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W. E. Nyer, Director

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SPERT PROGRESS REPORT
July, August, September, 1957

Edited by G.O. Bright

I. SUMMARY

Spert - I - The experimental program using the "A" core has been completed, and work is in progress on the "B" core. A brief resume of "A" core experiments is presented and the results of some of the static measurements with the "B" core are shown. It was found that four usable cores are possible with the removable-plate "B" fuel assemblies: 12 fuel plates/assembly; 12 fuel plates plus 12 dummy plates/assembly; 16 fuel plates/assembly; and 24 fuel plates/assembly. The measurements which have been made to date have been with a 24 fuel plate/assembly, 32 assembly core, and include rod worth, flux distribution, temperature coefficient and some preliminary void coefficient studies.

Spert - II - The construction contract was let and construction started in August. Most of the non-engineering work which is the responsibility of the Reactor Projects Branch is in the early preliminary stage.

Spert - III - The control system circuit drawings are essentially complete. Construction of some of the control system components, principally wiring, has begun.

Data Reduction and Interpretation - The progress which has been made in reducing the large raw data backlog from the Spert I "A" core experiments is shown. A report which will display all presently available data from the Spert - I subcooled transients is in preparation. Work is continuing on the calculation of reactivity behavior during a transient, with the principal area of investigation being that of the boiling or slightly subcooled experiments. The use of other calculational procedures as a check on the reactivity behavior work is being investigated.

Theoretical - The critical masses for the various possible "B" cores have been calculated, and agree quite well with the experimentally determined values. Some machine calculations of the temperature and

void effect with the 32 assembly core have been made, but no direct comparison with empirical values is yet possible. Two theoretical formulations, one a refinement of the Fuchs' model for reactor self-shutdown which seeks to explain the initial reactor transient behavior, and one which endeavors to show that the shutdown mechanism for reactor periods longer than 50 milliseconds can be ascribed to moderator density changes, have been evolved. Work is continuing on the determination of better nuclear constants for the Spert - III core.

Instrumentation - The primary activities of the Instrumentation section have been directed toward completion of all modification, maintenance, installation, and calibration of instrumentation necessary for performing static and transient tests on the Spert - I "B" core.

Secondary activities were as follows: investigating performance characteristics of linear and driver amplifiers with respect to stability and response; lowering the hum and noise threshold on driver amplifiers and thermocouple channels; measuring thermocouple response times; measuring pressure transducer characteristics; further developing boron-lined chambers; modifying thermocouple amplifiers; building a current step-function generator; studying transistor amplifier characteristics; studying the use of ultra-sonics as a measuring device; and continuing the Spert - III wiring.

Engineering - Various modifications have been made of the Spert - I reactor, including a new mechanism for clamping the fuel assembly in the core. Corrosion studies on aluminum have been made for the Spert - I and Spert - II cores. A program is underway to determine a suitable control rod construction for Spert - II. Flow tests have been made on Spert - II and Spert - III fuel assemblies. A complete description of the Spert - III control rod test facility is given. The results of the tests which have been made in this facility show that the present design of the Spert - III control rods and drives is satisfactory. An engineering completion schedule is given for both Spert - II and Spert - III.

II. SPERT - I

A. Recapitulation of "A" Core Experiments (G. O. Bright)

In June, 1957, the experimental part of Phase I of the Spert program, which consisted of work done with the "A" core in the Spert - I reactor, was completed. Phase II, which was started immediately after completion of Phase I, is concerned principally with experiments involving the "B" core. This report and those hereafter will be primarily annotating the "B" core work; therefore, a brief recapitulation of the experimental work which was done with the "A" core is presented for purposes of information.

Altogether, 606 power excursion experiments were performed, along with an estimated 885 auxiliary criticality determinations in the one year and ten months of operation. Of this time, about two months of operation were diverted to service work at the request of the Atomic Energy Commission, and about four months to modifications and repair of the reactor and auxiliaries.

An intensive program of static measurements was pursued, which resulted in a fairly complete set of empirical constants for the reactor system. These included rod worth, temperature coefficient, void coefficient and flux distribution studies.

The power excursion experiments were of three general types: step reactivity additions; ramp reactivity additions; and stability studies. The majority of the experiments were of the step reactivity addition variety. Reactor behavior was studied as a function of the parameters temperature and excess reactivity added to the system. The temperature range was from ambient (about 75°C sub-cooled) to boiling (about 97°C). The excess reactivity addition range was such as to result in asymptotic reactor periods of from 10 seconds down to about 5 msec. Preliminary work was done on determining the effect of varying the static water head over the core, changing the physical properties of the moderator by means of additives, and changing the hydrodynamic properties of the core by partially blocking some of the moderator channels.

Ramp rate reactivity addition experiments (the "startup accident" simulation) were conducted with starting power, temperature and reactivity addition rate as variables. Starting power was varied from 100 μ w to 200 kw, temperature from ambient to boiling and reactivity addition rate from 0.011% δ k/sec to 0.36% δ k/sec.

In the course of the ramp rate studies, it was found that under certain conditions the reactor would break into spontaneous uncontrolled power oscillations. This phenomenon was investigated under conditions in which the temperature, excess reactivity held in the system, and the static water head over the core were varied. These studies were carried to the limits of what was considered would be nondestructive operation of the reactor.

Much of the data from these experiments has not been reduced to usable form. As the data become available, topical reports on appropriate sections of the work will be prepared. A complete list of reports issued to date is included in the Bibliography section of this report.

B. Static Measurements - "B" Core

1. Critical Experiments (R. Zahn)

A series of critical experiments was performed to determine critical mass and core configuration for a number of fuel plate loadings with the "B" - type assemblies. The different core types maintained an approximate cylindrical geometry, but had varying total fuel masses and metal/water and uranium/hydrogen ratios.

The Spert -I reactor has been described elsewhere⁽¹⁾. The "B" type assemblies are in some respects a departure from the "A" type assemblies which have been used before. They are of aluminum construction with a stainless-steel lower end-box which incorporates a hold-down locking mechanism. The fuel is fully-enriched uranium-aluminum alloy in the form of plates, 0.020-inches thick and clad with 0.020 inches of aluminum on each side. The overall plate dimensions are 0.060 inches thick, 24-5/8 inches long and 2-3/4 inches wide. The nominal fuel content of each plate is 7 g of U²³⁵.

Four of these plates are brazed into aluminum side plates so as to form an assembly 3 inches x 3 inches x 27-7/8 inches. Additional grooves are provided in the side plates to accommodate 20 plates which may be slid in or out of the assembly. This allows a variation of the fuel loading from 28 grams/assembly to 168 grams/assembly. The principal object of using this type of assembly is to allow the void coefficient to be changed by varying the core parameters. Uniform fuel plate spacings in one assembly may be obtained with 4 plates/assembly, 8 plates/assembly, 12 plates/assembly, and 24 plates/assembly. The use of dummy aluminum plates in unoccupied fuel plate positions makes possible a variation in the hydrodynamical properties of an assembly without altering the total fuel content. The final assembly form is shown in Figure (1).

TOP VIEW

SPERT "B" ASSEMBLY

FIGURE 1
(57-4753)

Critical mass experiments were conducted with the following assembly types: 8 fuel plates/assembly; 12 fuel plates/assembly; 24 fuel plates/assembly; 12 fuel plates plus 12 dummy aluminum plates/assembly; and 16 fuel plates/assembly. The fuel plate spacing in the assembly was uniform in all of the types with the exception of the 16 fuel plates/assembly type.

Prior to the critical experiments, the "B" assemblies and fuel plates were assayed to determine their fuel content relative to one another. The method used was one developed by Forbes⁽²⁾ and used routinely for determining the fuel content of the MTR fuel assemblies. The relative fuel contents of the assemblies and of the fuel plates varied not more than $\pm 4\%$.

The critical experiments were conducted in the usual manner for this type reactor. An antimony-beryllium neutron source was used. Neutron detection was by two BF_3 and two B^{10} -lined chambers placed approximately at the horizontal centerline of the reactor as shown in Figure (2). A Spert boron chamber (SBC) was used during the final criticality determinations when reactor powers of the order of 100 watts were attained. Source level-order counting rates were about 3 counts/minute. The water temperature was approximately 20°C.

Fuel was loaded in pre-determined increments and multiplication measurements were made for each loading with the control rods fully inserted and fully withdrawn. A continuous estimation of critical mass was maintained, with each succeeding fuel loading contingent on this figure.

The results of the critical experiments are presented in Table I with the final critical core configuration for each fuel assembly type shown in Figures 3, 4, 5, and 6. The critical mass numbers were estimated from critical control rod position data, as the number of elements in each case was not necessarily integral. The results indicate that these configurations range from under-moderated to over-moderated and therefore should provide both negative and positive void coefficients for the kinetics studies. The calculated critical mass values shown in the Table are discussed in Section VI.A, page 45 of this report.

PLAN VIEW OF SPERT-I REACTOR INSTRUMENT LOCATION DURING STARTUP

FIGURE 2
(57 4861)

CRITICAL CORE CONFIGURATION

SPERT "B" ASSEMBLIES
12 FUEL PLATES/ASSEMBLY

FIGURE 3
(57-4864)

CRITICAL CORE CONFIGURATION

SPERT "B" ASSEMBLIES
12 FUEL PLATES
+ 12 DUMMY PLATES/ASSEMBLY

FIGURE 4
(57 4862)

CRITICAL CORE CONFIGURATION

SPERT "B" ASSEMBLIES
24 FUEL PLATES/ASSEMBLY

FIGURE 5
(57 4855)

CRITICAL CORE CONFIGURATION

SPERT "B" ASSEMBLIES
16 FUEL PLATES/ASSEMBLY

FIGURE 6
(57-4859)

TABLE 1

CRITICAL MASS DETERMINATIONS - SPERT "B" CORE

<u>Fuel Assembly Type (FP/FA)</u>	<u>No. Assemblies in Critical Core</u>	<u>Estimated Critical Mass</u>	<u>Calculated Critical Mass</u>
8	Not critical with 79 (4.42 Kg)	- - - - -	9.36 Kg
12	51	4.27 Kg	3.6 Kg
16	32	3.58 Kg	3.43 Kg
24	26	4.34 Kg	4.79 Kg
12 + 12 Dummy	76	6.38 Kg	6.75 Kg

2. Control Rod Worth (R. Zahn)

Upon completion of the critical experiments, the 24 fuel plates/assembly core was selected as the first configuration with which static measurements would be performed. Six assemblies were added to the critical assembly to form a symmetrical configuration which would have an estimated 4% excess reactivity (Figure 7). The reactor was then critical at ambient temperature with the control rods withdrawn about half way.

The control rods were calibrated as a group in terms of excess reactivity as a function of withdrawal above their critical position. This calibration forms the basis for many of the reactivity measurements which are made as a part of other static tests, in addition to acting as a guide in the performance of transient power excursion experiments.

The calibration was made by the period method. In this technique, the control rods are withdrawn above the critical position until a definitely increasing reactor power is noted. The period of this increase is measured and the excess reactivity in the system is determined by use of the Inhour relationship. In this particular experiment, the power was allowed to rise at least by a factor of 100, to insure that equilibrium with the delayed neutrons was reached. The reactor periods which were achieved ranged from about 10 to 100 seconds.

In order to permit withdrawal of the control rods without achieving a highly supercritical condition, the reactor was shimmed with a uniformly distributed poison; in this case boric acid. This was added incrementally, with period measurements being made at each step by the previously described technique, until the reactor was

CORE CONFIGURATION FOR CONTROL ROD CALIBRATION

32 ASSEMBLIES – 24 FP/FA

FIGURE 7
(57-4866)

just critical with the control rods completely withdrawn.

Results of the experiment are shown in Figure 8. The β values which were used in the Inhour formulation were those of Keepin and Wimett (3).

3. Transient Rod Worth (W. Neal)

The transient rod was intercalibrated with the control rods at a moderator temperature of 230°C. The method of calibrating the transient rod was as follows: first, the position of the control rods when the reactor was just critical was found, then the transient rod was withdrawn, thus increasing the poison in the core, in 2 inch steps and the new position of the control rods at critical was found after each step.

The position of the control rods at critical was found using a boron-lined ion chamber (SBC-4) in conjunction with a linear power recorder. The boron-lined ion chamber was placed in an instrument tube in approximately position 11-5 inside the reactor vessel, even with the horizontal centerline of the core.

Data previously calculated from the Inhour relationship for Δk vs control rod position were used to find Δk compensated by the transient rod. Figure 9 shows the results of the transient rod calibration.

4. Neutron Flux Distribution and Power Calibration (R.Wagner)

The neutron flux distribution in the 24 plates/assembly "B" core has been determined. A further, more extensive series of flux measurements using cobalt wires is to be performed prior to the transient tests. The purpose of this first measurement was to obtain vertical and horizontal profiles of the neutron flux and to calibrate the operational linear power recorder.

The flux mapping was done by activating gold foils (5/32 inch diameter x 0.005 inches thick). For the vertical profiles, 13 foils were fastened with Scotch electrical tape at 2-inch intervals on each of 9 plexiglass foil holders (0.060 inches x 0.25 inches x 24 inches). One of these holders was inserted into the center water channel (0.070 inches across) of each of the eight assemblies of the NW quadrant and into an "A" dummy assembly in the reflector. A series of foils placed at 2 inches below the centerline of the core was inserted to obtain two horizontal flux profiles of the NW quadrant, and a diagonal flux profile of the core. Figure 10 shows the core plan-form and the foil loading pattern.

FIGURE 8

(57-4867)

FIGURE 9
(57-4644)

- 13 FOILS PER HOLDER
- 1 FOIL PER HOLDER

FOIL HOLDER LOADING PATTERN SPERT I "B" CORE 24 PLATES / ASSEMBLY

FIGURE 10
 (57-4645)

The reactor was operated at a power of ~ 100 watts for about 10 minutes. The critical control rod position was 10.55 inches withdrawn, which means that the cadmium of the control rods extended 14.7 inches into the core.

The activity of the foils was determined in the Spert proportional flow counter. The counting rates averaged about 25,000 counts/minute. Two-minute counts were taken in order to keep the error due to statistical fluctuations within 1%. The counter was calibrated to absolute nvt by counting four of the foils in a calibrated counter system at the MTR. The factor obtained was $8.43 \pm 0.01 \times 10^6$ neutrons/cm² for 1 count/minute after correcting for radioactive decay of the foils. The principal source of error was considered to be in the positioning of the foils in the fuel assembly. The vertical positions were known to within $\pm 1/8$ inch but the horizontal positions are known to only $\pm 1/2$ inch.

No cadmium ratios were taken in the core due to the difficulty of getting cadmium-jacketed foils into the small water channels. The cadmium ratio taken in the reflector (position 2-2) was 9.3. For determining the thermal nvt a cadmium ratio of 2.7, which was obtained from the "A" core, was used.

The activation nvt was found by summing the integrals of the vertical flux plots.

$$\text{Average activation nvt} = \frac{\sum \text{nvt} (V) \Delta V}{\sum \Delta V} = 5.30 \times 10^{11} \frac{\text{neutrons}}{\text{cm}^2}$$

Applying the cadmium ratio, 2.7, the average thermal nvt is found to be:

$$\text{nvt} = 3.34 \times 10^{11} \text{ neutrons/cm}^2.$$

The flux/watt was determined as:

$$\frac{\phi}{P} = \frac{1}{\sigma_{\text{N}^{235}} \times 185 \frac{\text{mev}}{\text{fission}}} = 4.22 \times 10^6 \frac{\text{neutrons}}{\text{cm}^2 \text{ sec watt.}}$$

The energy released during the run is:

$$E = \frac{(3.34) (10^{11}) \frac{\text{neutrons}}{\text{cm}^2}}{(4.22) (10^6) \frac{\text{neutrons}}{\text{cm}^2 \text{ sec watt}}} = 7.91 \times 10^4 \text{ watt sec.}$$

The operational linear power instrumentation is composed of a Spert boron chamber, a linear amplifier and a 10 mv Brown recorder. The integration of the linear power record was 325 unit-sec or 70.2 μ amp-sec from the calibration of the linear amplifier and Brown recorder by a current source. The power calibration of the linear power instrumentation is thus:

$$\frac{(7.91)(10^4) \text{ watt-sec}}{70.2 \text{ } \mu\text{amp-sec}} = \frac{1130 \text{ watts}}{\mu\text{amp}}$$

This high sensitivity is due to the position of the boron chamber at the horizontal center line, 15 inches from the edge of the core.

Figures 11, 12, and 13 show respectively the vertical, horizontal, and diagonal neutron flux profiles through the core. Figure 14 is a contour map derived from the horizontal profiles. The ratio of the maximum in the core to the average over the core was 2. The maximum was in assembly 4-4 SE at 14 inches from the top of the core.

5. Temperature Coefficient (T. Quigley)

On August 2, 1957 a series of critical experiments was conducted to determine the change in reactivity as the temperature of the reactor was increased above ambient levels. A second series was conducted on September 3, 1957 in conjunction with the void coefficient measurements.

The 32 assembly "B" core was loaded in the reactor vessel. Four 30 kw heaters were placed in the reactor tank and a propeller type stirrer was used to keep the water at a uniform temperature. The water level was maintained at 2 ft. over the core.

The control rod position with the reactor just critical was found at ambient temperature and at approximately each 7°C increment up to and including boiling, which was 94.9°C. The heaters were shut off during the time the critical position of the control rods was being ascertained. This procedure was satisfactory at the lower temperatures where the temperature quickly stabilized after the heaters were turned off. However, at high temperatures the temperature fell rapidly after the heaters were turned off. At temperatures above 60°C when the heaters were turned off prior to finding the critical position of the control rods, the rapid drop in temperature with the resulting continuous change in reactivity made it difficult to make accurate measurements. At the 88.9°C point it was necessary to turn the heaters on for 90 sec after the control rods had been partially withdrawn, but the heaters were turned off before the critical rod position was recorded. At the boiling point the heaters were left on during the determination of the critical rod position.

VERTICAL NEUTRON FLUX PROFILES
 SPERT I "B" CORE
 24 PLATES / ASSEMBLY
 1.25 WATT REACTOR POWER

FIGURE 11
 (57-4649)

DIAGONAL NEUTRON FLUX PROFILE
 SPERT I "B" CORE 24 PLATES / ASSEMBLY
 NW→SE 1.25 WATT REACTOR POWER

FIGURE 12
 (57 4652)

HORIZONTAL NEUTRON FLUX PROFILE
 SPERT I "B" CORE 24 PLATE / ASSEMBLY
 S → N THRU 4TH. ROW
 1.25 WATT REACTOR POWER

FIGURE 13
 (57-4643)

ISO-NEUTRON MAP OF NW QUADRANT SPERT I-B 24 PLATE/ASSEMBLY

NEUTRON FLUX $\times 10^{-6}$
REACTOR POWER AT 1.25 WATT

FIGURE 14
(57-4647)

Figure 15 shows the control rod critical position as a function of temperature.

Figure 16 shows Δk loss vs temperature and $\Delta k/^\circ\text{C}$ loss (i.e., temperature coefficient) vs temperature.

The portion of the data which was obtained on September 3 with the stirrer on was not included since all previous data were obtained with the stirrer off. The reactivity loss was determined from the control rod worth data which were taken on July 29, 1957.

6. Void Coefficient (J. Grund, W. J. Neal)

In order to develop a method for measuring the void coefficient of a reactor at high temperatures and pressures such as will be found in Spert - II and Spert - III, a series of experiments was conducted using aluminum and lead to simulate voids. The use of materials such as these has the advantages of ease of handling and more precise volume control than has been possible with previous air void experiments. The results of these experiments will be correlated with the results of the air void experiments which are now being conducted.

The selection of materials for void simulation was governed by three major considerations; availability, low absorption cross-section, and low moderating ability. Aluminum and lead both fulfill these conditions to some degree. Aluminum has a slightly larger absorption cross-section and moderating power than does lead, but lead does have some inelastic scattering and this could make it a less desirable void simulator than aluminum. These experiments were an endeavor to resolve this question and evaluate which, if either, of these materials may be used for void studies.

The aluminum used was commercial 2S material in strips 2 inches x 25-1/2 inches x 0.032 inches; the lead was commercially pure 0.050 inches-thick sheets which were cut into strips 2 inches x 25-1/2 inches. The aluminum and lead strips were cut 3/8 inch longer than the fuel plates to facilitate their removal from the fuel assemblies. In the experiment which used the lead strips as simulated voids a 63 pound test braided nylon cord was attached to each lead strip through a 1/8 inch hole drilled 1/4 inch below the top of the strip. These cords were long enough to reach the top of the reactor vessel. Removal of the strips could then be accomplished by pulling the strips out with the attached strings which made removal of the fuel assembly unnecessary. Later, to avoid difficulties which were encountered with the nylon cord, a 4-inch length of soft iron wire was attached to each aluminum strip.

In order to obtain reproducibility of data the following procedure was adopted: the fuel assemblies which contained voids were placed in the reactor and locked in. The holding mechanism was

FIGURE 15

5/ 4/59)

FIGURE 16
(57-4735)

then not disturbed until after the position of the control rods with the reactor just critical was determined for both the voided and the non-voided conditions. During the course of the experiments the rods were never scrambled but were always driven in to avoid jarring the reactor. Where practical the reactor water was not changed during the course of an experiment. Whenever possible, the experimental series was completed in a period of one day. The instrumentation for the void experiments was the same as that used during the reactor critical experiments. All void experiments were conducted on the 32 fuel assembly, 24 fuel plates/fuel assembly core. (Figure 7.)

Lead was used as a simulated void in one experiment. A lead strip was inserted in each of the 23 water channels of an assembly in position 4-4. (Figures 1 and 7.) The position of the control rods with the reactor just critical was determined both with the lead strips in and out of the assembly at various temperatures between ambient ($\sim 21^\circ\text{C}$) and boiling ($\sim 95^\circ\text{C}$). The reactor was heated with four 30 kw electric heaters with the critical position of the control rods determined at approximately 25°C temperature increments. The average void coefficient over this temperature range corrected for neutron absorption in the lead is:

$$-6.5 \times 10^{-4}\% \Delta k/\text{cm}^3 \quad \text{or} \quad -1.06 \times 10^{-2}\% \Delta k/\% \text{ void.}^*$$

The correction made for absorption is calculated from the effect discussed in (4). The conventional $\Delta k/k = \Delta \Sigma / \Sigma$ relation has been weighted by the flux squared when the material is not uniformly distributed throughout the core. The macroscopic absorption cross-section for lead was taken as 0.007 cm^{-1} . Table 2 gives the void coefficient at the different temperatures both corrected and uncorrected for absorption effects.

Aluminum was used in five experiments as a simulated void. The first of these experiments was identical in procedure with that conducted with lead except that the temperature increments were approximately 18°C between the criticality determinations. The void coefficient using aluminum, averaged over this temperature range and corrected for absorption effects is:

$$-6.3 \times 10^{-4}\% \Delta k/\text{cm}^3 \quad \text{or} \quad -1.02 \times 10^{-2}\% \Delta k/\% \text{ void.}^*$$

The macroscopic cross-section for aluminum absorption used in the correction calculations was 0.0115 cm^{-1} . Table 3 shows the results of the experiment. Comparison of this with the results of the experiment using lead shows an agreement within 4%. The air void measurements are as yet incomplete, and no comparison can be made.

*

In this discussion the moderator volume referred to in expressing % voids is that contained in the assembly or assemblies involved in the test and not to the total core moderator volume.

A void position importance experiment using aluminum as the void was conducted in the following manner; an aluminum strip was placed in each of the 23 water channels of a fuel assembly loaded in the reactor. The position of the control rods with the reactor just critical was determined. Eleven of the aluminum strips were removed and the procedure was repeated. The remaining 12 strips were then removed and the critical position of the control rods was again determined. This procedure was followed for each of the fuel assembly positions in the NW quadrant of the reactor in the following order: 4-4, 4-3, 4-2, 3-2, 3-3, 3-4, 2-4 and 2-3. During the experiment, the moderator temperature was maintained at 22°C. The results of this study are graphically represented in Figure 17.

A third investigation was conducted by placing aluminum strips in all of the water channels of the four center fuel assemblies (positions 4-4, 4-5, 5-4 and 5-5). The control rod critical position was determined with the moderator temperature maintained at 23.1°C in each of the following steps:

1. Aluminum strips in each water channel of the four assemblies;
2. Aluminum strips removed from water channels 4, 8, 12, 16 and 20 of each assembly;
3. Aluminum strips removed from water channels 2, 6, 10, 14, 18 and 22 of each assembly;
4. Aluminum strips removed from water channels 3, 7, 11, 15, 19 and 23 of each assembly;
5. All remaining aluminum strips removed from the assemblies.

The average void coefficient as determined by this experiment is:

$$-6.2 \times 10^{-4} \% \Delta k / \text{cm}^3 \quad \text{or} \quad -4.3 \times 10^{-2} \% \Delta k / \% \text{ void.} *$$

Figure 18 is a plot of the void effect.

To determine an average void coefficient over the entire core aluminum strips were placed in water channels 2, 5, 8, 11, 14, 17, 20 and 23 of each fuel assembly in the core. The position of the control rods with the reactor just critical was found with 8, 7, 6, 5, 4, 3, 2, 1 and 0 aluminum strips per assembly. The strips were removed from the water channels in the following order: 11, 23, 5, 17, 8, 20, 2 and 14. The moderator temperature was maintained at 24.0°C. The average void coefficient was determined to be:

$$-4.2 \times 10^{-4} \% \Delta k / \text{cc} \quad \text{or} \quad -0.25 \% \Delta k / \% \text{ void.} *$$

Figure 19 is a plot of the void effect.

* See note on page 33.

TABLE 2

VOID COEFFICIENT STUDIES WITH LEAD AS SIMULATED VOID

Temp °C	# of lead strips	Volume of lead (cm ³)	% moderator void	% Δk loss uncorrected	% Δk loss corrected	% Δk loss/cm ³ uncorrected	% Δk loss/cm ³ corrected
21.6	22	849	52.4	0.68	0.56	8.0 x 10 ⁻⁴	6.6 x 10 ⁻⁴
47.4	23	888	54.8	0.69	0.57	7.8 x 10 ⁻⁴	6.4 x 10 ⁻⁴
71.1	22	849	52.4	0.68	0.57	8.1 x 10 ⁻⁴	6.7 x 10 ⁻⁴
95.3	21	809	49.9	0.63	0.52	7.8 x 10 ⁻⁴	6.4 x 10 ⁻⁴

(see text)

TABLE 3

VOID COEFFICIENT STUDIES WITH ALUMINUM AS SIMULATED VOID

Temp °C	# of Al strips	Volume of Al (cm ³)	% moderator void	% Δk loss uncorrected	% Δk loss corrected	% Δk loss/cm ³ uncorrected	% Δk loss/cm ³ corrected
22.4	23	568	35.0	0.52	0.39	9.2 x 10 ⁻⁴	6.9 x 10 ⁻⁴
41.2	23	568	35.0	0.49	0.36	8.6 x 10 ⁻⁴	6.3 x 10 ⁻⁴
57.1	23	568	35.0	0.49	0.36	8.6 x 10 ⁻⁴	6.3 x 10 ⁻⁴
75.5	23	568	35.0	0.43	0.30	7.6 x 10 ⁻⁴	5.3 x 10 ⁻⁴
95.3	23	568	35.0	0.50	0.37	9.0 x 10 ⁻⁴	6.7 x 10 ⁻⁴

(see text)

$\% \delta \kappa$ LOSS / $\%$ VOID

RELATIVE VOID POSITION IMPORTANCE
 $\% \Delta \kappa / \%$ VOID VS. VOID POSITION
 SPERT "B" CORE
 32 ASSEMBLIES
 24 FUEL PLATES / ASSEMBLY

FIGURE 17
 (57-4734)

VOID EFFECT SPERT I "B" CORE
 ALUMINUM VOIDS IN ASSEMBLIES
 4-4, 4-5, 5-4 AND 5-5
 32 FUEL ASSEMBLIES
 24 FUEL PLATES / ASSEMBLY

FIGURE 18
 (57-4737)

VOID EFFECT SPERT I "B" CORE
ALUMINUM VOIDS DISTRIBUTED
THRU ENTIRE CORE
32 FUEL ASSEMBLIES
24 FUEL PLATES / ASSEMBLY

FIGURE 19
(57 4740)

The effect of temperature on the void coefficient with eight aluminum strips in each fuel assembly in the core was investigated as a function of temperature from 21.5°C to boiling. The strips were placed in water channels 2, 5, 8, 11, 14, 17, 20 and 23 and the critical position of the control rods was determined at each 10°C rise in temperature. The above procedure was followed with the aluminum strips removed from water channels 5, 11, 17 and 23 of each fuel assembly and again with no aluminum strips in the reactor. The void and temperature effect are shown in Figure 20.

Figure 21 is a plot of void effect at different temperatures using data derived from Figure 20.

No void coefficient experiments were performed in the reflector. However, Spert - I "A" core data shows that an appreciable void effect exists in the first inch of reflector, and correction for the amount of void which extended into the reflector above and below the core was made on the basis of the "A" core experiments. About 7.5% of the total void was in the reflector, and as the "A" core data show that the reflector void worth in this region is about 20% of the average core void worth, the total void effects were reduced by 1.5%.

The void measurements which have been completed on the 24 plate B core and the conclusions from the results are summarized below:

1. The void coefficient for large void fractions in the core center has been measured as a function of temperature by using both Pb and Al as void simulators. The central four fuel assemblies were cycled between zero and 50% void for Pb and zero and 35% void for Al at selected temperatures from ambient to boiling. Under these conditions it was found that the void coefficient was independent of moderator temperatures and that the results from Pb and Al were in good agreement.

2. The void importance for a full core height void in one assembly was found as a function of void size up to 35% void and as a function of grid position for one complete quadrant of the core. The void coefficient was found to be independent of void size and varied with grid location as indicated in Figure 17.

3. The void coefficient was found for uniformly distributed voids in the entire core as a function of void size from zero to 12% void and as a function of reactor temperature from ambient to boiling. The void coefficient was found to vary with both void fraction and reactor temperature as indicated in Figures 19, 20 and 21. In other words the nonlinearities in void coefficient both as a function of void fraction and of temperature begin to become apparent when an appreciable fraction of the total core volume is uniformly occupied by voids.

VOID EFFECT SPERT I "B" CORE
TEMPERATURE EFFECT WITH UNIFORMLY
DISTRIBUTED ALUMINUM VOIDS
32 FUEL ASSEMBLIES
24 FUEL PLATES / ASSEMBLY

FIGURE 20
(57-4738)

DISTRIBUTED ALUMINUM VOIDS REACTIVITY
EFFECT AS A FUNCTION OF TEMPERATURE
AND VOLUME IN SPERT I "B" CORE
32 FUEL ASSEMBLIES
24 FUEL PLATES / ASSEMBLY

FIGURE 21
(57-4741)

III. SPERT - II

The work on Spert - II, except for design and construction of the reactor, is in the early preliminary stages. Planning has been initiated on the control room hardware, which includes the console and some instrumentation which is unique with a heavy water system. It is anticipated that much of the control system design which has been delayed for Spert - III will be applicable to Spert - II.

IV. SPERT - III

A. CONTROL SYSTEM

1. Control System Drives (F. Schroeder)

The contract with Westinghouse Electric Company for the control rod and transient rod drive motors has been cancelled and new specifications issued for bids. The new system specifications call for 3 speed, constant torque, squirrel cage induction gear motors, with integrally mounted speed reducers and brakes.

2. Control System Circuit Drawings (Don Jones, F. Schroeder)

Extensive changes in control circuitry were necessitated by the recent changes in drive motor specifications. These and other minor changes and corrections have been made on the drawings and it is now believed that these basic schematics reflect the desired control philosophy. The control system design has now been frozen and work is underway on the "wiring-schematic" drawings. It has been decided to dispense with the usual wiring diagrams in favor of "wiring-schematic" type drawings which will show wire numbers and terminal designations but will be in the layout style of a schematic drawing. Once these drawings are completed, tables will be prepared for use in the actual wiring. Inasmuch as most of the wiring will be point-to-point between terminal strips, it is felt that this system will save considerable drafting time with no loss of wiring ease.

3. Control System Construction (F. Schroeder)

Construction of the control system relay racks is now virtually complete. They are ready for wiring as soon as the circuit drawings and wiring tables are completed.

The drive motor control center, control power contactors, main power-standby power bus transfer contactors and reactor-drydock

switchover units have been designed and the switchgear is now on order.

Now that the control circuit design has been frozen and most of the components placed on order, layout and construction of the reactor control console has been started.

Bids were opened September 7th for the TV system for Spert - III. The award has been postponed pending the receipt of additional information from two of the bidders.

Bids were opened August 6th for the 3000-ft cables for the Spert - III control and transient instrumentation. The order for 70, RG8A/U cables was awarded to Federal Cable Division of Royal Electric Corp. with a quoted shipping date of October 5th. The order for 13, 19-conductor neoprene-jacketed control cables was awarded to John A. Roebling's Sons Corp, with delivery quoted as about October 1st, 1957. (Subsequent information indicates that this shipment will not be completed until November 1.) The order for a 19-conductor shielded communication cable was placed with Sequoia Wire Company with a quoted shipping date of October 4, 1957.

The cable conduits and trenches in the control room have been modified. The modifications, with the addition of some duct work, will permit a more convenient routing of the signal and control cables.

V. DATA REDUCTION AND INTERPRETATION

A. Data Reduction (J. C. Haire)

A compilation of data obtained from the sub-cooled step transient experiments which have been performed in Spert - IA is now in the rough draft stage and will be issued in the near future,⁽⁵⁾ These data have been summarized and discussed in a topical report⁽⁶⁾ but no detailed collection of the data has been published. This report contains data on reactor power, selected fuel plate temperatures, and end-box pressures as functions of time during the tests; tables of data; and a discussion of experimental uncertainties.

The primary data reduction process (conversion from oscillographic trace records to tabular and punched card data) is now complete for the boiling and intermediate temperature step transient experiments which have been performed in Spert - IA. Work is underway on the data compilations and correlations for these experiments and on the preparation of a topical report which will cover the higher-temperature results.

The current status of the data reduction activities for the Spert - IA kinetic tests is summarized in Table 4.

STATUS OF DATA REDUCTION ACTIVITIES FOR SPERT-IA KINETIC TESTS, SEPT. 19,1957

A.	Total number of times reactor made critical (7/11/55 - 5/24/57)	885
B.	Total number of kinetic tests performed (8/31/55 - 5/24/57)	606
C.	Total number of tests containing sufficient information of a quantitative nature to warrant some degree of data reduction * (Percent of B)	417 (69%)
D.	Number of tests for which <u>all</u> such data have been reduced.	89
E.	Number of tests for which the most significant data have been reduced	144
F.	Number of tests for which qualitative descriptions have been prepared containing approximate quantitative measurements of significant features.	88
G.	Total number of tests which have been reduced in some degree (D+E+F) (Percent of C)	321 (77%)

Most of the test data which is shown as being partially reduced (E and F) will not be further reduced except as may be required to answer specific questions. The 96 tests which remain to be reduced consist principally of ramp rate and stability experiments.

The 606 kinetic tests performed over a period of 21 months represents an average of about 1.4 tests per working day.

B. Data Interpretation (R. Miller)

Subsequent to the completion of an investigation of reactivity behavior during Spert - IA sub-cooled transients it⁽⁷⁾ appeared that a similar study of transients initiated at or slightly below boiling should reveal additional information on the reactor self-shutdown process. A series of 24 transients was chosen for this purpose covering the range of periods from 5.8 msec to 9 sec. As with the previous sub-cooled series, the procedure has been to obtain a graph of the most reliable power data for each transient and fit this with a smooth curve so that regularly spaced data for digital computation are obtained. This method artificially reduces the "noise" level of the input data. Preparation

* Many of the remaining tests were performed for the purpose of obtaining visual photographic data, performance data on instrumentation, or exploratory qualitative information and hence have also produced significant data. "Data Reduction" as used here refers to the process of transforming oscillographic trace records into tabular and punched card data and into graphs.

of data from these tests for entry into the IBM 650 computer has been completed and calculations have been successfully finished on about 19 of these problems. The remaining 5 problems have yielded unsatisfactory solutions for reasons which are yet uncertain but attempts to complete these are currently underway.

Studies are being conducted by the MTR computer group to determine the extent of systematic errors which may be inherent in the numerical solutions of the reactor kinetics equations as programmed for the IBM 650. Such errors may conceivably become important in the determination of the reactivity compensation at the time of peak power, $k_c(t_m)$, during a transient power excursion, since this quantity often represents small differences between large numbers. A further concern about the accuracy of $k_c(t_m)$, lies in the dependance of these numbers upon parameters which appear in the reactor kinetics equations. Uncertainties of appreciable proportions exist both in the neutron generation time, and in the effective delayed neutron fraction. Computations are underway to determine the importance of these parameters.

The analytic reactor model examined by Griffing⁽⁸⁾, which uses the reactor kinetics equations coupled with a reactivity feedback yields the reactor power as a function of time when the reactor is given an initial positive reactivity. If the reactor power behavior so calculated is entered in the equations which are presently used in the reactivity behavior calculations and solved for reactivity, a check on the self-consistency of the equations and programs used is obtained. Two such checks have been completed and more are planned.

VI. THEORETICAL

A. Critical Mass Estimates - "B" Core (T. Watanabe)

Critical size calculations based on two group diffusion theory were made for the Spert - I "B" core for different fuel concentrations as determined by the number of fuel plates inserted per fuel assembly. All calculations were done on the IBM 650 using the PROD II code. The results of these calculations are given in Table 5, along with the experimentally measured values.

TABLE 5

CALCULATED CRITICAL MASSES - SPERT - III CORE

Fuel Plates Fuel Assembly	Avg. Al/H ₂ O	Loading gU ²³⁵ /assembly	Calculated crit- ical mass (Kg)	Experimental * critical mass (Kg)
12	.509	84	3.6	4.27
24	1.247	168	4.786	4.34
16	.709	112	3.43	3.58
12 + 12 dummy	1.247	84	6.75	6.38

* See Section II.B.1, page 10 of this report.

B. Calculated Void and Temperature Effects - "B" Core (T. Watanabe)

The reactivity effect associated with uniform temperature change or with uniform void content has been calculated for the 24 plate B core using the PROD II code on the IBM 650. The reactivity effect is predicted from:

$$\rho = \frac{\Delta \kappa}{\kappa} = \frac{\Delta v}{v_0} = \frac{v - v_0}{v_0}$$

where $v_0 = 2.47$ for the just critical reactor according to the PROD II program, and v is the value obtained for this constant for changes in the core due to temperature of moderator void. The predicted reactivity effects of temperature and voids are shown in Figures 22 and 23. Comparison with the experimental data presented in Figures 16 and 19 shows both calculations to be high by about a factor of two. The same discrepancy was noted in the A core calculations.

C. Initial Behavior of the Spert - I Reactor by the Energy Model (G. M. Griffing, L. J. Deverall)

An extensive set of computations has been completed on the "Energy Model" which was introduced to better understand the initial behavior of the Spert - I transients. In the Energy Model it is assumed that the compensated reactivity is directly proportional to the energy available in the reactor system. This model is a refinement of the model introduced by K. Fuchs and investigated recently by the Spert staff. The refinements consist of including the delayed neutrons in the reactor kinetic equations and adding an energy escape term to the relation which describes the energy behavior. As a part of the study the parameters appearing in the equations were varied to study their effects on the power behavior. Although this model is clearly an over-simplification of the physical situation it does show many of the features observed in Spert - I. The results of this study are being reported (8).

D. Shutdown Process of the Spert - I Reactor (L.J. Deverall, G. M. Griffing)

For Spert - I transients initiated at approximately 20°C and in which the reciprocal period is less than 20 sec⁻¹, the fuel plate temperature does not rise sufficiently by the time of maximum power for boiling to occur. For this region an attempt has been made to determine whether moderator expansion due to thermal effects is sufficient to shut down the reactor. Using the observed temperature traces from the fuel plate thermocouples, an estimate of the temperature rise of the moderator can be calculated. The compensated reactivity is also obtainable from the use of the observed power traces. A comparison of the two values indicates that the computations based on the temperatures are lower than the ones based on powers by a factor of about two. However, the accuracy of the available information is such that it does not exclude the possibility that the shutdown of Spert - I is due solely to thermal expansion

CALCULATED REACTIVITY LOSS
VS TEMPERATURE
SPERT "B" CORE

24 FP/FA

FIGURE 22

(57-4868)

FIGURE 23
(57 4863)

of the moderator. This work will be reported in detail (9).

E. Development of Improved Methods for Obtaining Fast and Slow Group Constants (G. D. Marshall, D. R. Metcalf, and R. A. Grimesey)*

1. Fast Constants

In a previous two-region, critical size calculation of Spert - III the fast group constants (10) used were determined by a hand calculation. The determination of an appropriate flux distribution in energy by desk calculator methods was quite lengthy and involved. The method for obtaining a lethargy dependence of the flux has been programmed for the IBM-650. Cross-section data that is used in the program is in one-tenth lethargy intervals, thus enabling one to take quite accurate account of resonance structure. This cross section data can very well be used for other programs that may be needed later, such as the resonance escape probability and age.

The flux distribution for a five-region calculation on Spert - III is at present being determined.

2. Slow Constants

Because of the appreciable absorption of the stainless steel components and the high fuel density in the Spert - III core, a method is being used to obtain thermal constants that take into account $1/v$ absorption and its effect on the neutron spectrum. Wigner and Wilkins (11), by taking this absorption into account and by assuming moderation by a monatomic H gas, have calculated spectra for several values of the parameter

$$\beta\gamma = \sum_i \frac{\Sigma_{ai}}{\xi_i \Sigma_{si}}$$

and moderator temperatures,

where

Σ_{ai}, Σ_{si} = total macroscopic absorption cross-section at thermal and scattering cross-section at epithermal of i^{th} element.

$\xi_i = \frac{2}{A_i + \frac{2}{3}}$, $A_i > 10$ = average logarithmic energy decrement of element i .

Deviation from the Maxwell-Boltzmann distribution is significant, particularly for high values of $\beta\gamma$; $\beta\gamma = 0$ corresponding to the Maxwellian case.

* Theoretical Physics and Applied Mathematics Branch

Numerical integration by hand is being used to obtain thermal constants for the five-region Spert - III critical size and flux calculations. In particular:

$$\frac{1}{\Sigma_{Tr}} = \frac{1}{\Sigma_{Tr}^H + \Sigma_{Tr}^O + \Sigma_{Tr}^{ss} + \Sigma_{Tr}^U} = \frac{\int_0^{10kT} \frac{1}{\Sigma(E)} \varphi(E, \beta\gamma, kT) dE}{\int_0^{10kT} \varphi(E, \beta\gamma, dT) dE}$$

while

$$\bar{D}_2 = \frac{1}{3\Sigma_{Tr}} \text{ and } \bar{L}^2 = \frac{\bar{D}_2}{\bar{\Sigma}_a}$$

where

$\varphi(E, \beta\gamma, kT)$ = Wigner-Wilkins spectrum.

$10kT$ = Convenient upper thermal limit.

Σ_{Tr}^i = Macroscopic transport cross-section of element i.

$\bar{L}^2$ = Average slow diffusion length squared.

$\bar{\Sigma}_a$ = Average total macroscopic absorption cross-section.

$\bar{D}_2$ = Average slow diffusion constant.

The average diffusion length squared and $\bar{\Sigma}_a$ for Region II of Spert-III has been calculated by the two methods and the comparison is shown in Table 6. The Maxwell-Boltzmann $\bar{L}^2$ values given here differ additionally from the Wigner-Wilkins values in that the relation

$$\frac{1}{\bar{\Sigma}_{Tr}} = \frac{1}{\Sigma_{Tr}}$$

was used for the former.

TABLE 6

$\bar{\Sigma}_a$ and $\bar{L}^2$ for SPERT - III Core

T(°F)	Wigner-Wilkins		Maxwell-Boltzman	
	$\bar{\Sigma}_a$ (cm ⁻¹)	$\bar{L}^2$ (cm ²)	$\bar{\Sigma}_a$ (cm ⁻¹)	$\bar{L}^2$ (cm ²)
68	.205	.959	.251	.569
450	.158	1.734	.142	1.513
650	.139	2.751	.116	2.587

VII. INSTRUMENTATION

A. Instrumentation for "B" Core Critical Experiments (G. Brockett, E. Morris)

1. Counter System

For critical experiments, it is necessary to detect the very beginning of neutron multiplication. With this in mind, the following was done: Two aluminum tubes sealed at one end and fitted with an adjustable clamp at the other were fabricated for placing chambers directly in the reactor. Two B¹⁰-lined chambers and two BF₃ chambers were fitted with special cables and preamplifiers such that they could be lowered in the tubes or along the outside of the reactor tank. The input time constants of the preamplifiers were adjusted for the greatest sensitivity at low counting rates, which process included keeping the chamber signal cables as short as possible. The discriminator circuits in the linear pulse amplifiers were modified for greater precision and stability. The output circuits were changed to a dual cathode follower for delivery of a 5-volt discriminator pulse 3000 feet to the control room. The input amplifiers of the scalars were modified to accept a 5-volt pulse.

The counter systems were then calibrated after installation for their neutron and gamma sensitivity as a function of discriminator level and chamber supply voltage. Polonium-beryllium was used as the neutron source and Cobalt 60 for the gamma source. The chamber voltage and discriminator level were then adjusted for the greatest neutron sensitivity at the greatest neutron-to-gamma discrimination.

Since the actual reactor neutron source is antimony in a beryllium holder, gamma radiation is emitted as well as neutrons. Placing a typical counter system in the reactor tank one fuel assembly spacing away from this source resulted in 204,000 counts/min. However, when a 0.020 inch-thick sheet of cadmium was wrapped around the chamber

as a thermal neutron shield, the system recorded only 870 counts/min., which shows a satisfactory gamma discrimination. The other three counter systems were checked in the same manner and installed according to Figure 2.

2. Process Instrumentation

The reactor water temperature instrumentation was given thorough maintenance and calibration checks. The operational power instrumentation was also checked out thoroughly. The linear amplifier associated with this system was modified for improved stability as discussed below.

The TV system was set up in the shop for routine checks and an investigation of its poor picture stability. Several changes were made and the system was reinstalled. The quality was improved, but was still unsatisfactory in picture contrast and stability at the console. Both these conditions were improved with the installation of a new monitor at the console and a new control unit in the bunker. This equipment change was at the expense of the supplier.

B. Performance Characteristics of Transient Instrumentation.

1. Linear Amplifiers.

Previous to the termination of the Spert - I "A" core measurements, it was noted that in the higher power ranges the linear amplifiers (used with Spert Boron Chambers) had a tendency to become unstable, resulting in high frequency oscillations. The response characteristics of these amplifiers were studied and corrections were made to increase the natural high frequency cutoff, so that an artificial roll-off network of 6 db/octave could dictate the overall response characteristics. This gave a satisfactorily stable response to a step function in all ranges. These modifications were accomplished on all the Spert - I linear amplifiers and the units were reinstalled.

2. Galvanometer Driver Amplifiers.

During the shutdown following the "A" core experiments the response characteristics of the driver amplifier-galvanometer combinations were studied to determine the feasibility of extending the system frequency response and gaining better control of the overshoot characteristics. In the original circuit, it was found that placing a large capacitative load across the output of the amplifier was necessary to prevent a severe overshoot when a step function was applied to the input. This capacity also limited the system response to 3.8 kilocycles per second. After a few minor corrections in the amplifier and with a capacitor to represent the 3000-foot cable, it was found that the amplifier cutoff had been moved up two octaves,

still maintaining 6 db/octave roll off for good stability. However, upon connecting the 3000-foot cable, the resulting system response was a rough fall-off with a sharp slope--giving as expected an overshoot for a step function input. Further investigation showed that a large capacity (0.3 to 0.6 μ f) existed between the shield of the 3000-foot driver cables and earth ground. The cutoff point represented by the combination of this capacity and the 10-15 ohm series feed back resistor in the driver amplifier fell at the same frequency point as the sharp slope (14 db/octave) in the response of the complete system.

This led to several comparison tests of various correction circuits, including the use of Twinax cable (RG 22 B/U). The results showed that the best overall circuit allowed the system response to be determined by the galvanometer characteristics. Comparison of this circuit (which was recording rise times up to 0.25 milliseconds) showed that only a frequency increase of 1/3 octave and somewhat better control of the overshoot characteristics had been accomplished. Conclusions from this investigation were that only a major modification of the driver system would allow the 3000-foot cable shield to be placed at ground potential, and thus yield the improvement indicated. It was decided that this was not justified for Spert - I but should be considered for Spert - II and -III.

3. Hum and Noise in Transient Systems

On many galvanometer traces recorded during the "A" core experiments, a ripple and noise signal of the order of 2 to 3 percent of full scale deflection was apparent. Several sources for this noise were traced to problems such as: power supply ripple and inadequate regulation; unbalance of heater-to-cathode leakage; wiring capacity pick-up and close proximity of amplifiers to magnetic fields from chokes and transformers, particularly of the saturable core type. Correction of these problems reduced the residual noise in the system to below 0.1% of full scale deflection. However, the remaining hum was discovered to come from 60 cycle induced voltages in the 3000-foot cable. These signal cables are lying on the desert and are covered with earth about 1/3 of their total length. They are also running parallel to the reactor control cables with a separation of about 10 feet on the average. An attempt to establish a ground isolation plane between the signal cables and control cables by tying the shield of some unused RG 68/u cables to the main ground buss was unsuccessful. Further separation of control and signal cables might reduce the hum considerably.

4. Thermocouple Response Time

Tests have been conducted on twisted-soldered-and butt-welded-junction thermocouples to determine their time response to an effectively instantaneous change in temperature. The testing method was as follows: a thermocouple was suspended in a small electric oven above a long column of water. One end of a stretched spring was attached to the thermocouple

and the other end to the bottom of the tank containing the water. When the proper oven temperature was reached, the spring was released pulling the junction through about 8 inches of water at a high velocity. Preliminary results of these tests indicate little difference between twisted-soldered and butt-welded thermocouple junctions. Chromel-alumel thermocouples heated to 210°C and immersed in 40°C water had a response time of approximately 50°C per millisecond while a copper-constantan thermocouple under the same conditions had a response of approximately 100°C per millisecond.

5. Pressure Transducers

Hardware has been completed for adapting many types of pressure transducers to fit the Wianko dynamic pressure calibrator. Results from these tests should give an indication of the natural frequency characteristics of various transducers and their associated plumbing, as they would be used in reactor measurements.

6. Spert Boron Chambers

The Spert project has been collecting data from ionization chambers which were built at Spert and used for the last two years. These chambers are 9 inches long and have a diameter of 2.5 inches. They were built of brass, except for a boron-lined steel sleeve, and filled with argon gas at one atmosphere pressure. Output currents measured were from 10^{-10} to 10^{-5} amps and voltages applied to the high voltage electrode were 1000 v to 1500 v. These chambers are called Spert Boron Chambers, hereafter referred to as "SBC".

An attempt was made to improve the design, mainly through enlarging the center collector so as to more nearly approach a parallel plate type of chamber. Tests proved this "SBC" No. 5 to be superior to the first four "SBC's". The voltage applied to the high voltage electrode could be varied from 400 v to 1500 v and the output current would remain at a fixed level, providing the changes were not surges. When testing this chamber in the MTR thermal column currents of up to $19\mu\text{amp}$ were observed.

Since chambers will be needed for Spert - II and -III it was decided to further improve the device by installing a gamma compensating section, and to increase the areas of the high voltage and collection plates in order to obtain higher current output. In construction, a B^{10} -lined sleeve was placed between positive and negative sleeves, with gas volumes on each side of the B^{10} -lined sleeve calculated to be the same. With a positive field on one side of the collector and a negative field on the other, current resulting from gamma rays is reduced considerably. The neutron sensitivity remains, however, as the B^{10} is in only one side of the chamber. Construction of the chamber is shown in Figure 24.

The vacuum system used to fill the chambers with argon was a Consolidated Vacuum Corporation model LCI-12A vacuum coater, modified by replacing the bell jar with "O" ring flanges and tubing. A suitable vacuum (1×10^{-5} mm Hg) was obtained and argon gas under a pressure of one atmosphere was admitted to the chamber.

GAMMA COMPENSATED ION CHAMBER

FIGURE 24
(57-4642)

In testing for gamma sensitivity a brass sleeve was used instead of the B^{10} -lined collector sleeve. A cobalt 60 gamma source was placed a fixed distance from the chamber. The plates were connected to variable high voltage, low impedance power supplies. With zero potential on one high voltage electrode, the potential on the other high voltage electrode was varied in steps of 85 v from -85 v to -935 v. With a load resistance of 10^{10} ohms, the resulting output current was 6.4×10^{-11} amps, and was flat within 1.5% over this voltage range. Reversing the above procedure the voltage on the negative electrode was then fixed at -935 v, and the voltage on the positive electrode was varied from 85 v to 935 v. The resultant current indicated that a 96% cancellation of the original current resulting from the effect of gamma rays was obtained when the absolute values of the voltages were equal, and a 90% cancellation was obtained at the maximum difference in the absolute values of the voltages.

These tests were not totally conclusive because of power supply drift and noise problems, but gamma compensation appears to be quite satisfactory. Thorough tests are anticipated for determining neutron sensitivity with both brass and B^{10} -lined collectors.

C. Other Projects and Investigations

The 12 remaining thermocouple amplifiers have been modified with a new circuit for higher common mode rejection. They have been checked out and reinstalled.

Construction of a portable current step-function generator has been completed for the purpose of determining the stability and response characteristics of linear current amplifiers. This unit employs the principle of an interrupted light beam impinging on a photo-electric cell. The internal impedance is in excess of 3×10^{12} ohms. Initial adjustments allow a rise time of the order of 10 micro-seconds. The instrument will make possible the checking of current amplifiers while installed as well as in the laboratory.

Several transistor direct-current amplifiers have been studied with galvanometer-driver application in mind. Performance characteristics such as constant current output, low drift, and temperature stability were of special concern.

In view of the possibility of ultrasonics being useful in measuring water density, a simple magneto-strictive driver with a 30 kilocycle resonant frequency, has been fashioned, and a condenser type hydrophone has been constructed for use as a probe. Tests are in progress to determine the sensitivity of the system.

VIII. ENGINEERING

A. SPERT - I

1. Modification of the Reactor Internal Structure (T. R. Wilson R. E. Heffner *)

Prior to the initiation of experimental work on the Type "B" core, the reactor internal structure was modified to eliminate the upper fuel assembly hold-down grid. Experience with the grid had shown that it interfered with the removal of instrument leads from the core fuel assemblies; therefore, a bottom locking mechanism was designed, constructed, and installed for testing.

In order to utilize the Type "B" fuel assemblies with the new locking mechanism, an end box extension provided with locking lugs was installed on each assembly. The locking mechanism, which is fastened beneath the original bottom grid, consists of a series of movable horizontal bars with protruding rollers which, when properly positioned, engage the lugs on the fuel assembly end box, thereby locking the assembly in position. The mechanism is operated from the reactor top by an extension tool.

Other minor modifications which were completed during the shut-down period include (1) the design and installation of new shim rod magnets; (2) the design and installation of a new pneumatic rabbit shuttle which enters the aluminum cross and allows access to approximately the core horizontal center line; and (3) the design and fabrication of additional handling tools for the Type "B" assemblies.

2. Corrosion Studies - Type "B" Fuel Assemblies (T. R. Wilson)

During the course of the critical experiments with the Type "B" assemblies corrosion activity was observed on some of the removable fuel plates. A photograph of a removable plate showing the type and extent of the corrosion is shown in Figure 25. In conjunction with the MTR Metallurgy Section, an investigation of the cause of the corrosion was initiated. No definite conclusions have been reached; however, it appears that the corrosion is due to the lack of cleaning by the vendor after fabrication. The investigation is still in progress.

B. SPERT - II

1. Mechanical Design Status (R. E. Heffner * and T. R. Wilson)

The status of the mechanical design, drafting, and component procurement for those parts of the facility assigned to the Spert project is summarized in Table 7.

2. Control Rod Poison Section Development (R.E. Heffner*, T.R. Wilson)

The performance of the MTR control rod poison sections has demonstrated that a poison section fabricated from an Al-Cd-Al sandwich is

* Division Engineering Branch

TABLE 7

SPERT-II - MECHANICAL DESIGN AND MATERIAL PROCUREMENT STATUS REPORT

Component Description	% Comp. of Mech. Design	% Comp. of Drawings	% Comp. of Fab- rication	Est. Comp. or Del. Date	Remarks
<u>Control Rod Drive Mechanism</u>	90	90	----	----	Modified SPERT-III Design
<u>Transient Rod Drive Mechanism</u>	90	90	----	----	Modified SPERT-III Design
<u>Rod Drive Auxiliary Equipment</u>					
Motor and Motor Controls	95	Pur. Items	----	----	Modified SPERT-III Design
Reactor Control Wiring	50	0	----	----	
Air Compressor	0	Pur. Items	----	----	
Emergency Power Generator	0	Pur. Items	----	----	
Drive Unit Mounting Flange	75	50	----	----	Awaiting Vessel Design Details
Air Control System	50	0	----	----	
<u>Core and Internal Components</u>					
Core Structure	100	100	----	----	
Control Rods	100	100	----	----	
Transient Rod	100	100	----	----	
Fuel	100	100	----	----	Will Use SPERT Type B Fuel
Flow Skirt	100	100	----	----	
Core Structure Support	100	100	----	----	
Vessel Diaphragm	100	90	----	----	Awaiting Vessel Design Details
<u>Miscellaneous</u>					
Air Piping	0		----	----	
Handling Tools	0		----	----	
Fuel Storage Racks	0		----	----

FIGURE 25

(57-4522)

**CORROSION ON SPERT "B" ASSEMBLY
REMOVABLE PLATE**

unsatisfactory even for low temperature operation because of the inadequacy of the mechanical bond between the metal sheets and the resulting inability to hold the cadmium in position. The high operating temperature (400°F) proposed for the Spert - II reactor, and the availability and excessive cost of poison materials such as Hafnium or B¹⁰-bearing stainless steels necessitated a development program. In conjunction with the MTR Engineering Research Branch a test program was initiated to investigate the fabrication techniques and performance of sprayed metal sections. The test results are reported in part in (12); however, additional testing has been conducted on the stainless steel base-sprayed cadmium-sprayed stainless steel test specimen No. 1. The results are tabulated as follows:

TABLE 8AUTOCCLAVE TEST RESULTS FOR STAINLESS STEEL-SPRAYEDCADMIUM-SPRAYED STAINLESS STEEL SPECIMENS

<u>Sample No.</u>	<u>Autoclave Operation</u>		<u>Remarks</u>
	<u>Hours In</u>	<u>Temp °F.</u>	
1	19	380	No Change
	286	405	" "
	1/2	610	" "
	288	500	" "

Metallographic sections were prepared and photographs of the test section are shown in Figures 26 and 27. The section shows no indication of failure and therefore is considered suitable for operation at 400°F.

3. Corrosion Studies on Aluminum Structural Materials (R. E. Heffner)

The use of an aluminum core in the Spert - II reactor at maximum temperature conditions (400°F) presents a problem in material selection for the reactor internals. Experimental requirements necessitate the use of aluminum for the internal structure if at all feasible. A literature search on the corrosion of aluminum at 400°F indicated that the aluminum alloys are not predictable as to corrosion rates in this region although the alloys in general were somewhat better than type 1100 aluminum.

A number of corrosion tests are being run as only a limited amount of work has been done in this temperature range and the reported results are somewhat contradictory.

While testing metallized control rod sections in an autoclave at 400°F, it was observed that the 6061 aluminum alloy became coated with a considerable amount of white corrosion product after 90 hours of exposure. A chemical analysis of the corrosion product verified that the white substance was aluminum oxide. The corrosion was general in nature and no pitting was observed.

FIGURE 26
(57-4393)

**TEST SPECIMEN NO. 1 - STAINLESS STEEL BASE-
SPRAYED CADMIUM - SPRAYED STAINLESS STEEL**

FIGURE 27
(57 4396)

TEST SPECIMEN NO. 1 - METALLOGRAPHIC SECTION

Further testing of 6061 aluminum alloy and 1100 aluminum alloy has been started. The initial run of 86 hours at 400°F in demineralized water has been completed. The 6061 aluminum alloy showed no pitting and only a very light coating of gray aluminum oxide. The 1100 aluminum was somewhat similar except that the gray oxide was more pronounced. Neither of the samples showed adverse effects due to the first run. A 240-hour run at 400°F is now in progress on the 6061 and the 1100 aluminum samples.

Several major producers of elastomeric compounds were contacted for recommendations for the material to be used as gaskets for the Spert-II vessel. The architect-engineer had specified silicone rubber, which was considered to be unsatisfactory for the vessel closure gaskets. Although only one company recommended a specific material, several companies enclosed samples of their materials for testing since 400°F in water service was beyond the scope of their own test facilities.

In all, four special silicone rubbers, two samples of butyl rubber, two samples of Buna N (Chemigum), and one sample of neoprene were submerged in deionized water in an autoclave at 400°F for varying lengths of time up to a maximum of 96 hours. Most tests were made of the order of 48 hours.

None of the elastomers which was tested possessed the qualities required for gaskets for the Spert - II reactor. The silicone rubbers were the least satisfactory of all elastomers tested.

On the basis of these tests it was recommended that stainless steel tubular gaskets be incorporated in the Spert - II vessel design.

4. Flow Studies - Type "B" Fuel Assembly (J. A. Norberg)

Flow tests were conducted on the Spert type "B" fuel assembly in the MTR hydraulic test loop. The objective of the study was to determine the safe flow rates that could be maintained through the assembly without plate buckling. Of secondary importance was the total pressure drop versus flow relationships. The pressure measurements were obtained from static tubes inserted in and around the fuel assembly.

Figure 28 shows the location of the pressure points with relation to the assembly. The overall pressure drop versus flow is plotted in Figure 29. On the basis of the tests, the recommended maximum flow rate is 300 gpm at the condition tested. The overall pressure drop for flows above 100 gpm can be described to $\pm 20\%$ by the following empirical equation:

$$\Delta P = 1.78 \times 10^{-3} (\text{gpm})^{1.68} \quad (\text{at } 80^\circ\text{F and atmospheric pressure})$$

LOCATION OF STATIC
PRESSURE POINTS
TYPE "B" ASSEMBLY

FIGURE 28
(57-4634)

SPERT TYPE "B" FUEL ASSEMBLY
 OVERALL PRESSURE DROP
 VERSUS FLOW RATE
 80° F.

FIGURE 29
 (57 4636)

C. SPERT - III

1. Calculation of the Heat Generation in the Thermal Shields and Vessel Wall (V.C. Kobold)

The heat generation within the various thermal shields and the pressure vessel wall of the Spert - III reactor has been calculated. A detailed analysis was performed to check the preliminary design estimates and evaluate the adequacy of the shield design.

In the calculation of the heat generation the contributions due to the absorption of gamma photons originating from fission, and from neutron capture within the core, reflector, and shield materials are considered. The inelastic scattering of fast neutrons in the steel of the shields and vessel wall was neglected. The neutron flux values employed in the study⁽¹³⁾ were calculated by the MTR Theoretical Physics section:

A cross section of the Spert - III reactor lattice showing the thermal shield layout is shown in Figure 30, and a graphical representation of the calculated heat generation is shown in Figure 31.

2. Calculation of Thermal Stress in the Core Skirt and Thermal Shields (V.C. Kobold, T. R. Wilson)

Preliminary calculations of the temperature distribution and thermal stress in the core skirt and thermal shields have been performed. These preliminary calculations were based on the heat generation rates as shown in Figure 31 with the reactor operating at 60 Mw and with the reactor outlet coolant temperature at 650°F.

According to an approximate formula reported by Langer⁽¹⁴⁾ the stress at any point in the wall is:

$$\sigma_x = \frac{E\alpha}{2(1-\mu)} \Delta T \left(1 - \frac{2x}{t} + \frac{g}{k\beta^2}\right) \left[\frac{1 + \frac{2}{\beta t} - \frac{2x}{t}}{e^{-\beta t} - 1} + 2e^{-\beta x} \right]$$

- σ = Stress (psi)
- E = Modulus of elasticity (psi)
- α = Linear coefficient of thermal expansion (in/in°F)
- μ = Poisson's ratio
- T = Temperature difference between inner and outer wall (°F)
- x = Distance from inner wall (in.)
- g = Heat generation per inch volume (BTU/hr in³)
- t = Wall thickness (in.)
- k = Thermal conductivity (BTU/hr in °F)
- β = Linear absorption coefficient (in.⁻¹)

The results of the calculations are plotted in Figure 32. The calculations have been made under the assumption that the inner and outer walls of the shields are at the same temperature. This assumption is essentially valid for Spert - III since the inner and outer surfaces of the shields

SPERT III REACTOR THERMAL SHIELD LAYOUT

FIGURE 30
(57-4313)

FIGURE 31

(57-4974)

FIGURE 32
(57-4638)

are exposed to water at the same temperature and velocity. The stresses obtained are tensions, or compressions, in the plane of the shell plate. In any cylindrical shell the tangential stress is exactly equal to the axial stress at the two surfaces, and for small curvatures this condition is approximately true throughout the plate thickness. Therefore, the stress values which are given can be used for both axial and tangential stresses when calculating combined stresses. The formula is considered to be accurate within 0.5% for cylindrical shells whose thickness-to-diameter ratio does not exceed 1:10.

Additional calculations are in progress both for stresses in the thermal shields and within the vessel wall.

3. Hydraulic Calculations and Flow Tests (J. A. Norberg)

Preliminary hydraulic calculations for the core components at temperatures of 50°F, 450°F, and 650°F and the corresponding saturation pressures, have been completed. The calculations will be utilized to design flow orifices for the core filler pieces, and to determine the flow distribution in the core for various operating parameters such as core loading, coolant flow rates, and heat transfer rates.

Figure 33 shows the calculated overall pressure drops versus flow for each component of a Spert - III core with the average bulk coolant temperature at 650°F. Using these curves, the calculated flow distribution in the core with an average bulk coolant temperature of 650°F, a total flow of 20,000 gpm, and a core loading of 52 fuel assemblies including control rod fuel section (30.8 kg U²³⁵), is as follows:

TABLE 9

CALCULATED FLOW DISTRIBUTION IN SPERT - III CORE

<u>Description of Component</u>	<u>Number of Each</u>	<u>Flow per * Component, gpm</u>	<u>Total Flow gpm</u>
Fuel Assembly (type 1-S)	40	387	15,480
Fuel Assembly (type 2-S)	4	275	1,100
Control Rods	8	275	2,200
Transient rod slots	1	215	215
Filler (type 4F)	16	38	608
Filler (type 3F)	8	27	216
Filler (type 2F)	4	24	96
Filler (type 1F)	4	22	88
Total			20,003 gpm.

* Includes the interstitial space between components.

The calculated overall pressure drop through the core under the above conditions is 8 psi.

Flow studies have been conducted on a prototype stainless steel fuel assembly, type 1S, to verify the calculations. The prototype assembly was fabricated by shop personnel and is identical to the Spert - III

SPERT III CORE COMPONENTS
OVERALL PRESSURE DROP
VERSUS FLOW RATE
650°F.

FIGURE 33
(57 4635)

assembly with the exception of a center reinforcing bar. The flow tests were conducted in the MTR Hydraulic Flow Test Facility with the assistance of the MTR Engineering Research Section. Measurements were made of the pressure distribution throughout the assembly over the operating flow range which is anticipated for the Spert - III reactor. The basic design of the flow test facility limited the operating conditions to room temperature and atmospheric pressure. The pressure measurements were obtained from static tubes inserted in and around the fuel assembly.

Figure 34 is a sketch of the fuel assembly and indicates the location of the pressure points in the assembly. The overall pressure drop versus flow is shown in Figure 35, and a channel pressure profile at position C is shown in Figure 36. For the anticipated flow rates up to 400 gpm, the greatest pressure difference between any two consecutive channels is well within the safety limits of the plate strength. The average overall pressure drop as a function of flow at the temperature and pressure conditions of the test is described by the empirical equation:

$$P = 1.11 \times 10^{-4} (\text{gpm flow})^{1.995},$$

which is obtained from Figure 35.

4. Control Rod Drive Design and Testing Program (R. S. McPherson)*

A major effort of the Engineering section has concerned the design and testing of a Spert - III control rod drive mechanism which can be used interchangeably for either a pair of control rods or for the transient rod.

A prototype drive mechanism and dummy transient rod assembly and a drive test facility have been fabricated. The drive mechanism consists of a three-speed, 60 cycle, 3 ϕ , 1 hp drive motor, a gear box, a drive screw section, an air cylinder section, a shock absorber, and a drive rod seal unit. A schematic elevation of the control rod drive unit is shown in Figure 37. The drive test facility consists basically of an insulated pressure vessel that duplicates the reactor pressure and temperature and that will also contain a full size mock-up of either the control or transient rods and the corresponding guide tubes. A photograph of the facility is shown in Figure 38.

The assembly is being employed for tests of the general operating characteristics and reliability of the basic drive mechanism design, and in particular, to establish the "scram" and "hold" air pressures to be used for various reactor vessel pressures, the seal leakage rates, and the air scram times for the transient rods and a pair of control rods.

Air pressure below the air piston and/or reactor pressure serve to hold the piston against the ball nut extension. The required pressure, designated "hold" air pressure, has been determined to be approximately 225 psi. The air pressure required above the piston "scram" air, varies from 185 psi when the reactor is at zero pressure to approximately 585 psi with the reactor at full pressure.

* Division Engineering Branch

* STATIC TUBE HOLES

+ TEST JIG

PRESSURE TAPS

x PLENUM STATIC
TUBE HOLES

SPERT III PROTOTYPE FUEL ASSEMBLY STATIC PRESSURE POINTS

FIGURE 34
(57-4646)

SPERT TYPE I-S FUEL ASSEMBLY
 OVERALL PRESSURE DROP
 VERSUS FLOW RATE
 PROTOTYPE FUEL ASSEMBLY
 (NOT REINFORCED)

FIGURE 35
 (57-4633)

SPERT III
 FLOW TEST ON MODIFIED TYPE I-S
 FUEL ASSEMBLY
 CHANNEL PRESSURE PROFILE
 PRESSURE POINT C

FIGURE 36
 (57-4650)

SPERT III CONTROL ROD DRIVE SCHEMATIC ELEVATION

FIGURE 37
(57-4736)

FIGURE 38
(57-2765)

SPERT DRIVE TEST ASSEMBLY

Leakage rate tests of the drive unit seal with the test vessel at 2450 psi and 666°F have shown satisfactory results. A monitored run of 22 hours under these conditions showed an average constant leak rate of only 1.2 lbs of water/hour (design limit - 5 lbs of steam/hour.) However, friction between the drive rod and the drive rod seals is believed to result in excessive load being exerted on the ball screw, and various methods for reduction of the friction are at present being considered. In addition, a study will be made of the feasibility of replacing the present ball screw with a larger size.

From the time that testing was initiated in April 1957 the facility has been actively in use a total of twenty-seven 8 hour periods. The total number of air scrams from various heights using the dummy transient rod is in excess of fifty. Air scram tests of the dummy transient rod from a 43 inch height, using a "scram" air pressure of 585 psi and a "hold" air pressure of 225 psi, have achieved elapsed times of 0.18 sec. This indicates that no difficulty will be experienced in achieving required control rod scram times.

Analysis and correction of several component deficiencies of the original design is in process. Perhaps the greatest difficulty in the development is in producing a reliable shock absorber unit. Initial tests were run using steel plungers operating in a steel cylinder housing with silicone oil as a fluid. Although adequate shock absorbing characteristics were noted, the plungers were found to have a tendency to gall. Further attempts using bronze plungers and silicone oils reduced the difficulty; however, this did not result in adequate reliability as the shock mechanism failed to return to the extended position. Additional clearances between the plungers and cylinders, the use of a fluid which is a mixture of silicone oil and molybdenum disulfide powder, and the replacement of the original return springs with a stronger unit are methods which are hoped will insure adequate operation.

It is expected that in October 1957, testing with a dummy control rod pair replacing the transient rod will be initiated.

5. Mechanical Design Status (R. E. Heffner and R. S. McPherson)

The status of the mechanical design work and the procurement of reactor components assigned to the Spert - III project is summarized in Table 10. With the exception of miscellaneous items such as handling tools, air piping, core instrumentation, etc., the mechanical design and detail drawings have been completed. Wiring diagrams for the reactor control system are currently in progress.

TABLE 10

SPERT-III - MECHANICAL DESIGN AND MATERIAL PROCUREMENT STATUS REPORT

Component Description	% Comp. of Mech. Design	% Comp. of Drawings	% Comp. of Fab- rication	Est. Comp. or Del. Date	Remarks
<u>Control Rod Drive Mechanism</u>	100	100	80	11- 1-57	One Unit Complete and Under Test
<u>Transient Rod Drive Mechanism</u>	100	100	90	10-15-57	
<u>Rod Drive Auxiliary Equipment</u>					
Air Piping (Head)	95	75	0		Requires Reactor Vessel for Installation
Air Control Components	100		Pur. Items	11-30-57	Installation by Phillips Petroleum Company
Reactor Control Wiring	100	10	5	12- 1-57	Installation by Phillips Petroleum Company
Air Compressor	100		Pur. Items		Installed, October, 1957
Emergency Generator	100		Pur. Items		Installation by Phillips Petroleum Company
Rod Drive Motors & Controls	100		Pur. Items	11-15-57	
<u>Core and Internal Structure</u>					
Upper Grid	100	100	40	10-30-57	
Lower Grid	100	100	99	10-11-57	To be Inspected September 23, by Phillips
Core Skirt	100	100	85	10- 4-57	
Core Filler Pieces	100	100	20	12-15-57	
Control Rod Guide	100	100	100		
Transient Rod Guide	100	100	Pur. Items	9-30-57	
Control Rod Yokes	100	100	100		
Drive Rod Couplings	100	100	100		
Upper Grid Bail	100	100	100	3-15-57	
Flow Scoop	100	100	85	10-15-57	Minor Rework Necessary
Control Rods	100	100	15	11-30-57	Awaiting Boron S. S. and Fuel Section
Transient Rod	100	100	15	11-30-57	Awaiting Boron S. S. and Fuel Section
Fuel Assemblies	100	100	Pur. Items	12-15-57	Final Shipment, January 1958
Thermal Shield Instrum.	25	10	0	12- 1-57	Instal. after Receipt of Thermal Shield
Pneumatic Rabbit Facility	10	0	0	3-15-58	Installation by Phillips Petroleum Company
<u>Miscellaneous</u>					
Handling Tools	25	25	0	12- 1-57	
Fuel Storage Rack	100	50	0	11-15-57	Material Ordered
Air Piping (Building)	90	75	0		Installed After Building Completion
Hot Fuel Coffins	100	60	0	3-15-57	
Fuel Flow Test Apparatus	100	100	100		

100000

6. Final Design Report (T. R. Wilson)

Preparation of a final design report on the Spert - III facility has begun. The report is to include complete design details of the facility. In addition to the design report, an operating manual is under preparation and will be issued either as a part of the design report, or as a separate report. At the present time report preparation is limited by the lack of information available from the construction contractor on the individual items of equipment to be supplied.

IX. BIBLIOGRAPHY

The following is a complete list of reports which have been issued by Phillips Petroleum Company relevant to the Spert Project. They are divided into two sections; those which have been issued directly from the Reactor Projects Branch; and those which have been issued from the Theoretical Physics and Applied Mathematics Branch. These are further broken down into groups: Journal and Magazine articles and IDO reports, which are distributed in accordance with TID-4500, Ed. 13, and are generally available.

A. REACTOR PROJECTS BRANCH REPORTS

1. IDO Reports

- a. W. E. Nyer, S. G. Forbes, F. L. Bentzen, G. O. Bright, F. Schroeder and T. R. Wilson, "Experimental Investigations of Reactor Transients", IDO-16285 (April 20, 1956)
- b. S. G. Forbes, F. Schroeder, and W. E. Nyer, "Instability in the Spert - I Reactor - Preliminary Report", IDO-16309 (October 10, 1956)
- c. F. L. Bentzen, "Spert - I Instrumentation", IDO-16316 (March 15, 1957)
- d. R. W. Miller, "Calculations of Reactivity Behavior During Spert - I Transients", IDO-16317 (June 1, 1957)
- e. T. R. Wilson, "An Engineering Description of the Spert - I Reactor Facility", IDO-16318 (June 14, 1957)
- f. G. O. Bright and C. R. Toole, Jr., "Note on the Effect on Reactivity of Water Annuli in Large Void Spaces in the Spert - I Reactor Core", IDO-16382 (July 1, 1957)
- g. F. Schroeder, "Stability Tests with the Spert - I Reactor", IDO-16383 (July 1, 1957)
- h. G. O. Bright, S. G. Forbes, W. E. Nyer and F. Schroeder, "An Elementary Model for Reactor Burst Behavior", IDO-16393 (August 2, 1957)
- i. T. R. Wilson, "The Spert - III Reactor Facility Preliminary Design Report", IDO-16341 (May 6, 1957)
- j. J. C. Haire, "Sub-cooled Transient Tests in the Spert-I Reactor - Experimental Data", IDO-16342 (1957)
- k. W. E. Nyer and S. G. Forbes, "Spert Program Review", IDO-16415 (September 27, 1957)

1. G. O. Bright, "Photographic Investigations of Spert - I Transients", (Film) PTR-131 (December 1, 1956)
(Film on loan basis only)

2. Journal and Magazine Articles

- a. W. E. Nyer, S. G. Forbes, F. L. Bentzen, G. O. Bright, F. Schroeder and T. R. Wilson, "Transient Experiments with the Spert - I Reactor", Nucleonics 14, No. 6 (1956)
- b. S. G. Forbes, F. Schroeder and W. E. Nyer, "Instability in Spert - I", Nucleonics 15, No. 1, (1957)
- c. F. Schroeder, S. G. Forbes, W. E. Nyer, F. L. Bentzen and G. O. Bright, "Experimental Study of Transient Behavior in a Subcooled, Water-Moderated Reactor", Nuclear Science and Engineering: 2, 96-115 (1957)
- d. T. R. Wilson, "Special Power Excursion Reactor Test III", Nucleonics 15, No. 8 (1957)

B. THEORETICAL PHYSICS AND APPLIED MATHEMATICS BRANCH REPORTS

1. IDO Reports

- a. H. L. McMurry, "Temperature Distribution in a Fuel Plate with Exponentially Rising Power Part I - Theoretical", IDO-16214 (March 25, 1955)
- b. H. L. McMurry and Staff, "Nuclear Calculations for Spert - III", IDO-16325 (March 25, 1957)
- c. D. R. Metcalf and R. Haldin, "Three and Two Group Constants for a Spert - III Core", IDO-16326 (April 11, 1957)
- d. C. W. Reich, "Local Thermal Flux Distribution in Spert - III Fuel Elements by the Spherical Harmonics Method", IDO-16327 (March 13, 1957)
- e. D. R. Metcalf, "Central Control Rod Calculations for Spert Reactors", IDO-16330 (March 21, 1957)
- f. G. A. Cazier and B. L. Hanson, "Nuclear Calculations for Spert - III Zirconium Core, IDO-16359 (PTR-110) (October 1, 1956)
- g. D. R. Metcalf and V. C. Kobold, "A Four Region, Two Group Calculations of Spert - III Critical Size and Flux Distribution", IDO-16390 (July 18, 1957)
- h. L. I. Deverall and G. W. Griffing, "Initial Behavior by the Energy Model", IDO-16393 (1957)

- i. L. I. Deverall and G. W. Griffing, "On the Initial Shutdown of the Spert - I Reactor for Periods Greater than 50 Milliseconds", IDO-16404 (1957)

X. REFERENCES

1. T. R. Wilson, "An Engineering Description of the Spert - I Reactor Facility", IDO-16318 (June 14, 1957)
2. S. G. Forbes, "A Non-destructive Method for Fuel Assaying", IDO-16114 (1953)
3. G. R. Keepin, T. F. Wimett, R. K. Zeigler, "Delayed Neutrons From Fissionable Isotopes of Uranium, Plutonium and Thorium", LA-2118 (1957)
4. Glasstone and Edlund, "Elements of Nuclear Reactor Theory", New York: D. Van Nostrand Company (1952)
5. J. C. Haire, "Subcooled Transient Tests in the Spert - I Reactor - Experimental Data", IDO-16342 (1957)
6. F. Schroeder, S. G. Forbes, W. E. Nyer, F. L. Bentzen and G. O. Bright, "Experimental Study of Transient Behavior in a Subcooled, Water-Moderated Reactor", Nuclear Science and Engineering: 2, 96-115 (1957)
7. R. W. Miller, "Calculations of Reactivity Behavior During Spert - I Transients", IDO-16317 (1957)
8. L. I. Deverall and G. W. Griffing, "Initial Behavior by the Energy Model", IDO-16393 (1957)
9. L. I. Deverall and G. W. Griffing, "On the Initial Shutdown of the Spert - I Reactor for Periods Greater than 50 Milliseconds", IDO-16404 (1957)
10. D. R. Metcalf and R. Haldin, "Three and Two Group Constants for a Spert - III Core", IDO-16326 (April 11, 1957)
11. E. P. Wigner and J. E. Wilkins, Jr., "Effect of the Temperature of the Moderator on the Velocity Distribution of Neutrons with Numerical Calculations for H as Moderator", AECD-2275 (1944)
12. "Quarterly Progress Report for MTR Technical Branches", Second Quarter, 1957, IDO-16394 (August 22, 1957)
13. D. R. Metcalf and V. C. Kobold, "A Four Region, Two Group Calculation of Spert - III Critical Size and Flux Distribution", IDO-16390 (July 18, 1957)

14. B. F. Langer, "Approximate Formulas for Calculation of Thermal Stress in Reactor Vessels, Thermal Shields and Thimbles", WAPD-CE-39 (March 11, 1957)