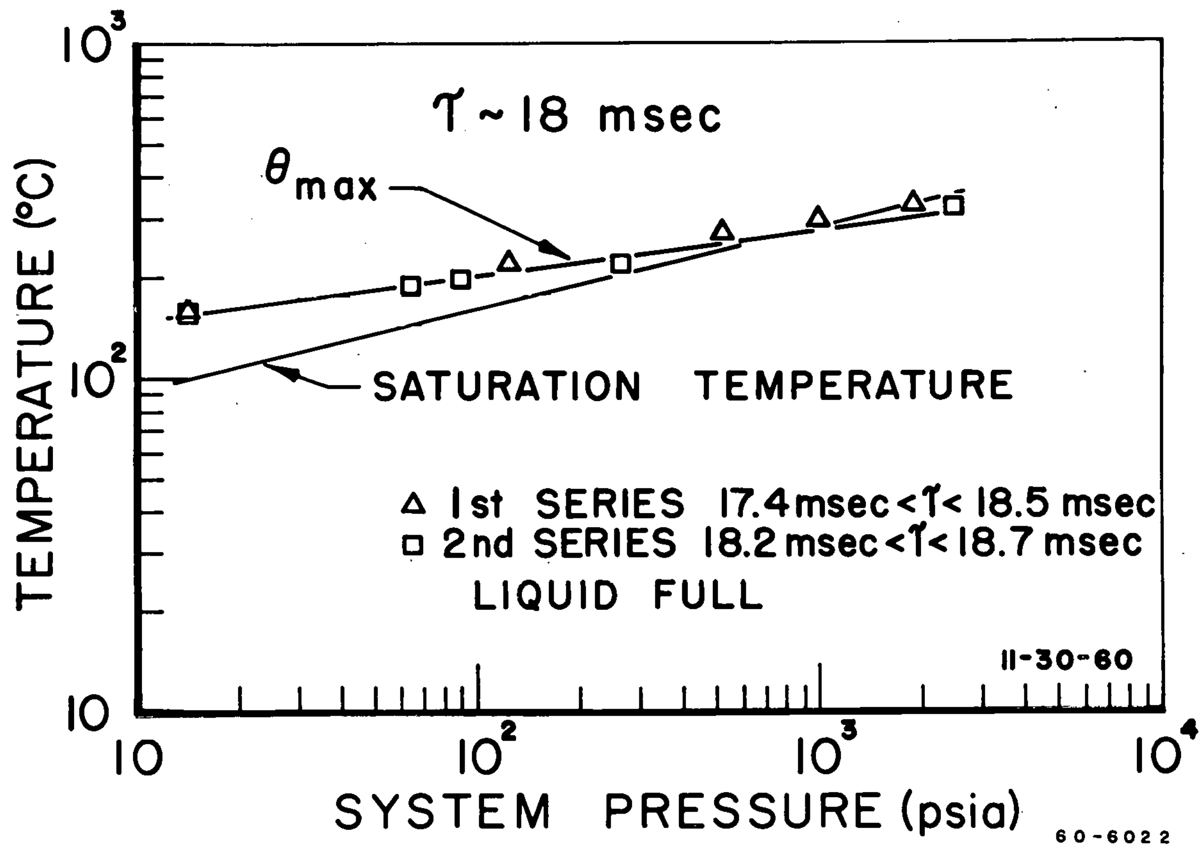

Fig. 7 - Maximum Fuel Plate Surface Temperature vs System Pressure for 18-msec Tests

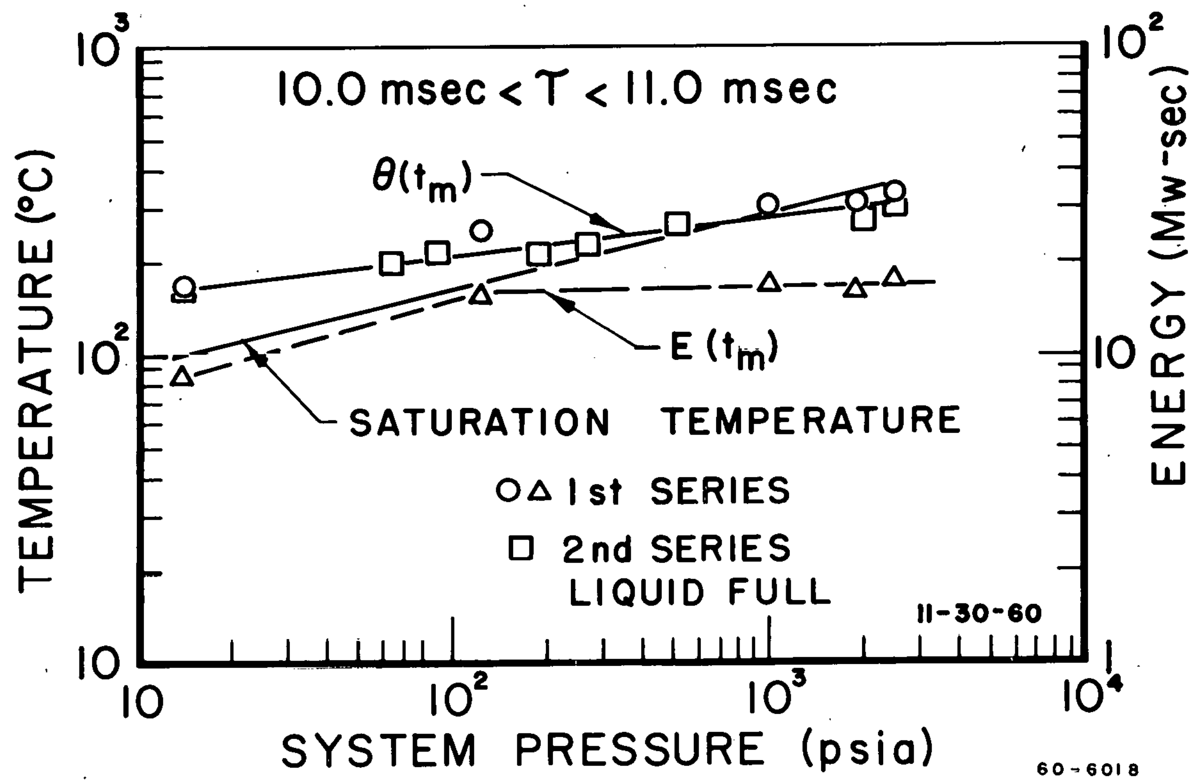

Fig. 8 - $\Theta(t_m)$ and $E(t_m)$ vs System Pressure for 11-msec Tests

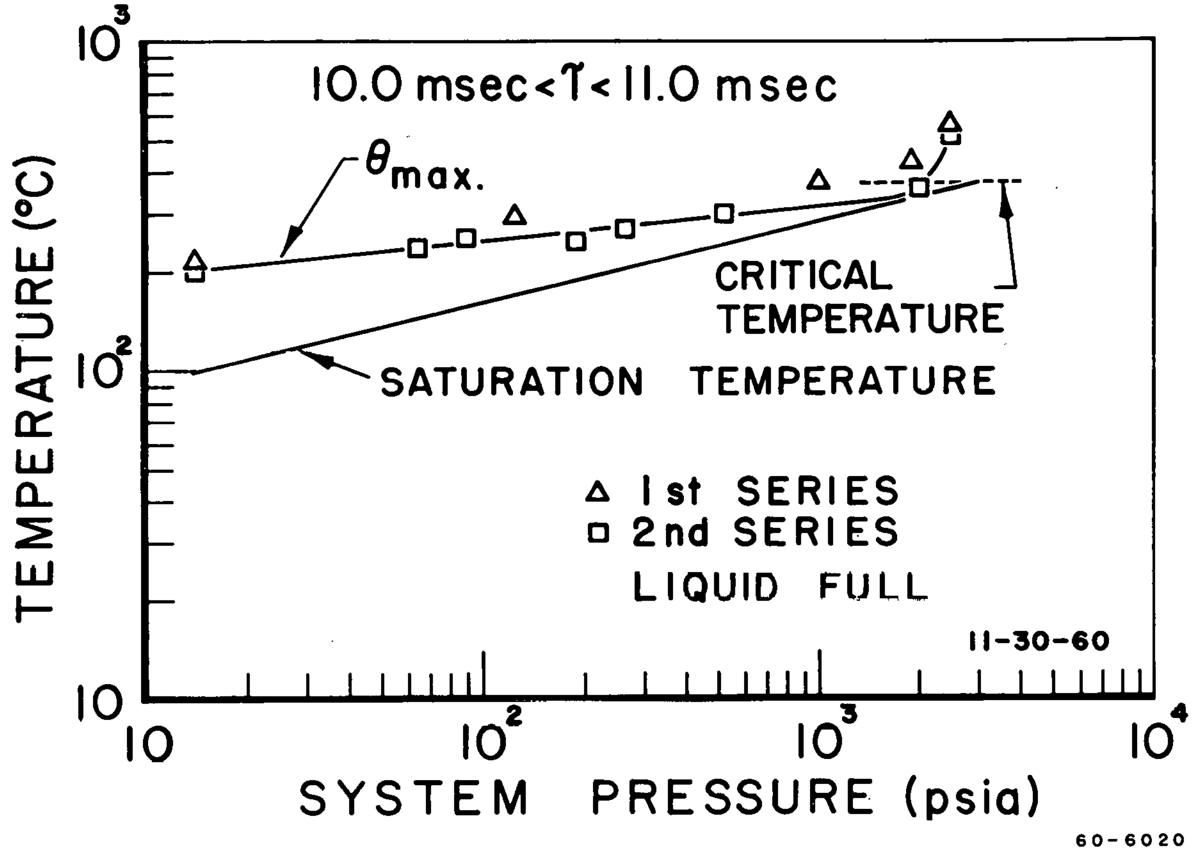

Fig. 9 - Maximum Fuel Plate Surface Temperature vs System Pressure for 11-msec Tests

is almost certainly associated with the rapidly decreasing heat transfer rates as the water immediately adjacent to the plate surface approaches the critical temperature.

In Fig. 10, the power and fuel plate surface temperature data are shown as functions of time during two 11-msec tests, one at atmospheric pressure and the other at 2500 psi. The 2500 psi test reaches a power maximum of about 930 Mw, about 30% higher than the atmospheric peak. The time of peak is delayed about 10 msec and the burst is noticeably broader for the high pressure test. The atmospheric pressure test displays a more rapid power drop following the peak and the power recovers rapidly toward an equilibrium power, whereas for the high-pressure test, the power continues to decrease throughout the post-burst region of the test.

The plate surface temperature for the 2500 psi test shows further indication of the poor heat transfer after the temperature reaches the critical point. The plate surface remains substantially above the 354°C saturation temperature for a relatively long time after the power level has decreased by a factor of one hundred, whereas for the atmospheric pressure test, the temperature drops below saturation soon after the heat source is removed. That this change in behavior is sudden, and associated with the high temperatures reached, is illustrated more clearly in Fig. 11, where the same 2500 psi test is compared with a 1900 psi test on a more expanded time scale. In this figure the plate

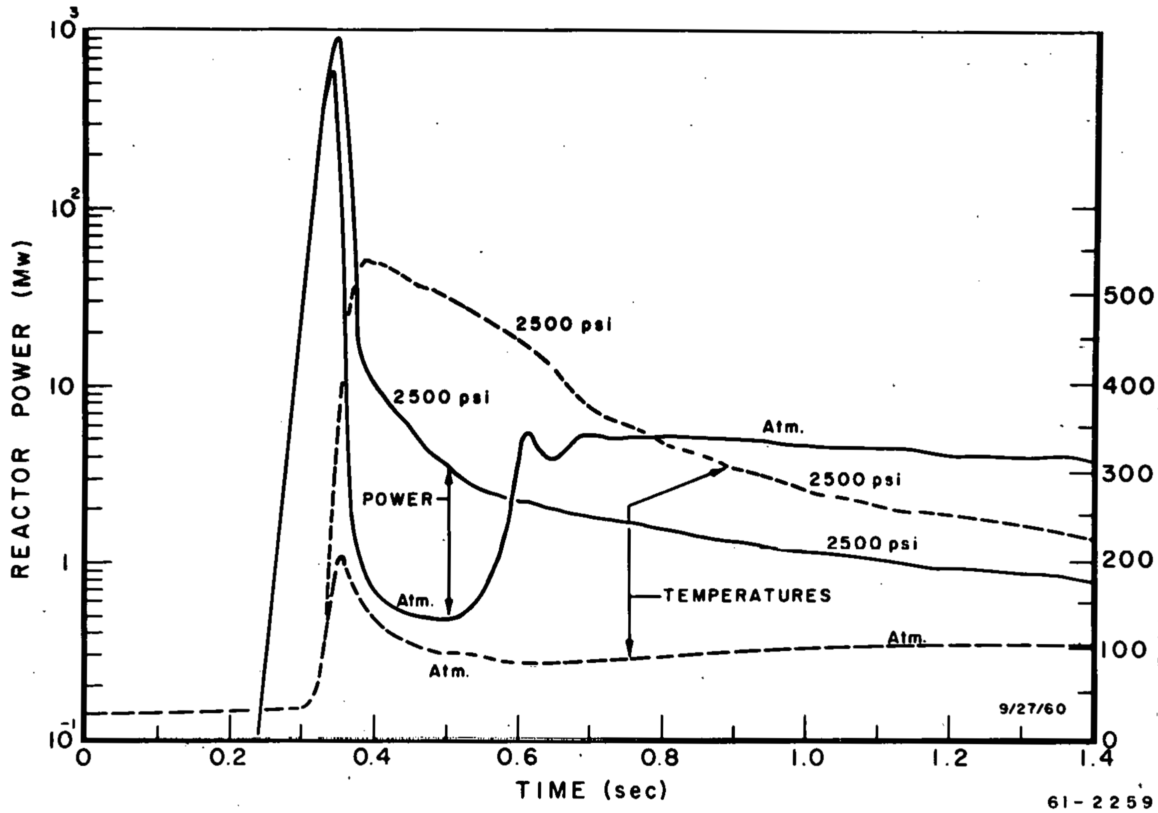

Fig. 10 - Reactor Power and Fuel Plate Surface Temperature for Tests at Atmospheric Pressure and at 2500 psi. (Period ~ 11 msec)

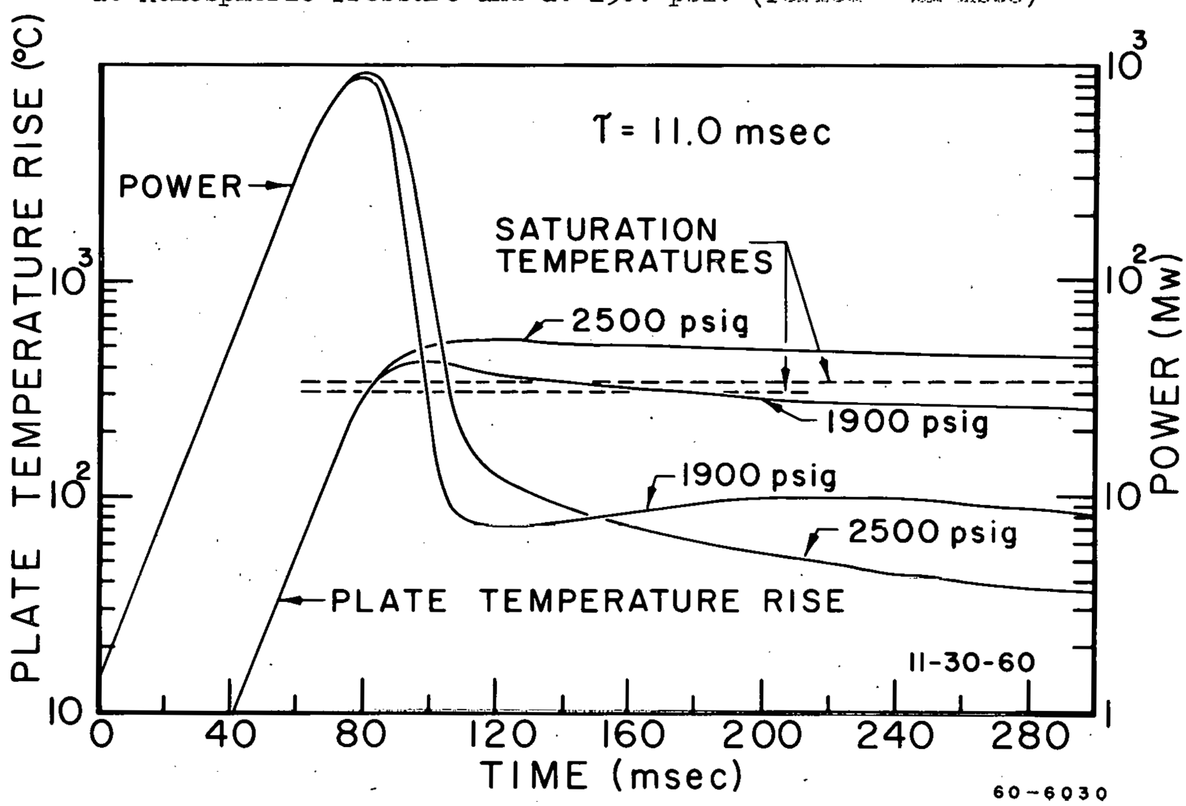

Fig. 11 - Reactor Power and Fuel Plate Surface Temperature for Tests at 1900 psi and at 2500 psi. (Period ~ 11 msec)

surface temperature rise has been plotted on a logarithmic scale in order to facilitate shape comparisons. The temperature for the 1900 psi test peaks sharply soon after saturation temperature is reached and the plate surface remains above boiling temperature for only about 85 msec. For the 2500 psi test, on the other hand, the plate surface temperature trace has a broad top and remains above saturation for about 400 msec, even though the power has dropped to the same or lower level.

The results of the Spert III tests have shown that as the system pressure is raised the delay and eventual suppression of boiling results in an increase in the temperature rise of the fuel plates, with a consequent increase in the reactivity compensation arising from water and fuel-plate expansion, and with only small changes in the power burst behavior. For room-temperature power excursions at 2500 psi with initial asymptotic periods of about 11 msec, boiling is suppressed until after peak power, but the energy release is increased only about a factor of two as compared with atmospheric pressure tests. Detailed calculations are presently being made of the reactivity compensation arising from fuel-plate and water expansion based on measured temperatures, the power history, and the measured reactivity coefficients. The results of these calculations will be compared with the reactivity compensation at the time of the power maximum obtained by analysis of the power burst shape.

Additional data on the transient pressures developed in these tests are still being reduced and correlated.

B. The Effect of Forced Coolant Flow on Room-Temperature Power Excursions

The presence of forced coolant flow during a power excursion will, by its effect on heat transfer properties, influence the partition of energy between the fuel plates and the moderator and will also, by increasing the rate of energy removal from the core region, decrease the effective power coefficient of reactivity. Preliminary data have previously been reported(7) from tests performed in Spert III to determine the effect of coolant flow on power excursion behavior. Some additional data are now available and can be reported, although the correlation and analysis are still in a preliminary stage.

The tests were performed for a range of water velocities from 0 to 18 ft/sec in the reactor core and for initial asymptotic reactor periods from 500 msec to 10 msec. All of the tests were initiated at room temperature (~30°C) from a power level of about 10 w. The system pressure was 230 psi for all tests that will be discussed here, except as specifically noted.

As previously reported,(7) the addition of flow velocities as high as 18 ft/sec does not produce significant changes in the initial power burst for excursions with periods shorter than about 50 msec,

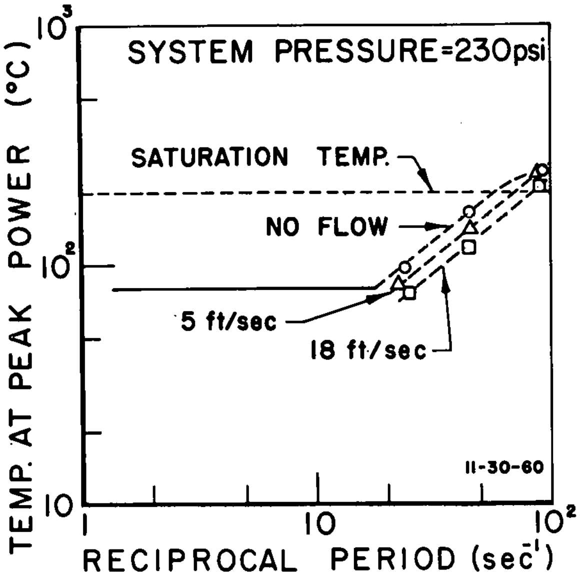

Fig. 12 - $\Theta(t_m)$ vs Reciprocal Period for Various Coolant Flow Rates

whereas for longer period tests, the flow velocities are sufficient to eliminate the occurrence of an initial power peak. In both cases the equilibrium power level following the burst is approximately proportional to the flow rate, as would be expected. In no case was the test allowed to run long enough for the cooling water to make a complete cycle through the heat exchangers and back to the core, so that for these tests the quasi-equilibrium power reached is independent of the heat removal capacity of the heat exchanger.

Fig. 12 shows the temperature reached by the hottest measured fuel plate surface (outside plate of a central fuel assembly) at the time of the power maximum, $\Theta(t_m)$, as a function of the initial reciprocal period for various flow rates. $\Theta(t_m)$ is approximately constant for periods longer than about 100 msec and increases approximately linearly with reciprocal period for shorter periods. The increased flow rates result in decreased temperatures as would be expected. In the short-period tests, $\Theta(t_m)$ increases at about the same rate for all flow rates toward a maximum of about 230°C, which is slightly above saturation temperature for the system pressure of 230 psi. For periods of about 20 msec, $\Theta(t_m)$ is well below boiling. Increasing the flow rate decreases $\Theta(t_m)$ even though the peak power remains essentially the same. Since at 18 ft/sec the coolant moves less than five inches in a time of one 20-msec period, it is obvious that a study of the change in vertical temperature profiles is necessary to an understanding of the reactivity compensation for these tests. Such a study is being made.

One test was performed in which a power excursion with an initial period of about 9.7 msec was initiated from room temperature with a system pressure of 2500 psi and a flow rate of 18 ft/sec. In Fig. 13 the power and the fuel plate surface temperature for this test are shown and compared with similar data for a test at atmospheric pressure with no forced flow. For the high-pressure, high-flow test a peak power of about 1000 Mw is reached, followed by a sharp minimum. After a few damped oscillations the power levels off at about 130 Mw. It is of interest to note that the maximum fuel plate surface temperature for this test is approximately the same as for the 11 msec, 2500 psi, no-flow test shown in Fig. 10 of the preceding section, but in this case there is a sharp decrease in temperature following the peak, indicating that the forced coolant flow is sufficient to offset the large decrease in heat transfer rate near the critical point, although the temperature maximum is not appreciably reduced.

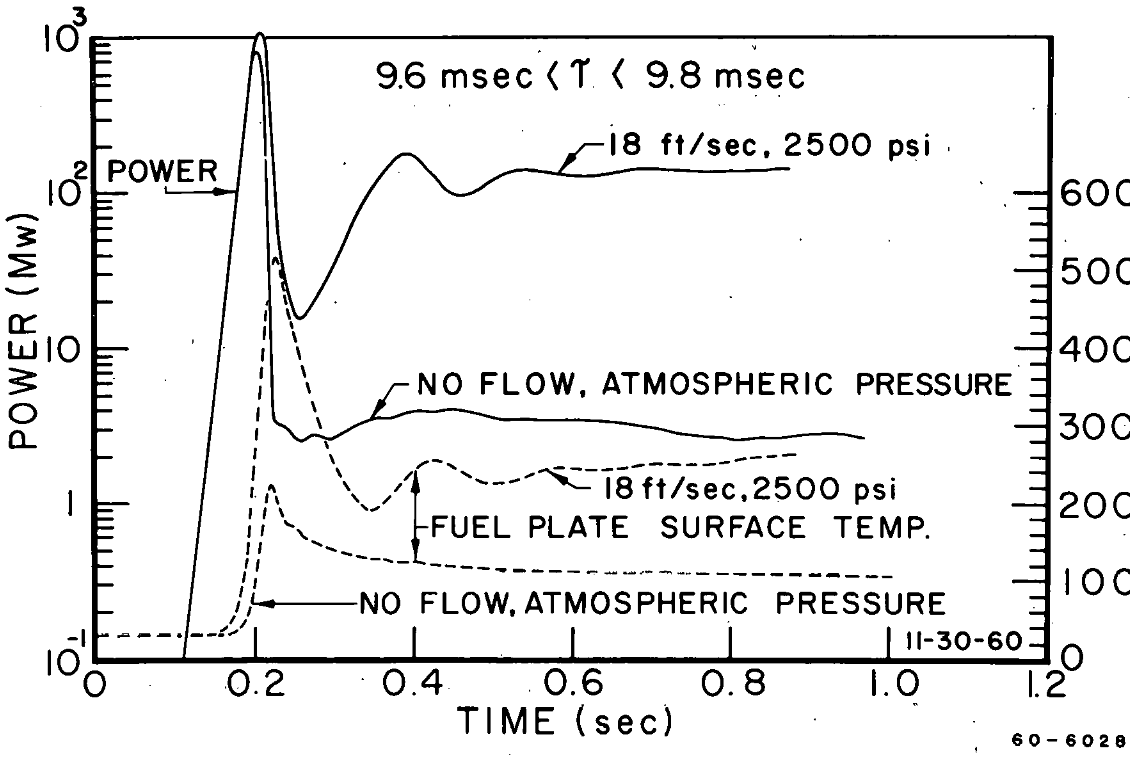

Fig. 13 - Reactor Power and Fuel Plate Surface Temperature for Tests at 18 ft/sec, 2500 psi and at No-Flow, Atmospheric Pressure (Period ~ 9.7 msec)

Detailed examination and correlation of the results of these tests are continuing.

C. Fuel Assembly Damage

During venting operations following one of the tests investigating the effect of system pressure on power excursion behavior,[6] unusually high air activity was observed in the Spert III reactor building. Subsequent analysis of the reactor water revealed the presence of low levels of short-lived fission products. A visual examination of the fuel assemblies was made and blisters were observed on the outside fuel plates of three fuel assemblies. These blisters were located in an area from three to twelve inches from the bottom of the fuel plates, which corresponds to the region of maximum neutron flux for the Spert III core when operated at room temperature. The blistered assemblies had been located in regions of the core corresponding to substantially different flux levels, and assemblies in symmetrically located core positions were not blistered, nor were adjacent plates of the same assemblies.

Rippling of fuel plates was generally observed throughout the core and was more severe than had been observed subsequent to previous short-period tests at atmospheric pressure. Dial indicator measurements taken on the outside plates of several of the assemblies showed that, in addition to the rippling, a general bowing of the fuel plates in the vertical direction had occurred. This bowing was as much as 0.04 inches in the lower one-third of the fuel plate. Peak-to-valley measurements of the ripples showed an average height of about 0.03 inches. The combined bowing and rippling was in some cases sufficient to close up more than half of the water channel. There was further evidence that at least one of the plates had bowed and rippled sufficiently to touch the adjacent control rod guide boxes, thus locally closing off the water channel. Visual inspection of the inner plates indicated similar rippling throughout the assemblies.

The rippling and bowing of stainless-steel, enriched UO_2 fuel plates had previously been observed with the Spert I P-core(8,9) and is thought to be the direct result of high thermal gradients during tests with periods less than about 20 msec.

Blisters had also been observed with the P-core. In that case, the results of metallurgical examination suggested that the blistering may have been due to steam explosions originating at cladding cracks caused by intergranular corrosion(9). Preliminary metallurgical examination of the Spert III type "C" blistered plates does reveal the presence of a limited number of intergranular cracks. None appears to have penetrated the cladding, however, and the cracks may have been formed as a consequence of the blistering. The cause of the blistering of the Spert III plates is at present indeterminate, although there appears to be some correlation between fuel segregation (stringers) at the fuel-clad interface and the formation of blisters in the high-temperature, high-flux region of the fuel plates(10).

After removal of the blistered fuel assemblies, power excursion tests were continued in Spert III to complete the study of pressure and flow effects at short periods. No further fission product release has been observed and visual inspection has failed to disclose any additional blistering.

A. Flow Distribution Measurements for Type-C Assemblies

The channel flow distribution for a Spert III, type-C, 1-S fuel assembly has been obtained. The work was done at the ETR hydraulic test facility using a dummy fuel assembly. The flow distribution was obtained by inserting two static tubes in each channel. The channel frictional pressure drop was recorded as a function of assembly flow and converted to channel flow by means of the Fanning correlation. The channel flow distribution is shown in Fig. 14 for various flow rates. The flow distribution is relatively flat except for the flow depression of the outside channels. For an average assembly flow of 400 gpm, which corresponds to a total flow rate of 20,000 gpm in the Spert III reactor, the percent standard deviation from the mean is 2.1%, and the maximum deviation from the mean is 19.6%.

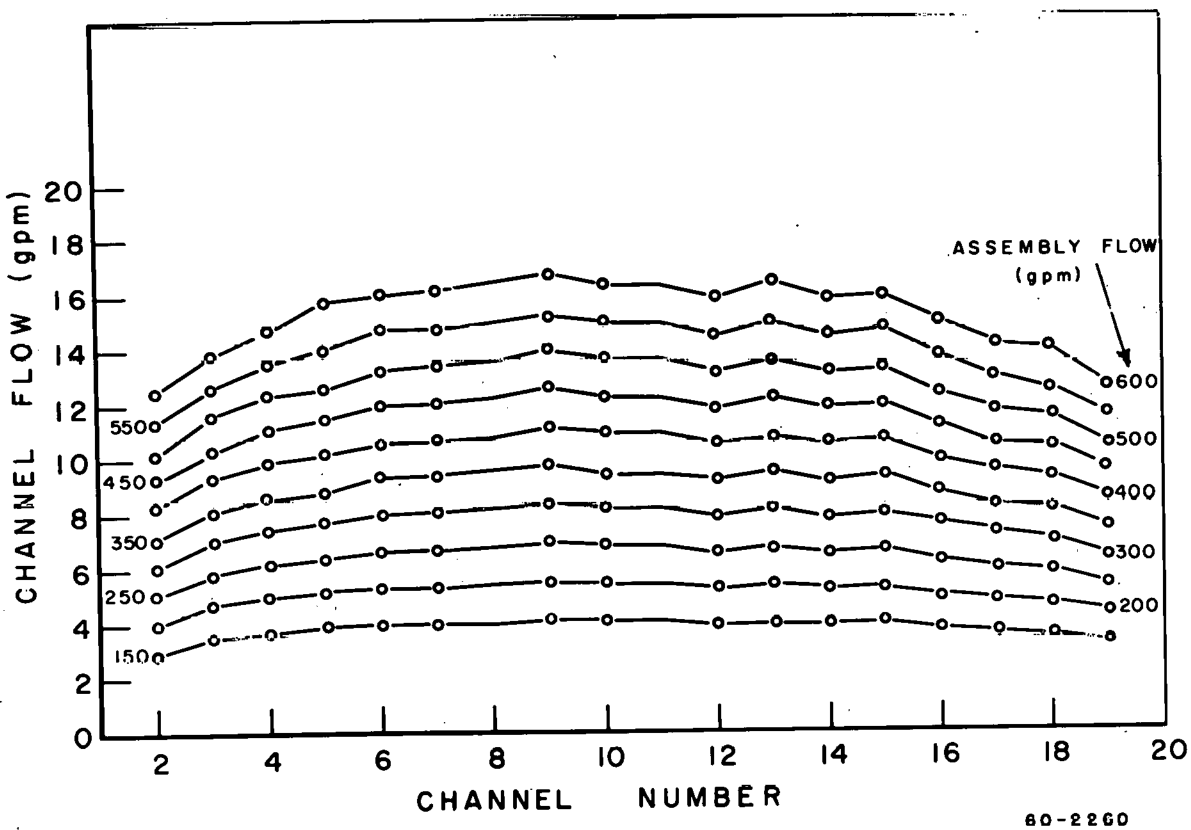

Fig. 14 - Channel Flow Distribution for Type-C Fuel Assembly

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