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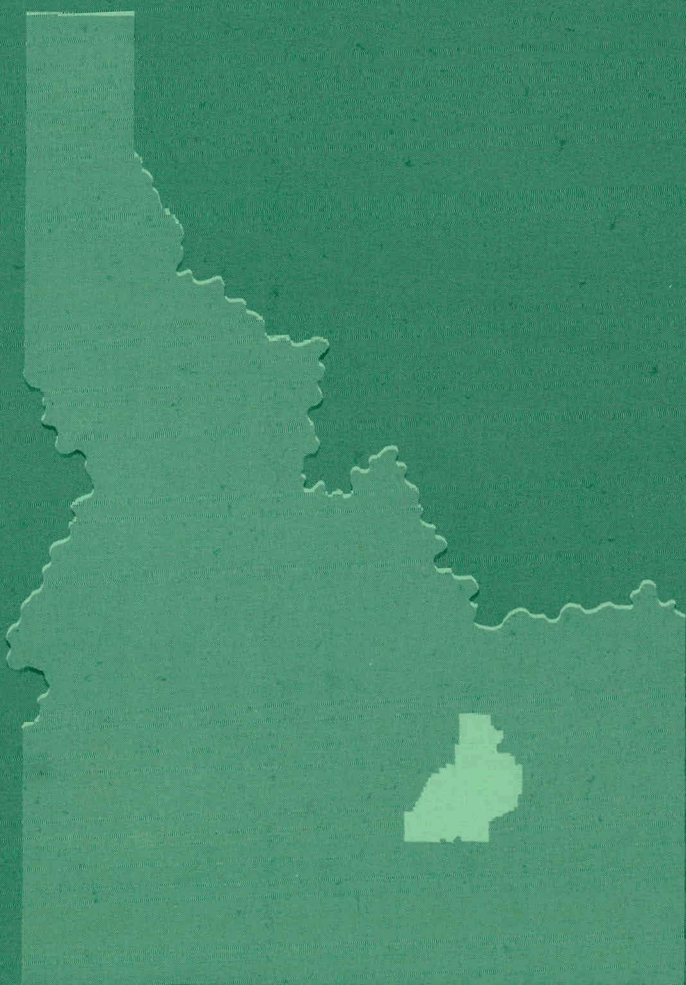
QUARTERLY TECHNICAL REPORT
SPERT PROJECT
July, August, September, 1960

F. Schroeder, Ed.



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NATIONAL REACTOR TESTING STATION
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SPERT

SPECIAL POWER EXCURSION REACTOR TESTS

QUARTERLY TECHNICAL REPORT
SPERT PROJECT
JULY, AUGUST, SEPTEMBER, 1960

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SPERT I - A series of in-pile experiments has been initiated in the Spert I reactor to study the dynamics of void growth in a water-filled capsule containing fuel plates. The objective of these experiments is to obtain information concerning the rate of transient boiling in the self-shutdown of water-moderated reactors. The capsule, which is essentially a water-filled tube in which fuel plates are mounted, is placed in a centrally located flux trap in the P-core. Using the reactor as a neutron source, measurements are made of the instantaneous pressure, volume expansion, and fuel plate surface temperatures. In a preliminary series of tests which was performed with non-fuel-bearing dummy plates in the capsule, no moderator displacement was observed which could be attributed to radiolytic gas formation in the capsule as a result of radiation emanating from the reactor region surrounding the capsule. The approximate range of reactor periods for these tests was 10 to 900 msec. The first series of tests with fuel-bearing plates in the capsule included tests with asymptotic reactor periods of 19, 14 and 9 msec, for selected capsule pressures in the range between 0 and 1600 psig. Reduction and analysis of the data obtained from these tests has been initiated and preparations are under way for the installation of a second set of more heavily loaded fuel plates and for photographic study of transient bubble formation in the capsule.

Examination of the results of the large amplitude, low power, oscillator tests previously performed in Spert I has indicated that the describing-function phase results are in relatively good agreement with theoretical predictions. The magnitude results, however, show a significant frequency-dependent deviation from the theory. Several possible effects are under consideration for the cause of the disagreement.

SPERT II - Critical experiments have been performed for a close-packed, D_2O-moderated core configuration. The final configuration comprised 72, 12-plate, modified D-type fuel assemblies and 8, 8-plate, fuel-poison control rods. This loading contained 6.36 kg of U^{235} and had an available cold, clean excess reactivity of approximately \$6.50.

Critical experiments and measurements of nuclear parameters were made for an expanded D_2O-moderated core in which the 3-in. x 3-in. fuel assemblies are spaced on 6-in. centers. The control rods for this core configuration were tubular, with a cadmium poison section and an aluminum follower. The operational core loading comprised 24, 22-plate, BD-type fuel assemblies with a total mass of 3.70 kg of U^{235}. This core had an available excess reactivity of approximately \$8 at ambient temperature. The measured temperature coefficient varies from -2.0¢/°F at 80°F to -4.4¢/°F at 300°F, and the temperature defect from 70°F to 300°F was found to be about \$7.25.

SPERT III - Tests have been performed in the Spert III reactor to determine the effect of coolant flow on the response of the reactor to step-changes in reactivity. The tests covered a range of water velocities

from 0 to 18 ft/sec in the reactor core for initial asymptotic periods of from 500 msec to 10 msec. All of the tests were initiated at room temperature with system pressures of 230 psi or 2500 psi. For short-period excursions no significant changes are observed in the initial power burst, but the post-burst equilibrium power is approximately proportional to flow rate. For power excursions with initial periods greater than about 100 msec, 18 ft/sec flow velocity is sufficient to eliminate the occurrence of an initial power peak. The power rises monotonically toward an equilibrium power level which increases with flow rate. Data from these tests are still being reduced.

ENGINEERING - The first use of D_2O in the Spert II primary system is described. During the initial critical studies performed in Spert II this quarter, approximately 294 lb of D_2O was lost as a result of evaporation from the vessel, small spills, and evaporation of droplets and films clinging to fuel assemblies, handling tools and other parts removed from the reactor tank. An additional 74 lb is estimated to have been lost from deuterization of the ion-exchange columns in the D_2O cleanup system.

Additional burnout heat flux calculations have been made for the Spert III reactor including the effects of two-phase flow.

Hydraulic tests have been continued for the type-D fuel assemblies to determine the channel flow distribution for both up-flow and down-flow conditions. The results indicate that for a Spert IV core composed of 20 18-plate, D-type assemblies, and for a flow rate of 250 gpm per assembly, the maximum deviations from the average channel flow would be 14% and 6% for up-flow and down-flow respectively.

A study has been made to investigate the feasibility of soluble poison emergency shutdown systems for the Spert II and Spert III reactors. The proposed systems would be designed to provide additional reactivity shutdown margin in the event of an extreme control rod malfunction during elevated temperature operation in order to permit slow cool-down of the primary coolant system. The study indicated that gadolinium and samarium nitrate solutions decompose into insoluble compounds at temperatures of about 300°F. It is concluded that boric acid injection by means of the plant makeup pumps would provide a satisfactory means of poisoning the systems by an amount equal to the available cold excess reactivity in 5.5 hours and 3 hours for Spert II and Spert III respectively.

II. SPERT I

A. Mechanism Studies

1. Summary of Experimental Work

A series of in-pile experiments has been initiated in the Spert I reactor to study the dynamics of void growth in a water-filled capsule containing fuel plates. The objective of these experiments is to obtain information concerning the role of transient boiling in the self-shutdown of water-moderated reactors. Preparation of the capsule with its associated instrumentation was completed during this quarter. Foil activation flux measurements were made to determine the neutron flux distribution throughout the capsule fuel plate region and to determine the ratio of peak flux in the capsule fuel region to the peak flux in the reactor core. In addition, flux-shaping experiments were carried out to increase this ratio and to enhance the peaking of the capsule fuel flux in the vicinity of the window through which photographs are obtained. A preliminary series of transient tests was then performed, with aluminum plates substituted for the capsule fuel plates, in order to measure any volume changes resulting from radiolytic gas formation in the capsule due to radiation originating from the reactor region surrounding the capsule. The first series of tests with fuel-bearing plates in the capsule was also completed during this period. This series included tests with asymptotic reactor periods of approximately 19, 14 and 9 msec, for selected capsule static pressures in the range between 0 and 1600 psig. No motion pictures were taken during this first test series.

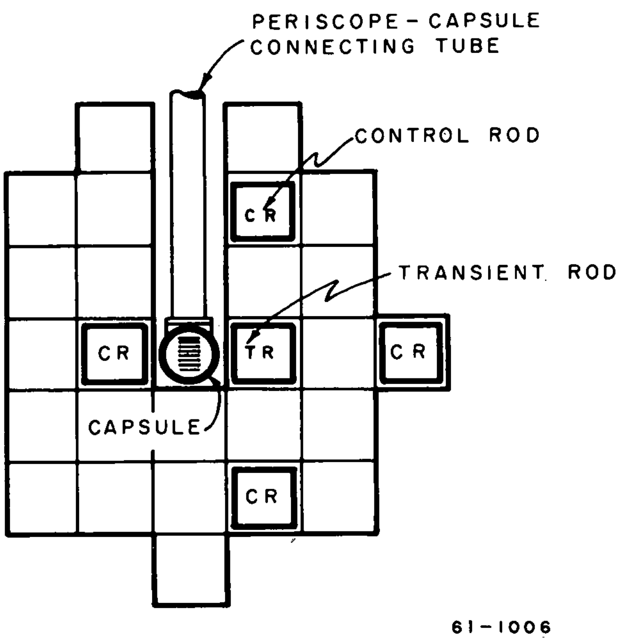

Fig. 1 - Cross Section of P-Core Showing Flux Trap Slot and Capsule

2. Capsule Description

The capsule used for in-pile void growth studies in Spert I was placed in a centrally located flux trap in the stainless steel clad, highly enriched P-core, which functioned as the neutron driving source for the capsule transient experiments. The flux trap slot was arranged to accommodate both the capsule and the associated periscope connecting tube. A cross section of the core is shown in Fig. 1.

The capsule unit itself, as shown in the schematic drawing in Fig. 2, is basically a closed, water-filled, cylindrical steel tube, with a piston at one end to provide a measure of the expulsion of the water moderator from the capsule fuel plate region during the course of a power excursion. The fuel plates are mounted in the capsule chamber and are instrumented with thermocouples to measure the temperature of the cladding surface. Below the fuel plates, at the bottom end of the capsule chamber, is a differential pressure transducer, which measures the difference between the instantaneous pressure developed during the power excursion and the initially adjusted static capsule pressure. This measurement of the pressure rise in the capsule requires that the static pressure in the bottom chamber below the pressure transducer be initially set equal to the capsule static pressure. A sapphire window and a periscope-camera arrangement permit photographs to be taken of bubble formation during a power transient.

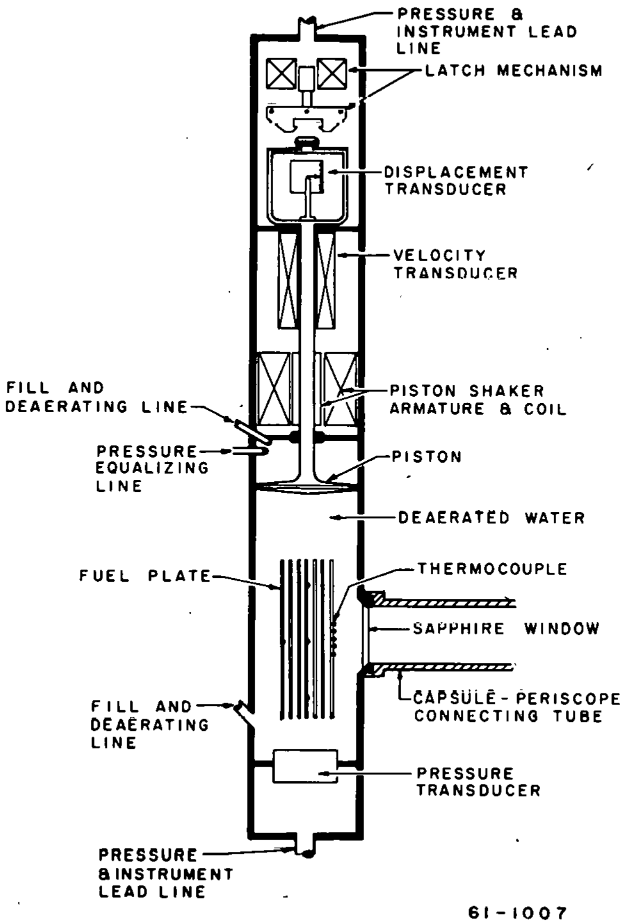

Fig. 2 - Schematic Drawing of the Capsule

During a power excursion, the piston is displaced upward by the moderator expulsion resulting from the combined effects of fuel plate expansion, steam formation, and radiolytic gas formation in the capsule. The speed and the distance moved by the piston are measured by velocity and displacement transducers mounted in the upper instrumentation section of the capsule unit. Measurement of the instantaneous displacement of the piston and the transient pressure requires elimination of those errors arising as a result of the compliance of the capsule water due to the entrained gases normally present in fresh, deionized water. Prior to each transient run, the capsule water is deaerated by alternately pressurizing the capsule water for several seconds at about +50 psig and then depressurizing for several seconds at -15 psig. After each cycle, a measure of the relative "stiffness" of the water is obtained by measuring the amplitude of a 60 cycle acoustical wave set up by the piston shaker armature located in the instrumentation section (see Fig. 2). As the water becomes deaerated, the magnitude of the propagated acoustical wave measured by the pressure transducer at the bottom of the capsule chamber increases to a maximum value. This point, which is reached after a few pressurization cycles, corresponds to the maximum deaeration of the capsule water for the existing experimental conditions.

3. Capsule Tests with Simulated Fuel Plates

A series of twelve power excursions was run using pure aluminum plates to simulate fuel plates in the capsule. The purpose of these tests was to determine if any moderator displacement attributable to radiolytic gas formation in the capsule chamber could be observed during power excursions as a result of beta, X-ray and fast-neutron radiation emanating from the reactor region surrounding the capsule. The tests were run at ambient temperature and at 0 psig capsule static pressure, and the approximate range of reactor periods covered in these tests was from 10 to 900 msec. The experimental results obtained indicated no observable void growth during or after the power bursts. The minimum detectable piston displacement is approximately 0.003 in., corresponding to a void volume of 0.004 in.^{3} or about 0.01% of the total water volume in the capsule chamber.

4. Capsule Tests with Initial Capsule Fuel Loading

The initial fuel loading in the capsule consisted of a set of seven aluminum-clad fuel plates, each 1 in. x 5 in. x 0.050 in. in dimension, with fuel plate cladding thickness of 0.015 in. and a fuel density of 0.038 g U^{235}/cm^2 uniformly distributed over the entire plate area. The capsule fuel plates were cut from a larger fuel plate, and the resulting open edges of the cut plates were then sprayed with aluminum to provide a complete cladding for the capsule plates. Tests were run at reactor periods of approximately 19, 14 and 9 msec, with various capsule static pressures between 0 and 1600 psig. All transients were initiated from ambient temperature. Measurements of piston displacement and velocity, of fuel plate surface temperature, and of capsule pressure were obtained as a function of time during the transient. Typical time plots of piston displacement, plate temperature and capsule pressure are shown in Fig. 3.

Of interest in the curve of fuel plate surface temperature shown in Fig. 3, is the abrupt decrease in slope, which takes place following a period in which the temperature is rising exponentially. This break in the curve is thought to reflect the sudden increase in heat transfer rate due to the onset of boiling. This manifestation of boiling in the curve of fuel plate temperature has not been observed in previous measurements at Spert due to the poorer time-response properties of the thermocouples used in the earlier measurements. The newly developed thermocouples used in the capsule studies were 0.0005-in. diameter, peened and welded chromel-alumel couples, in contrast to the previously used 0.005-in. diameter thermocouples. As a result, the figure of merit

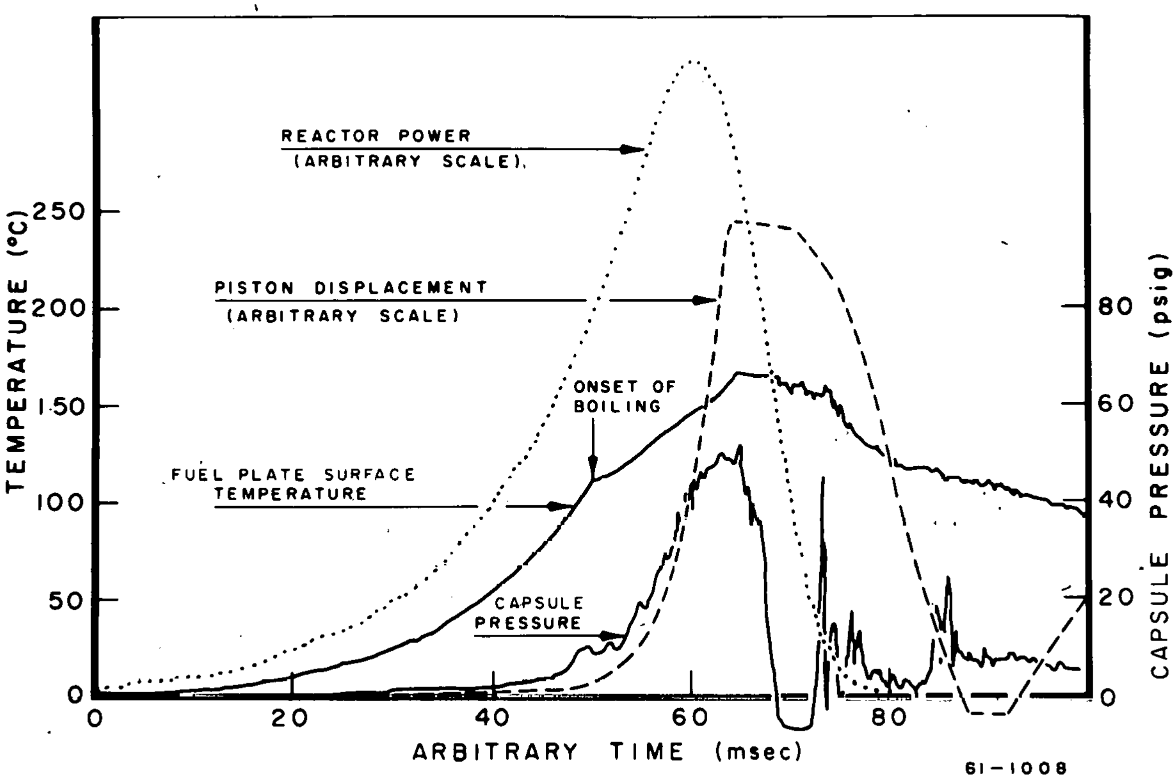

Fig. 3 - Typical Data from Capsule

ratio of contact area to heat capacity for the capsule thermocouples was an order of magnitude larger than for the other thermocouples, permitting an attendant increase in thermocouple time response, with consequent greater resolution in the transient behavior of the fuel plate temperature.

Reduction and analysis of the data obtained from these tests has been initiated, and preparations are under way for the installation of the second set of more heavily loaded capsule fuel plates and for photographic study of transient bubble formation in the capsule.

B. Reactivity Oscillator Program

Detailed examination of the results of the large-amplitude, low power oscillator test of June, 1960 [1] has indicated that the describing-function phase results are in relatively good agreement with theoretical predictions based on the analytical formulation of Akcasu[2]. The describing-function magnitude results, however, show a significant, frequency-dependent deviation from this theory. If the magnitude data are normalized at the higher frequencies, agreement is good down to about 0.5 cps. Below this frequency, the experimental results exceed the theoretical by a difference which grows roughly inversely with frequency to a difference of about one reciprocal dollar at about 0.002 cps. Similar results are obtained for three different values of reactivity amplitude ranging from \$0.037 to \$0.156.

These tests were carried out using a simple, direct-recording instrumentation system, and a detailed re-examination of this system has not revealed any effects which might account for the discrepancy in the describing-function magnitude results. The earlier tests in the summer of 1959, using a quite different, nulling-wave-analyzer type of instrumentation, led to results of substantially the same character. It has been concluded therefore that, barring a remarkable coincidence, the disagreement between the experimental and theoretical describing-function magnitude results are not due to faults in the part of the instrumentation which follows the ion chamber neutron detector.

Other possibilities under consideration for the cause of the disagreement are as follows: (a) errors in the analysis or in the calculations based on the analysis, (b) faults in the operation of the ion chamber, (c) faults in the operation of the reactivity oscillator, and (d) additional spatial or neutron propagation effects not included in the analysis.

Errors in the analysis of the problem as formulated seem unlikely; the analysis has been independently checked in detail. Errors in the computer program calculation also seem unlikely; a hand calculation at a single frequency and reactivity amplitude value gave results identical with the results of the computer calculation. Hand calculation at another frequency and reactivity amplitude will be undertaken to further corroborate this point.

A series of tests on the performance characteristics of the ion chamber used in the June, 1960 oscillator tests has been carried out. These include a detailed static calibration of the chamber in the MTR thermal column (saturation and linearity characteristics tests) and a comparison of the chamber dynamic performance (in reactivity oscillation tests at the Treat reactor) with that of another chamber which has given satisfactory results at Treat. The results of the tests at the MTR indicate excellent static characteristics for the chamber under the conditions used in the Spert oscillator tests. The results of the Treat tests are still being processed. In addition, because the chamber is known to be microphonic as a result of vibration of the parallel plate structure, some consideration is being given to the effect which plate vibration induced by the oscillator drive system might have on the signal generated by the chamber.

With regard to possible faults in the operation of the reactivity oscillator, no reasonable mechanism has been envisioned which could yield sufficiently large increases in reactivity amplitude with decreasing frequency to account for the observed disagreement in describing-function magnitude. The results of early preliminary experiments in the oscillator design stage of the program indicate that the largest conceivable vibration or wobble of the oscillator would yield a change in reactivity amplitude which would be at least a decade smaller than that required.

Some consideration is being given to the possibility that peculiar spatial or neutron-propagation effects may be involved due to location of the oscillator and detector on opposite edges of the reactor core. No such spatial effects are explicitly included in Akcasu's theoretical formulation of the problem.

III. SPERT II

A. Close-Packed, D_2O-Moderated Cores

Power excursion tests will be performed in Spert II using a close-packed, D_2O-moderated core comprising highly enriched uranium-aluminum, plate-type, fuel assemblies in order to compare the kinetic behavior of such a core with that of an H_2O-moderated core. The comparison should yield direct information on the effects of prompt neutron lifetime on reactor self-shutdown behavior. The test results themselves will also have direct applicability to reactors of this type which are in operation or being constructed.

With the objective of achieving a heavy-water-moderated core of closely similar geometry to previous Spert light-water-moderated cores, several configurations of fuel assemblies in the core and fuel plate spacings within the fuel assemblies have been investigated, some of which proved to be unacceptable. A core comprised of modified D-type fuel assemblies[3] with 12 fuel plates per assembly was chosen for further testing.

The light-water-moderated cores previously investigated in the Spert II reactor facility have used the B-type fuel assembly[1], but this assembly was found to be unsuitable for use in heavy water because of the braze material and the stainless steel end-box. These materials caused a sizeable reactivity loss due to parasitic neutron capture. For this reason a BD-type fuel assembly has been used in the heavy water cores. The BD-type assembly, which is compared with a B-type assembly in Fig. 4, has an aluminum end-box and

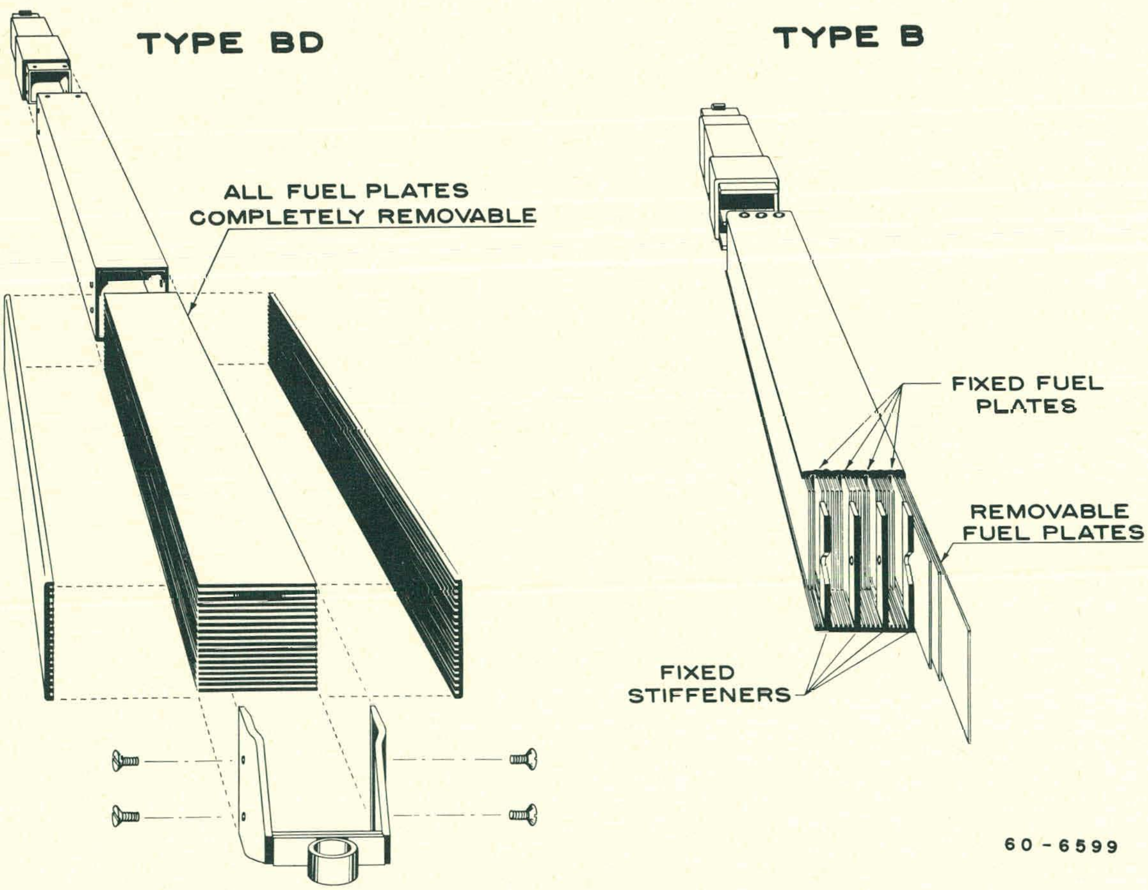

Fig. 4 - Comparison of Type-BD and Type-B Fuel Assemblies

no braze material. An insufficient number of these assemblies was available for use at the time of these experiments, so the D-type fuel assembly, which is identical with the BD-type assembly except for the end-box, was modified to resemble the BD-type assembly. This was done by placing a spacer in the bottom of the fuel can so that the fuel was raised to the same elevation above the lower grid as it would have been had the BD-type fuel assemblies been used.

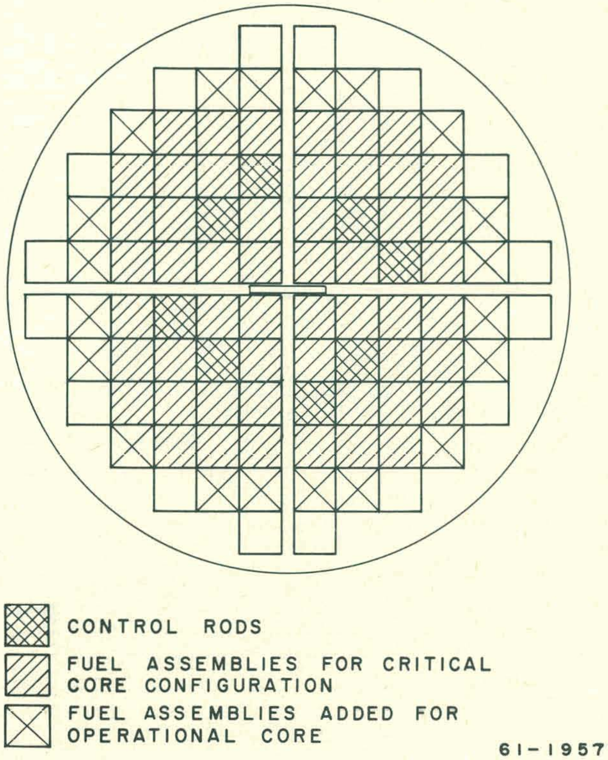

Fig. 5 - Initial Critical and Operational Core Configurations for Close-Packed D_2O Core

The reactor was critical in D_2O with 52 modified D fuel assemblies containing 12 fuel plates each, and with the 8, 8-plate fuel poison control rods fully withdrawn. The core contained 4.68 kg of U^{235}. Additional fuel was then loaded to obtain an operational core loading. The final core configuration comprised 72, 12-plate fuel assemblies and 8, 8-plate fuel-poison control rods. This loading contained 6.36 kg of U^{235} and had an available excess reactivity of approximately \$6.50. The critical and final core configurations are shown in Fig. 5.

B. Expanded D_2O-Moderated Core

The most commonly used core configuration for highly enriched uranium-aluminum plate-type fuel assemblies in heavy water is one in which the fuel assemblies are spaced apart from one another with a sizeable water gap between assemblies (e.g., MITR, DIDO, etc.). A core of this type was obtained in Spert II by loading BD-type fuel assemblies into alternate grid positions so that they were on six-inch center spacing. Each fuel assembly contained 22 fuel plates. The control rods used for this core were tubular with a cadmium poison section and an aluminum follower piece.

The reactor was critical with the loading of 20 fuel assemblies, shown in Fig. 6, containing 3.08 kg of U^{235} in the core. The control rods were withdrawn 20 inches of a possible 24 inches.

The operational core loading, shown in Fig. 7, was obtained by loading four additional fuel assemblies for a total of 3.70 kg of U^{235} in 24 fuel assemblies, and the overall temperature coefficient for uniform heating of the core, moderator and reflector was measured at a system pressure of 260 psig. As shown in Fig. 8, the temperature coefficient varys from -2.0¢/°F at 80°F to -4.4¢/°F at 300°F. The temperature defect due to increasing the reactor temperature from 70°F to 300°F was found to be about $7.25.

The temperature defect was used as a reactivity shim in order to obtain a preliminary calibration of the control rod worth over an extended range. At each temperature the differential rod worth was determined near the critical position by the period method. The control rods, as a bank, were found to be worth from $1.15/in. at 13.00 in. withdrawn to $0.51/in. at 22.50 in. withdrawn. This rod worth-information, together with the rod-worth information obtained during the approach to the operational loading, indicates that the operational core, shown in Fig. 7, contains an excess reactivity of approximately $8 at ambient temperature and atmospheric pressure.

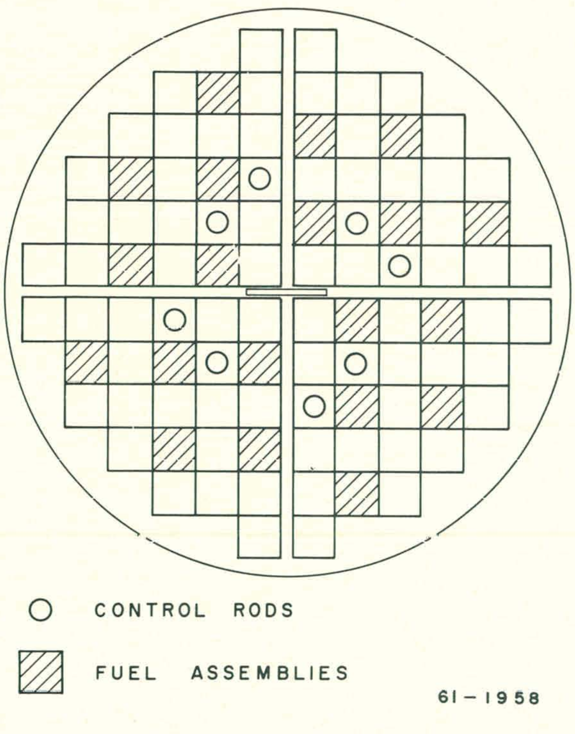

Fig. 6 - Initial Critical Configuration for Expanded D_2O Core

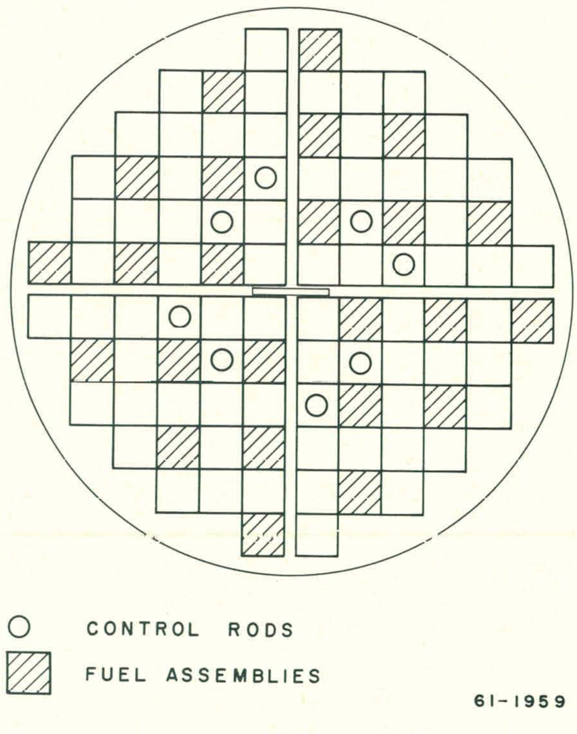

Fig. 7 - Operational Core Configuration for Expanded D_2O Core

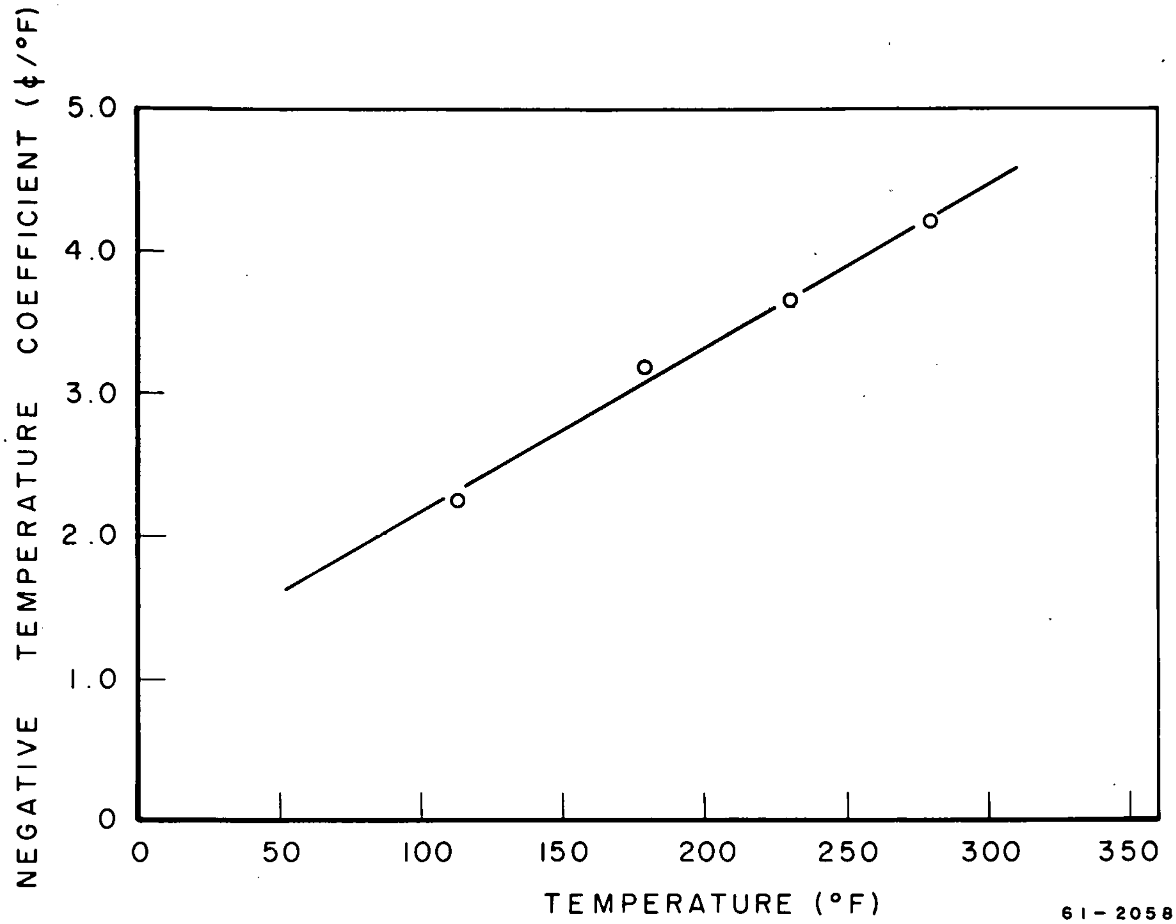

Fig. 8 - Temperature Coefficient for Spert II Expanded D_2O Core

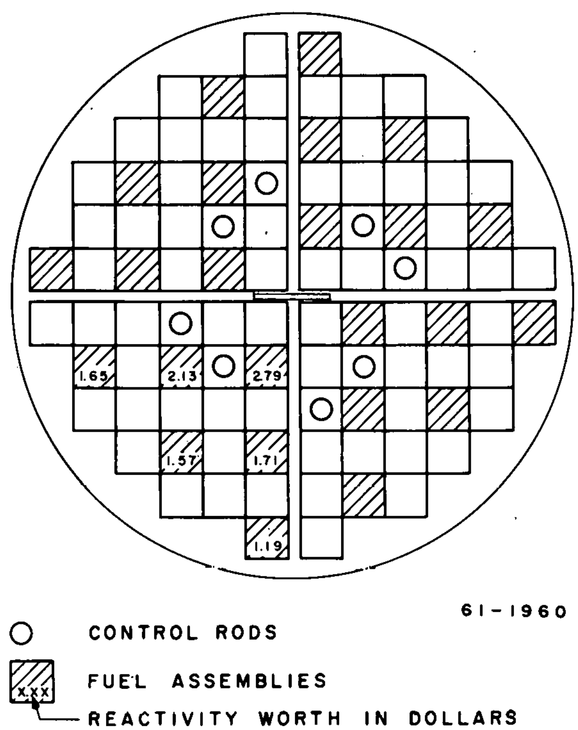

Fig. 9 - Reactivity Worth of Fuel Assemblies in Expanded D_2O Core

The reactivity worth of the fuel assemblies in each position of the D_2O-moderated core, shown in Fig. 7, was measured by determining the excess reactivity of the core when all of the assemblies were in position and then with one assembly removed. The difference in the reactivity for the two cases is taken as the reactivity worth of the fuel assembly in question. The reactivity worth of fuel assemblies in the west quadrant of the core was determined. Since the core has polar axial symmetry, these values are applicable to symmetrical positions in the remaining quadrants. Fig. 9 shows the reactivity worth of each of the fuel elements as a function of their position in the core.

IV. SPERT III

A. Effect of Coolant Flow on Power Excursion Behavior

The self-limitation of power excursions in plate-type, water-moderated reactors is due in large part to the density changes resulting from thermal expansion of the fuel plates and the moderator. Forced coolant flow will, by its effect on heat transfer properties, influence the partition of energy between the plates and the moderator and will also, by increasing the rate of energy loss from the core region, decrease the effective power coefficient of reactivity. Tests were performed in the Spert III reactor during this quarter to determine the effect of coolant flow on the response of the reactor to step-changes in reactivity. These tests covered a range of water velocities from 0 to 18 ft/sec in the reactor core for initial asymptotic 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 for most of the tests was about 230 psi but several were performed at 2500 psi.

For power excursions with initial periods greater than about 100 msec, flow velocities up to 18 ft/sec are sufficient to eliminate the occurrence of an initial power peak. The reactor power rises monotonically toward an equilibrium level which increases with increasing flow rate. For excursions of a given period, greater than 100 msec, the temperature rise of the hottest measured fuel plate surface decreases with increased flow velocity. Fig. 10 shows the effect of flow velocity for tests with initial asymptotic periods of about 160 msec.

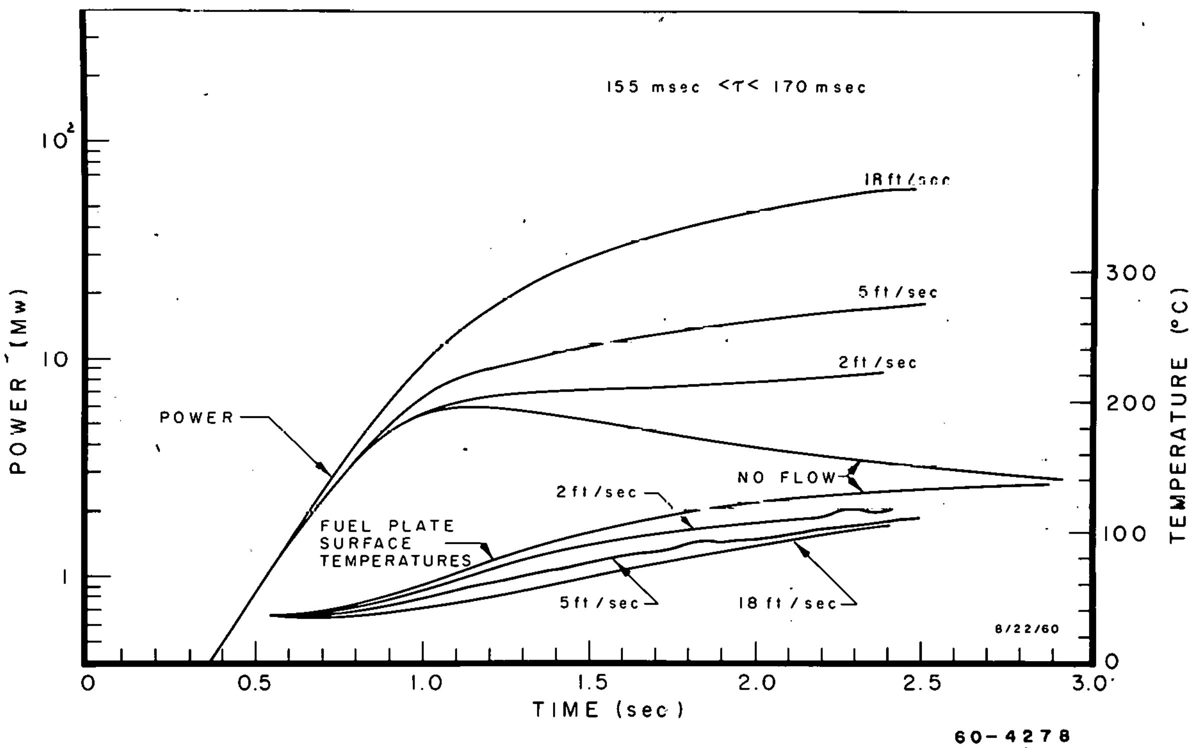

Fig. 10 - The Effect of Coolant Flow on Reactor Power and Fuel Plate Surface Temperature for 160 msec Tests at 230 psi

For power excursions with initial periods shorter than about 50 msec, it is observed that flow velocities as high as 18 ft/sec do not produce significant changes in the initial power maximum. The equilibrium power level following the burst is approximately proportional to the flow rate. As in the case of the longer period tests, the temperatures reached by the hottest measured fuel plate surfaces decrease with increasing flow velocity, even though the peak power for the burst remains essentially constant. It should be noted, in this regard, that at 18 ft/sec flow velocity the heated coolant has moved only a few inches in a time equal to one period and that an understanding of the role of heated water in the self-shutdown for such excursions will require an examination of the water temperature distribution in the core.

Fig. 11 shows the effect of flow velocity for tests with periods of approximately 20 msec, and for system pressures of 230 and 2500 psi. Since the power behavior is essentially unchanged by this pressure variation, only two power curves are shown. The addition of the high flow rate reduces the maximum measured temperature for the 2500 psi test by about a factor of two and tends to offset the effect of the elevated pressure on the maximum temperatures reached in no-flow tests, as witnessed by the similarity of the 230 psi and 2500 psi temperature traces for 18 ft/sec flow velocity.

The data from the flow test series are still being reduced and the above conclusions should be regarded as preliminary.

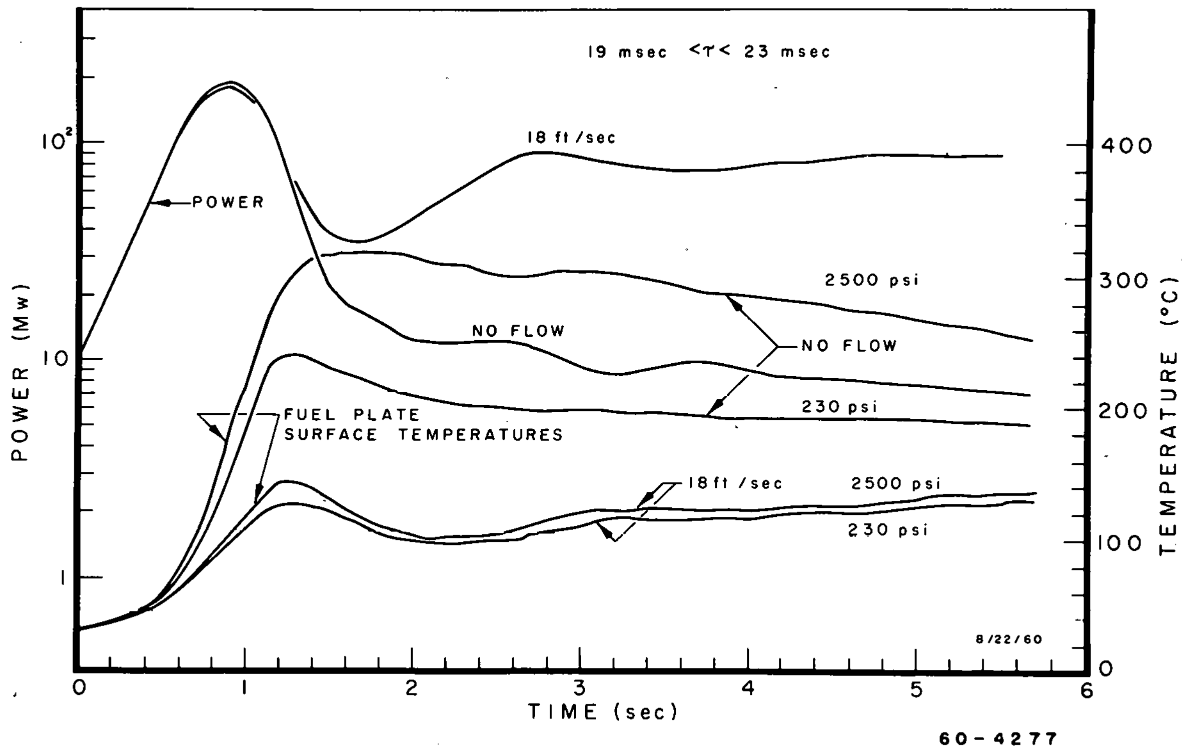

Fig. 11 - The Effect of Coolant Flow on Reactor Power and Fuel Plate Surface Temperature for 20 msec Tests at 230 and 2500 psi

V. ENGINEERING

A. D_2O Handling Experience in Spert II

In preparation for the critical studies described in Section III of this report, it was necessary to prepare the Spert II plant for its first operation with D_2O. Since the primary circulating system was not to be used for these tests, the reactor vessel was isolated from the remainder of the system to minimize the system inventory and possible D_2O contamination and losses. All light water piping was isolated by removing spool pieces and installing blind flanges.

The reactor tank, clean-up piping system, outdoor heat exchanger and its associated piping, and the D_2O transfer pump piping were vacuum dried to remove any residual light water and/or vapor remaining in the systems. Wherever possible, visual inspections were made to determine dryness of the various systems. The piping was rinsed with 780 gal of D_2O which was then removed and stored in stainless steel drums for future use as a rinse. There was no detectable change in the isotopic purity of this D_2O before and after the rinse. On July 28, 6400 gal of heavy water were transferred from the D_2O storage tank to the reactor tank. The loading of fuel was initiated on August 1.

A dry helium gas atmosphere is maintained over the surface of the water in the heavy water storage tank at Spert II. The experimental studies required frequent transfer of material between the storage tank and the reactor tank. When transferring material back to the storage tank, excess gas was vented to the reactor tank. When material was transferred to the reactor tank, excess gas was vented through a cold trap to the atmosphere. It is estimated that the reactor tank was open to the atmosphere for 115 hours; this was occasioned by frequent insertions and removals of fuel assemblies, control rods, and mechanical parts. During this time a dry air purge, dew point $< - 30^{\circ}F$, was maintained over the surface of the heavy water to exclude atmospheric air. All parts to be removed from the reactor tank were allowed a short drip-dry period in the tank before actual removal. Despite this precaution, a considerable amount of material was removed in the form of droplets and films clinging to the parts. It was realized this material, being in intimate contact with the outside air, could be a source of contamination when the parts were returned to the reactor tank. To prevent any such contamination, all parts were dried with hot air before their return to the tank. Frequent density measurements made during the experimental program indicated little, if any, deterioration of D_2O quality. Routine checks of quality as given by electrical conductivity, pH or pD, and amounts of metallic ions present in the material were also made at regular intervals.

A loss of 538 lb of D_2O was incurred during this period; of this amount, $\pm$ 170 lb may be due to a 1% uncertainty in the initial heavy water inventory and does not necessarily represent a loss. In addition, a loss of 74 lb is estimated to be the result of deuterization of the ion-exchange columns in the clean-up system. The remaining 294 lb is attributed to evaporation of D_2O to the atmosphere, small spills, and direct evaporation of droplets and films clinging to fuel assemblies, handling tools and mechanical parts removed from the reactor tank.

B. Two-Phase Flow Effects on the Spert III Burnout Heat Flux

The Spert III reactor core is composed of a large number of parallel rectangular coolant channels with varying heat generation rates. Occurrence of two-phase phenomena in the hot channel restricts the coolant flow through that channel and thus reduces the burnout heat flux. Preliminary calculations of the burnout heat flux in the Spert III reactor core have shown that two-phase flow is present for some of the operating conditions(5). Calculations have been made to allow for these two-phase flow effects in the determination of the burnout heat flux. No attempt has been made to couple the two-phase phenomenon to the nuclear properties of the reactor core.

The pressure drop associated with the flow of a two-phase mixture has been evaluated using the Modified Martinelli-Nelson correlation(6). In this relationship the friction and momentum pressure drops are calculated according to the method of Martinelli and Nelson(7) with the friction pressure drop corrected for mass flow rate effects using the values obtained by Sher(8).

The incorporation of the two-phase flow effects into the burnout heat flux calculation has necessitated revision of the preliminary computer code(5) to the extent that the use of the IBM-650 computer is impractical and the revised code is being written for the IBM-704 machine. To assist in the compilation of the code, a sample calculation was performed by hand for a typical set of operating conditions in the Spert III reactor. This calculation was made for the type "C"-2S fuel assemblies in the center portion (hot spot) of the reactor core at system conditions of 2500 psia pressure and 500°F inlet water temperature. The flow area per channel of the 2S fuel assembly is 1.02×10^{-3} ft^2, and the equivalent diameter is 0.0192 ft.

For the sample calculation, friction pressure drop in the channel prior to the formation of steam was calculated using the Fanning friction equation:

$$\Delta P_o/L = \frac{f\, V^2 \rho}{2 g_c\, D_e} \tag{1}$$

where

$\Delta P_o/L$ = Pressure drop per unit length of channel

ρ = Water density

g_c = Conversion factor from pounds mass to pounds force

D_e = Equivalent diameter

V = Velocity

f = Friction factor

A roughness factor of $\epsilon/D_e = 0.00025$ was used to determine the friction factor.

The two-phase flow pressure drop was calculated by the Modified Martinelli-Nelson relationship(6).

$$\Delta P_{TP} = \Delta P_o \left(\frac{\Delta P_{TPF}}{\Delta P_o} \right) \frac{\Phi_L^{\ 2}}{(\Phi_L @ 10^6)} + r \frac{G^2}{g_c} \qquad (2)$$

where

ΔP_{TP} = Two-phase flow pressure drop

ΔP_o = Pressure drop in flow circuit for flow of saturated water

ΔP_{TPF} = Two-phase frictional pressure drop

$\Phi_L^{\ 2}$ = Ratio of two-phase frictional pressure drop gradient to corresponding isothermal liquid gradient

$\Phi_L^{\ 2} @ 10^6$ = Same as $\Phi_L^{\ 2}$, except evaluated at a mass flow rate of 1.0×10^6 lb/hr-ft^2

G = Mass flow rate

r = Multiplier of G^2/g_c to calculate momentum pressure loss

The ratio $(\Delta P_{TPF}/\Delta P_o)$ has been determined by Martinelli and Nelson as a function of exit quality and pressure(7). The ratio $(\Phi_L^{\ 2}/\Phi_L^{\ 2} @ 10^6)$ is the correction factor applied to the friction pressure loss to account for the effect of mass flow rate(8). The multiplier r is used to determine pressure drop due to momentum changes and is a function of the exit quality and system pressure(7).

The two-phase pressure drop equation assumes a linear relationship between the length of channel and the quality. However, for Spert III, the nonuniform heat generation along the channel required the friction pressure drop to be calculated for incremental channel lengths. Summation of the incremental pressure drops obtains the total channel loss. The momentum pressure drop was determined using only the exit quality.

Expansion and contraction losses for two-phase flow through the upper end-box region of the fuel element were calculated by the standard single-phase flow equation,

$$\Delta P = K \frac{V^2}{2 g_c} \ ; \qquad (3)$$

however, an apparent density of the two-phase fluid ρ Apparent = (ρ steam) (X quality) + (ρ water)(1-X) was used to calculate the linear velocity, V. K is an area dependent constant for expansion or contraction.

Fig. 12 presents the change in pressure drop per inch of channel length at the condition prior to burnout where bulk boiling occurred in the hot channel. The burnout conditions are those calculated by the preliminary IBM-650 burnout heat flux code for Spert III which were reported in Reference (5). Integration of the area under the curves gives the total pressure loss due to friction. The straight portion of the plot is for single-phase flow and the curved portion is for two-phase flow. The addition of the pressure loss due to momentum changes and entrance and exit losses results in the overall channel pressure drop. Table 1 is a compilation of the calculational results for the hot channel in the two-phase flow region.

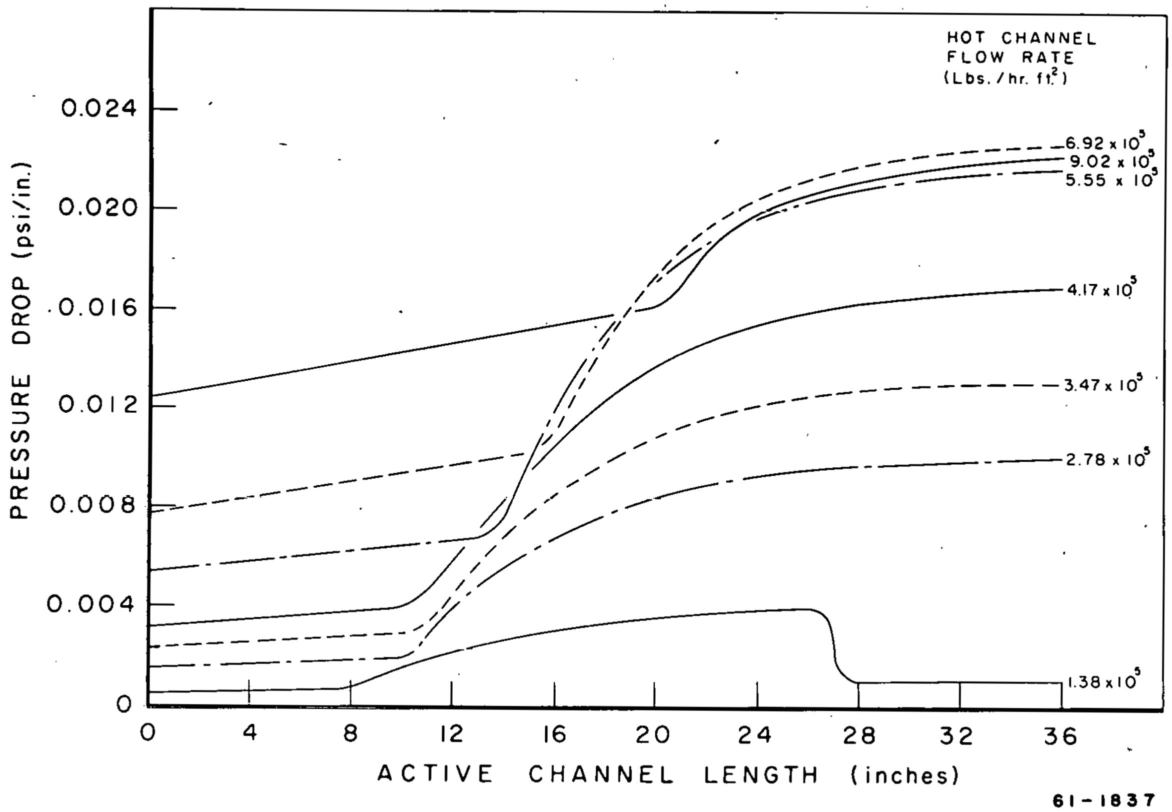

Fig. 12 - Pressure Drop Per Unit Channel Length vs Position for Various Flow Rates - Spert III

TABLE 1

HOT CHANNEL PLATE SECTION PRESSURE DROP

$550^{\circ}F$ INLET TEMPERATURE - 2500 PSIA SYSTEM PRESSURE

Hot Channel Flow Rate ($lb/hr\text{-}ft^2$)	Power (Mw)	Friction ΔP (psi)	Momentum ΔP (psi)	Plate Section Entrance and Exit ΔP (psi)	Total ΔP (psi)
1.38×10^5	25	0.079	0.0381	0.001	0.118
2.78×10^5	35	0.233	0.0655	0.003	0.302
3.47×10^5	40	0.309	0.0843	0.005	0.398
4.17×10^5	45	0.393	0.1058	0.007	0.506
5.55×10^5	50	0.508	0.1185	0.012	0.639
6.92×10^5	55	0.563	0.1275	0.017	0.709
9.02×10^5	60	0.638	0.1145	0.029	0.782

C. Hydraulic Test of the Spert Type-D Fuel Assembly

The initial hydraulic studies of the Spert type-D fuel assembly[3] revealed the need for a modification to the upper lifting bail to help flatten the flow distribution through the center plate section of the assembly. This modification was made and hydraulic tests were conducted on the 18-plate, type-D fuel assembly to determine the channel flow distribution for both up-flow and down-flow conditions. The tests were run in the ETR Hydraulic Test Facility which has previously been described[3]. The channel flow distributions were obtained by inserting two tubes in each channel with one tube measuring the static pressure near the bottom of the channel while the other tube measured the static pressure near the top of the channel. The channel frictional pressure drop was correlated to the flow by means of the Fanning friction correlation. In addition, the overall fuel assembly pressure drop was measured.

A plot of overall pressure drop vs flow is shown in Fig. 13 for both up-flow and down-flow at a water temperature of approximately 85°F. The relationship may be expressed by

$$\text{GPM} = 74.7\, \Delta P^{0.549} \qquad (4)$$

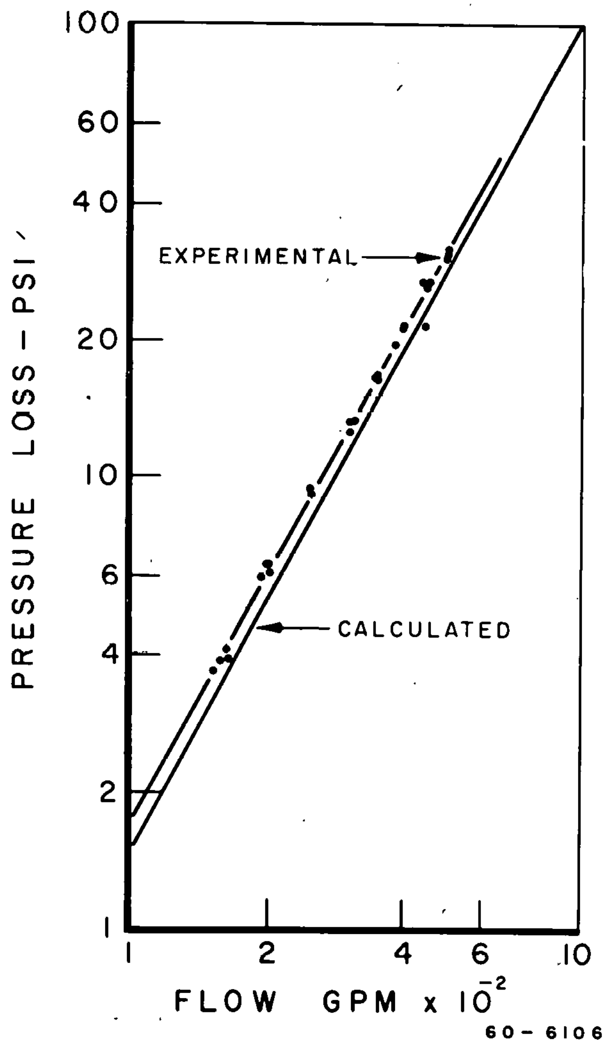

Fig. 13 - Overall Pressure Drop vs Flow for Type-D Fuel Assembly

where the pressure drop is in psi. The calculated relationship for up-flow is also shown for comparison purposes. The calculation deviates from the experimental correlation by approximately 10%. The overall pressure drop with the modified lifting bail was not appreciably different from the pressure drop with the original lifting bail.

In Figs. 14 and 15 the channel flow distribution is shown as a function of overall flow rate and flow direction. The channel flow was measured for 11 of the 19 channels. The channel flow for the remaining 8 channels was determined by assuming symmetrical flow distribution. For down-flow, Fig. 14, the distribution is relatively flat except for a flow depression in the outer two channels. For up-flow, Fig. 15, the flow distribution is relatively flat except for the increased flow in the outside channels. The flow direction dependence in the outside channel flow rate is probably due to entrance effects.

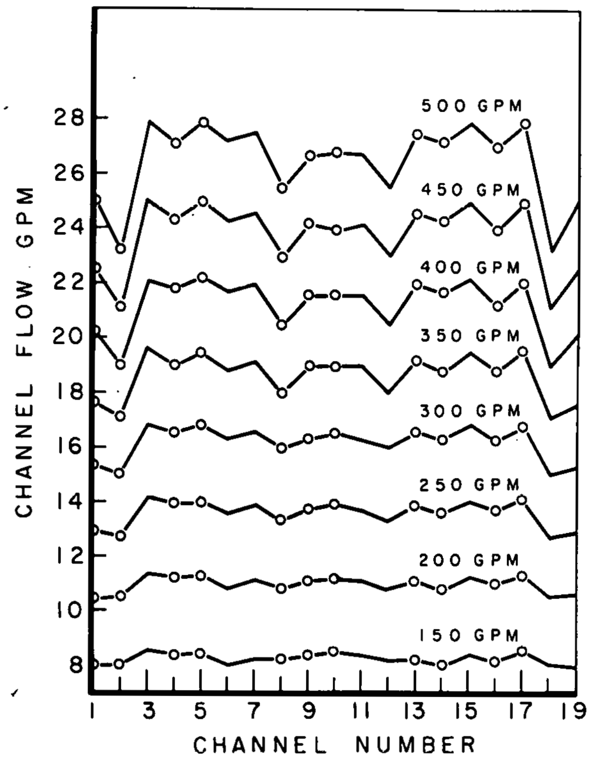

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Fig. 14 - Channel Flow Distribution as a Function of Overall Flow for Down-Flow - Type-D Assembly

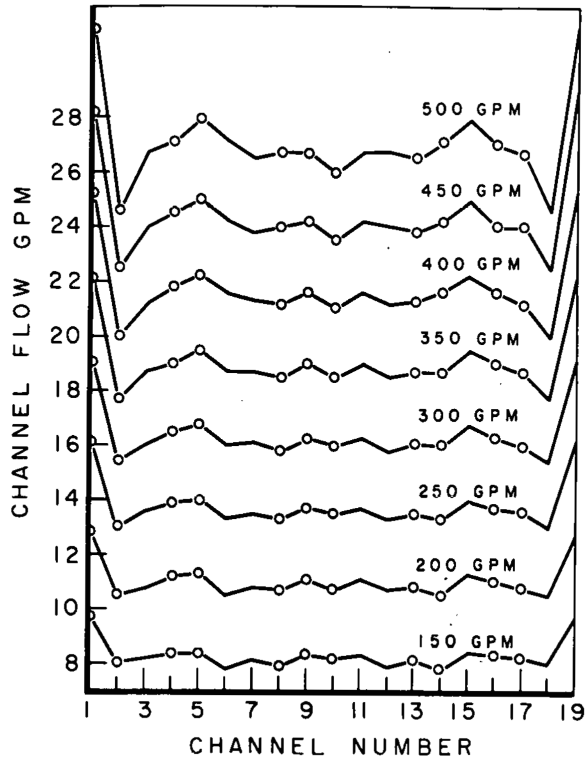

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Fig. 15 - Channel Flow Distribution as a Function of Overall Flow for Up-Flow - Type-D Assembly

The actual flow in any given channel may deviate from that shown in the figures since it was assumed that each channel spacing was the average width between plates (0.092 in.) rather than the maximum (0.105 in.) or the minimum (0.079 in.). However, motion pictures taken of the fuel plates during previous flow studies showed that all of the plates exhibited random vibration in their support slots.

For up-flow, the most erratic channel flow distribution occurred at the highest flow rate that was measured, 500 gpm. At this flow rate, the maximum deviation from the average was 16% while the standard deviation from the average was 1.4%. The maximum deviation occurred in the outside channels.

For down-flow, the maximum deviation and standard deviation from the average channel flow were 12% and 1.3%, respectively, for an assembly flow rate of 500 gpm. In general, the maximum deviation and the standard deviation decreased with decreased assembly flow rates.

For a Spert IV core composed of 20, 18-plate, type-D fuel assemblies, the fuel assembly flow rate would be approximately 250 gpm per assembly. At this flow rate, the maximum and standard deviation from the average would be 14% and 1.5% for up-flow 6% and 0.8% for down-flow.

D. Studies of Soluble Poison Injection Systems for Sperts II and III

1. Introduction

A study has been made to investigate the feasibility of soluble poison emergency shutdown systems for the Spert II and Spert III reactors. At present, the occurrence of a severe malfunction of the control rods during high temperature operation might require removal of moderator to achieve a safe cold shutdown margin before cooling down the system. This action would necessarily be done in a strictly controlled fashion in order to prevent rapid cooling of the system and to avoid possible boiling instabilities but, in any event, might result in severe thermal stresses and damage to the primary coolant system. The injection of soluble poison to the hot system would permit a slower cool-down of the system. The poison injection system investigated is not intended to provide fast nuclear shutdown or to be used as a routine means of obtaining additional shutdown margin.

2. Selection of Soluble Poison

Of the elements with high thermal neutron absorption cross sections, boron, cadmium, europium, gadolinium and samarium are commercially available in the form of water-soluble salts. Cadmium is unsuitable for use in the Sperts II and III stainless steel systems because it has a high potential for plating out, as shown by its position in the electromotive series relative to stainless steel. Europium was eliminated as a soluble poison material because of higher cost (~ 20 times greater than Gd or Sm) and lower neutron absorption cross section. The salts of gadolinium, samarium and boron have been investigated in detail.

Of the soluble salts of gadolinium and samarium that were considered, only the nitrates are compatible with the reactor systems. Gadolinium nitrate and samarium nitrate are available in purities ranging from 80% to 99.9%. Because of economic considerations, only 90% gadolinium nitrate and 80% samarium nitrate were considered. No information could be found in the literature on the high temperature properties of gadolinium and samarium nitrate. Since the neutron absorption cross section and room temperature solubility of these salts were very favorable, a small amount of gadolinium nitrate was obtained and simulated environmental studies were conducted in an autoclave. These studies showed the material to be unsuitable for use in either the Spert II or Spert III systems since it decomposed into an insoluble compound at a temperature of about 300°F. It was assumed that samarium nitrate would undoubtedly decompose because of its chemical similarity, and would also be unsuitable.

Boron in the form of boric acid was found to be acceptable.

3. Method of Poison Addition

Three methods for the addition of poison solutions to the systems were investigated. The three methods are described below:

Method A: With the primary pumps operating, the pressure drop of the system can be used to force part of the primary coolant through a poison tank containing an excess amount of saturated solution of the poison material, thus mixing the poison solution into the primary coolant.

Method B: A saturated solution of poison material can be added continuously with the plant make-up pump and the excess water removed through the blow-down valve until the desired concentration of poison in the primary coolant is reached.

Method C: A saturated solution of poison material can be injected into the reactor vessel by using high pressure inert gas.

For Spert II, Method "A" is undesirable since the primary pumps are isolated from the reactor vessel for some reactor operations at elevated temperature. Method "C" can be used and has a shorter poison injection time than for Method "B" but has the disadvantage of a larger initial installation cost.

For Spert III, Method "C" is not desirable since commercially supplied inert gas is not readily available at the higher system pressures at which Spert III will be operated. Method "A" can be used and has the shorter injection time, but the initial installation cost is estimated to be at least a factor of ten greater than for Method "B".

4. Conclusion

Since the speed with which shutdown can be accomplished is not of primary concern, the results of the study indicate that, of those systems investigated, boric acid injection by the plant make-up pump provides the best poison shutdown system for both the Spert II and Spert III reactors. Calculations indicate that the Spert II reactor can be poisoned sufficiently to remove \$12 in reactivity in approximately 5.5 hours by this method. For Spert III, \$23 can be removed in approximately 3 hours. These reactivity values represent the largest available excess reactivities anticipated for Sperts II and III.

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