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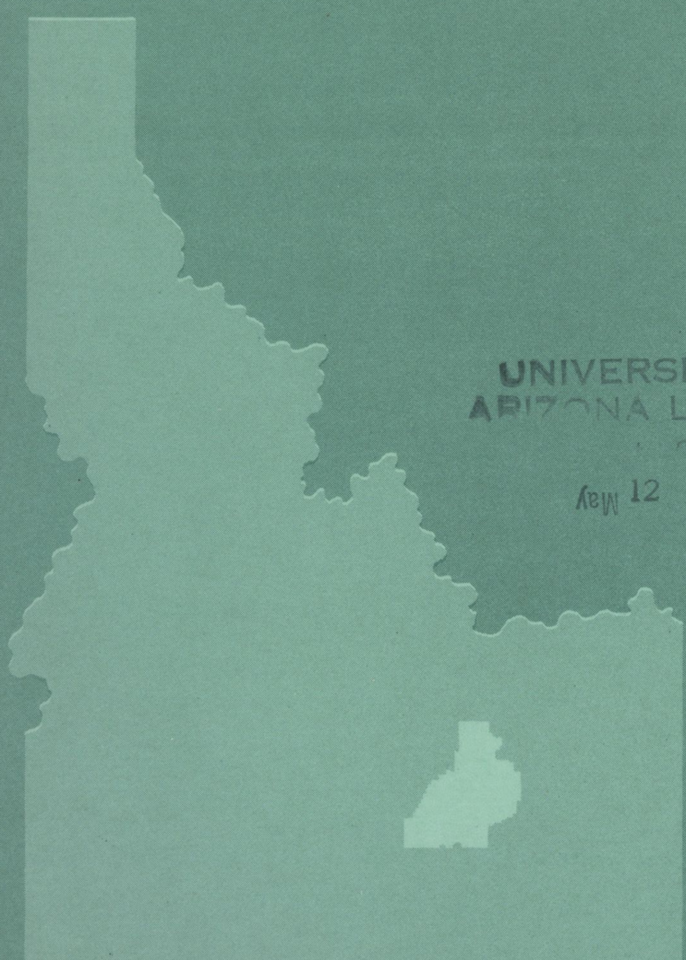
ADDENDUM TO THE SPERT IV HAZARDS SUMMARY REPORT---CAPSULE DRIVER CORE

R. W. Miller, R. K. McCardell, and T. F. Lagier



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by

R. W. Miller
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T. F. Lagier

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Atomic Energy Division
Contract AT(10-1)-205
Idaho Operations Office
U. S. ATOMIC ENERGY COMMISSION

ABSTRACT

The Addendum to the Spert IV Hazards Summary Report was prepared and submitted to the Atomic Energy Commission to explain all important features pertaining to a new pulsed irradiation reactor, the Capsule Driver Core, and to analyze the potential problems and hazards of operating this reactor in the existing Spert IV facility. The report is in the form of an addendum because, with the exception only of the reactor and certain aspects of the control system, all other features discussed in the previous Spert IV Hazards Summary Report still apply. Pertinent mechanical characteristics are explained in detail including changes made in the original Spert IV control system design. Nuclear characteristics of the core as determined both by calculation and experiment are included along with predicted kinetic behavior for stepwise reactivity insertions. Finally, several accident situations are postulated together with preventive measures, and a maximum referent accident is analyzed to obtain numerical estimates of the worst possible radiological hazards. It is concluded that the Capsule Driver Core can be operated as intended and stay within the safety policies of the AEC.

SUMMARY

The following report is an addendum to the original Spert IV Hazards Summary Report, IDO-16689; Spert IV Facility Report, IDO-16745; and other supplemental information provided to the AEC about Spert IV. Given in this report is information about a new pulsed irradiation reactor, the Capsule Driver Core, which is to be installed in the Spert IV facility.

The Capsule Driver Core is water-moderated and -reflected, composed of about 1600 stainless steel clad fuel rods, each containing about 1600 grams of 3 percent-enriched UO_2 powder. The core is built about a steel tube, extending through the center, which accommodates various capsules. The capsules, in general, will contain fuels for transient response testing.

Previous experience and calculations indicate that, when the core is subjected to power excursions, Doppler effects will shut it down automatically and that periods as short as 2 and 3 msec may be executed in the core without risk of damage to the core.

Accidents are postulated which conceivably could occur during the operation and usage of the CDC as a pulse irradiation facility, and from this analysis it is concluded that no credible accident could produce hazards of a greater magnitude than one involving the accidental insertion of 5$ reactivity. This 5$ accident is termed the maximum referent accident and is analyzed to obtain quantitative radiological dose risks.

It is concluded in this report, from the analysis of accidents and their preventive measures, that the risk of operating the Capsule Driver Core is extremely low and within the general safety policies of the AEC.

ADDENDUM TO THE SPERT IV HAZARDS
SUMMARY REPORT -- CAPSULE DRIVER CORE

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I. INTRODUCTION

During the past several years of reactor kinetic testing at Spert, the need has grown for an increased effort in the direction of subassembly testing to balance an extensive integral core testing program. It has been apparent that some of the vital experimental information required for reactor safety can only be obtained in capsules or subassemblies which are inserted into a flux-trap of a "burst facility" reactor which provides short-period power excursions. Completion of the first Spert Testing program of a low-enrichment, water-moderated, oxide core [1, 2, 3] indicated that this type of a core had many of the features which are desirable for a "burst facility" including: a high-integrated flux per unit power density, high fuel and cladding melting points, relatively low neutron lifetime, and particularly, an apparently superior resistance to damage arising from high temperatures and thermal shock.

Since a large quantity of oxide fuel from the N. S. Savannah critical studies was readily available, studies were conducted of this fuel, and these studies resulted in the design of a core with highly satisfactory "burst facility" characteristics. The core has since been designated the Capsule Driver Core or CDC, and it is the hazards of operation of this core for fuels and materials irradiations which is the subject of this document.

At the time of this writing, the CDC has successfully undergone criticality and core augmentation until it contains about 3$ excess reactivity which is believed to be adequate for the intended materials testing program. The CDC also has completed an initial program of "static" nuclear tests (described later in this report) in which many of the calculations were verified and in which the core appeared to have very favorable flux-trap capabilities. The CDC is to be permanently installed in the Spert IV facility where it will undergo next a program of kinetic testing prior to its ultimate use as a "burst facility" for the study of fuels and materials.

This report presents a description of a Capsule Driver Core and an analysis of the possible hazards involved in its operation. Inasmuch as the major change to the Spert IV facility is the replacement only of the core, this report is written as an addendum to the original <u>Spert IV Hazards Summary Report</u> [4] as well as the other supplementary information [5, 6] supplied to the Atomic Energy Commission at the time of the original review of the safety of operation of Spert IV. Information contained in the previous submission to the AEC for safety review purposes which is not pertinent, altered, or negated by installation of the Capsule Driver Core (CDC) in the Spert IV facility is not reproduced in this report except for excerpts or abbreviated discussions which are presented for completeness. This report does describe in detail all facets of the CDC installation which are pertinent to an evaluation of its safety. However, because of the anticipated diversity of in-pile tests, detailed description of capsules and capsule tests is not presented in this report. The philosophy of capsule test procedures and the type review necessary before approval is obtained for each individual test is performed are discussed, however.

II. DESCRIPTION OF FACILITY

This section contains a brief description of the Spert IV facility. A complete description of the facility exists in the report, Spert IV Facility, IDO-16745 by R. E. Heffner et al, which may be consulted for details not discussed here.

Modifications to the facility in order to accommodate the CDC are slight, and those modifications which have import to the present safety analysis (viz, the control system) are detailed in Appendix A.

1. DESCRIPTION OF THE SPERT IV FACILITY

The Spert IV reactor site is located approximately 1/2 mile from the Control Center building and approximately 3/4 mile from the nearest reactor facility (Spert III). Spert IV consists of a high-bay main reactor building housing two large, open-top pool tanks. The reactor is to be located in the north pool tank, and the south pool tank will be used for fuel and irradiated test material storage. Two low-bay wings of the building contain an instrumentation room, an electronics work area room, an office, a mechanical work area room, a change room, a process control room, and a furnace and equipment room. Nuclear operation of the Capsule Driver Core will be carried out by remote control from the control console located in the Control Center building.

2. ENGINEERING DESCRIPTION OF THE CDC

The Capsule Driver Core is a low-enrichment, rod-type oxide core. Figure 1 is a cutaway view of the CDC, and Figure 2 depicts the top of the core. The core is divided by twelve aluminum rod-guide crosses which house the eight

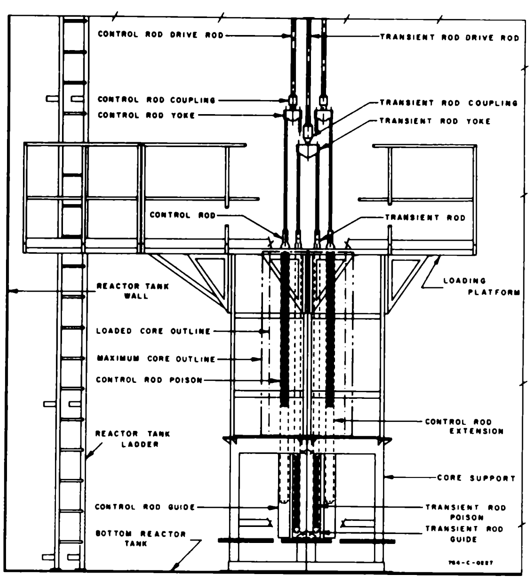

Fig. 1 Capsule Driver Core, elevation view in the Spert IV reactor tank.

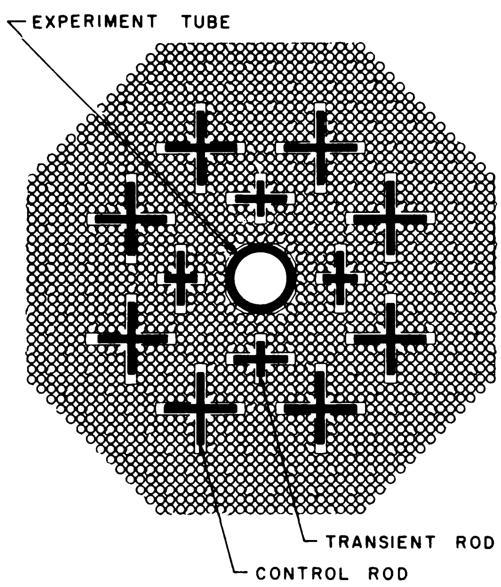

Fig. 2 Plan view of the Capsule Driver Core.

cruciform-shaped control-rod blades and the four cruciform-shaped transient-rod blades. A cylindrical, stainless steel experiment tube in which test samples will be placed for kinetic testing lies along the vertical centerline of the core. Figure 3 shows a photograph of the CDC support structure, the control and transient rods, and the experiment tube as assembled in the Spert I reactor vessel for initial critical loadings and static measurement experiments. Principal nonnuclear characteristics of the Capsule Driver Core are summarized in Table I.

2.1 Fuel

The fuel rods comprising the Spert IV Capsule Driver Core are those previously used in the Babcock and Wilcox, N. S. Savannah critical assembly [7]. The fuel rod is a six-foot-long, welded-seam, stainless steel tube, containing low-enrichment UO_2 powder, swage-compressed to an effective density of approximately 85 percent of the theoretical density of UO_2. The rods contain nominally 1600 grams of UO_2 (3 percent enriched) and are clad with a nominal 28 mils of type-304 stainless steel.

2.2 Control Rods and Transient Rods

Reactor control is accomplished by the use of four control rod units, each consisting of two cruciform-shaped, neutron-absorbing blades, connected by a yoke assembly and shaft to the armature of a coupling magnet. The four transient rods, which are ganged and used for step insertions of reactivity, are similar to the control rods, except that their poison sections are normally below the active core region. The transient rods are raised to decrease reactivity and dropped to increase the reactivity of the core. Figure 4 is a photograph of the CDC support structure, as assembled in the Spert I Reactor Vessel, showing the control- and transient-rod yokes. The poison sections of the control and transient rods are 12 weight percent natural boron as boron carbide in an aluminum matrix and have follower sections which are constructed of aluminum. These follower sections serve principally to guide the various rods through their respective slots.

The basic control-rod drive system originally used in Spert IV will be utilized for the Capsule Driver Core. This drive system consists of electromagnetically coupled control rods driven by a single variable-speed, motor transmission combination which allows only ganged movements of the electromagnets. An adjustable spring is provided for separation of the magnetic couple and for initial acceleration when the control rods are scrammed. The drive unit is mounted on the movable bridge spanning the reactor vessel. By changing the variable-speed transmission gear-head, the control rod maximum withdrawal rate can be varied between 0 and 12 inches/minute. A total control rod travel of about 24 inches will be used. The bottom of the control-rod poison section at the lower limit position is about 11 inches above the bottom of the fuel rods.

As mentioned previously, the transient rod consists of four cruciform-shaped blades, operated as a bank and connected together by a yoke located above the core. From the transient-rod yoke, a shaft extends upward to a piston-cylinder drive mechanism which accomplishes all transient rod motion and positioning by compressed gas. A remotely operated system of pressure valves allows the operator to either raise the transient rod to its upper limit (by placing an over pressure below the transient rod piston) or to accelerate

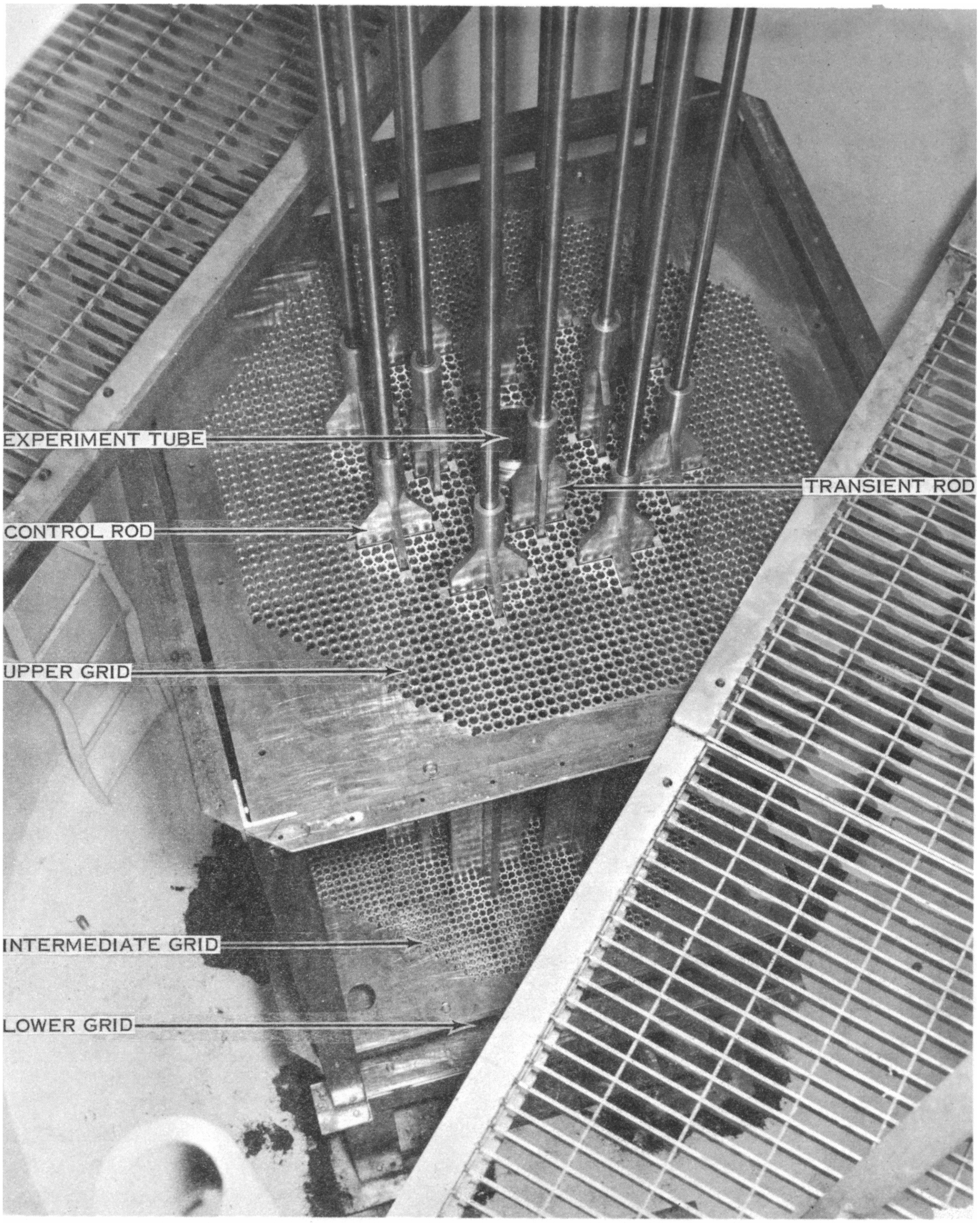

Fig. 3 CDC support structure assembled in the Spert I reactor vessel. View showing intermediate grid and rod-guide penetrations.

TABLE I

CHARACTERISTICS OF SPERT IV CAPSULE DRIVER CORE

Vessel	
Type	Free standing cylindrical tank
Composition	Welded, rolled, type-304 stainless steel plate
Size	25 feet high, 20 feet diameter Top 23 feet, 5/16 inch Bottom 2 feet, 5/8 inch
Bottom thickness	1/2 inch
Design pressure	Below gate: 25-foot hydrostatic load plus 50-psi static surcharge Above gate: 6-foot hydrostatic
Core	
Moderator reflector	Light water
Pitch	0.663 inch, square pitch
Nonmoderator-to-moderator ratio	0.807
Fuel	
Type	Compressed UO_2 powder in rod form
Average length of fuel rods[a]	72.8 inches
Active length	67 ± 1 inch
Fuel rod OD	0.500 inch
Clad thickness	0.028 inch
Enrichment	3 wt% U-235
UO_2 density	9.28 g/cm^3
Mass of UO_2 per fuel rod	≈ 1600 grams
Mass of U-238 per fuel rod	≈ 1364 grams
Cladding	Cold-worked, type-304 stainless steel
Control Rods	
Number	4 gang-operated with two cruciform-shaped sections per control rod
Composition	12 wt% boron in a sintered dispersion of B_4C in Al
Control Rods (Cont.)	
Withdrawal rate	0 to 12 inches/min.
Scram time	Approximately 300 msec from upper limit
Length of poison section	61 inches
Total length	107 inches
Width	4-3/4 inches
Thickness	0.245 inches
Transient Rods	
Number	4 gang-operated, cruciform-shaped blades
Composition	12 wt% boron as B_4C in Al
Length of poison section	28 inches
Total length	118-1/2 inches
Cruciform dimensions	2-1/8 inches x 3-3/8 inches 0.245 inch thick
Experiment Tube	
Tube material	Type-304 stainless steel
Length	6 ft
OD	4.779 inches
ID	3.750 inches
Wall thickness	0.514 inch
Design pressure	Static yield, 8000 psi Static bursting, 18,000 psi

[a] Typically, the total fuel rod lengths vary from 72.4 inches to 73.1 inches.

Fig. 4 CDC support structure assembled in the Spert I reactor vessel. View showing control and transient rod yokes.

the transient rod to its lower limit (by allowing "hold" air beneath the piston to vent to the atmosphere so that high pressure "fire" air above the piston causes the rod to accelerate downward). The transient rod has only two static positions: upper limit (poison fully inserted) and lower limit (poison fully removed). The transient rod is held in the upper limit position by an air-operated mechanical latch which is designed to prevent an inadvertent rod drop. As a transient is initiated, the transient rod latch is opened and the transient rod is accelerated by an air-piston arrangement. The full stroke of the transient rod is about 36 inches, with the last 8 inches of rod travel being decelerated by a hydraulic shock absorber acting near the upper end of the assembly.

A multisection timer unit with associated relays is used to initiate selected experimental functions in a given sequence during a reactor transient, ie, the ejection of the transient rods, the starting and stopping of various recording equipment, and, as an experimental convenience, the scramming of the control rods at the termination of the transient test. Because of the philosophy of operation and the type of operation, no automatic scram circuits are used; reactor shutdown can be initiated by the sequence timer or by manual scram action.

2.3 Core Support Structure

The CDC support structure consists of three aluminum grid plates, an aluminum-core support base plate and the various grid supports, support ties, and the core support base structure. Each of the three identical 3/8-inch-thick aluminum grid plates has provisions for a maximum loading of 2172 fuel rods through holes which are 0.663 inch center-to-center on a square pitch. Smaller holes, interpositioned between the fuel rod holes, allow for free convection flow of coolant. The plates are positioned at the top and bottom of the fuel rods and in the neighborhood of the peak of the axial flux distribution (about 17-3/4 inches above the bottom of the core). The fuel rods may expand upward in the axial direction without bowing, but the grids prevent any radial movement of the fuel rods under transient conditions.

The core support plate, which is about one inch below the lower grid plate, contains no fuel rod holes, but it does provide 1768 flow holes, each 1/4 inch in diameter, and slots for the control rod and transient rod guides. The core support plate provides the base support for the fuel rods.

2.4 Experiment Tube

The experiment tube, which extends the length of the core, consists of a six-foot-long, 304 SS cylinder with inner and outer diameters of 3.750 and 4.779 inches, respectively. The tube, which has a welded end plate on the bottom with a two-inch hole to allow for possible flow service to the capsule, is attached to the lower grid support. During operation the hole will normally be plugged at the bottom with an insert; and the top end of the experiment tube, which is threaded, may be capped when necessary. Static yield and bursting pressures of the experiment tube are calculated to be 8000 psi and 18,000 psi, respectively. The static bursting pressure corresponds to 50 percent strain of the tube at failure.

2.5 Instrumentation

The Capsule Driver Core instrumentation includes neutron detection, reactor bulk-water temperature, reactor water level, and radiation detection instrumentation.

In addition to the normal complement of neutron-detecting instruments used for critical and subcritical operation of the reactor, a series of neutron-sensitive chambers will be positioned at varying distances from the core to provide transient power level measurements. Similar measurements may be made within the core.

The gamma radiation levels, directly over the reactor vessel and at other points in the reactor area, are measured by high range gamma-sensitive chambers. Air in the reactor building is continually sampled and monitored for gaseous or particulate radioactive material by means of constant-air-monitor instruments. The signals from all of the aforementioned detectors are transmitted to recorders in the Spert IV reactor control room at the Control Center. Warning bells at the reactor building are actuated whenever the radiation level measured by any of the instruments exceeds a predetermined set point

Transient temperature measurements of specially selected in-core samples, such as fuel-rod-cladding surfaces, will be obtained through the use of attached thermocouples. Pressure transducers may be located within the core region to permit measurements of the transient steam pressures which may occur during the short-period power excursion tests. An evaluation of the effects of pressure and temperature on selected components such as the reactor vessel and the fuel-rod cladding will be facilitated through the use of strain gauges attached to these surfaces.

III. NUCLEAR PROPERTIES OF THE CAPSULE DRIVER CORE

Calculations were performed to design and predict the performance of the Capsule Driver Core, and these included evaluation of both the static characteristics and the kinetic response of the driver core together with the expected response of selected test fuels. Calculation of the static characteristics of the core were performed using the one- and two-dimensional diffusion theory codes, FOG [8] and PDQ-4[9]; and the kinetic response of the core was evaluated using the calculational model derived by Spano [10]. A CDC statics test program has been performed in the Spert I facility to provide experimental data to complement some of the calculations. This section contains both the experimental and calculated static characteristics of the core, the calculated kinetic response of the core, and the expected response of selected test fuels.

1. STATIC CHARACTERISTICS OF THE CDC

The experimental determination of the static characteristics of the CDC was completed in the Spert I facility. Objectives of the experiments performed during this program were: (a) to determine a core loading suitable for the performance of the subassembly program, and (b) to obtain operational data on the static characteristics of the core required for planning safe and efficient operation of the reactor. In order to meet these objectives, the following experiments were performed:

(1) critical and operational core loading

(2) control- and transient-rod worth experiments

(3) reactivity coefficient measurements

(4) measurement of flux distributions

(5) reduced prompt neutron lifetime measurements

(6) measurement of the figure-of-merit for two types of test fuel (The figure-of-merit is defined as the ratio of the maximum power density in the test fuel to the maximum power density in any fuel rod of the driver core. In both cases, the power densities are taken as averages over the cross section of the fuel.)

Results of experiments are given in the following sections.

1.1 Critical and Operational Core Loading

Initial criticality was obtained with 1431 fuel rods loaded in the geometric arrangement shown in Figure 5 and with the control rods withdrawn to upper limit (36.4 inches above the bottom of the fuel rods). Loading then continued until the available excess reactivity in the core, as indicated by the control-rod critical position, was about 3.0$. Loading operations were terminated with 1659 fuel rods located in the geometric core arrangement shown in Figure 5. Control-rod-bank critical position for the operational core was 27.6 inches above the bottom of the fuel rods. A shutdown margin from critical of about 9$ was determined by the integral-count, rod-drop technique [11].

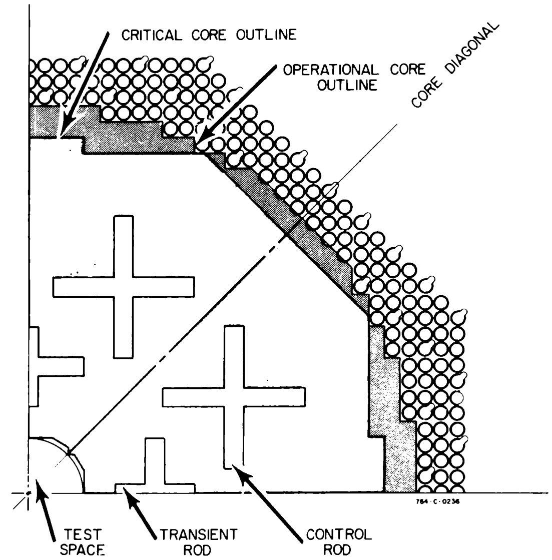

Fig. 5 Quadrant of the CDC showing critical and operational core outlines.

1.2 Control- and Transient-Rod Worth

Reactivity calibration of the ganged control rods was obtained over the full range of rod travel, from the cold clean critical position to upper limit, by using a gadolinium nitrate solution for a uniform reactivity shim and the period measurement technique for reactivity evaluation. Figure 6 shows the differential control-rod-worth curve obtained from the reactivity measurements together with the integral rod-worth curve which indicates an available excess reactivity of 2.9$ from critical for the operational core loading.

The transient rod has only two static positions, upper and lower limit, and, consequently, a measurement of the incremental transient-rod worth was not performed. However, since the reactor would not go critical with the transient rod at upper limit, the reactivity worth of the transient rod is greater than the 2.9$ excess reactivity of the core.

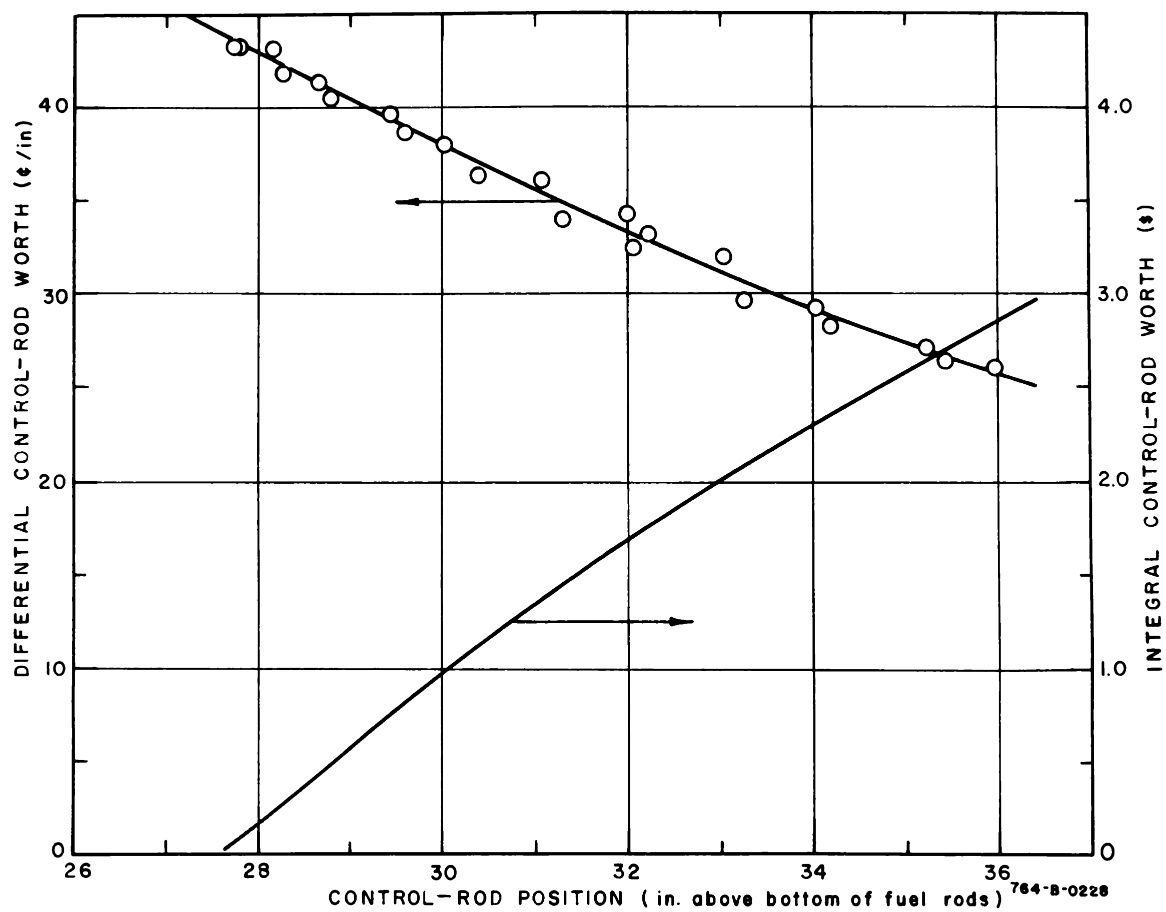

Fig. 6 CDC integral and differential control-rod-worth curves.

1.3 Reactivity Coefficients

The purpose of the reactivity coefficient measurements was to obtain data which may be used to evaluate dynamic reactivity effects caused by moderator boiling and temperature changes. Experiments were performed to measure void coefficients in the core and test space and to measure an isothermal temperature coefficient. Reactivities were measured from the change in critical position of the calibrated control rods.

1.31 Void Coefficients. Void coefficients were measured for both uniform and nonuniform void configurations in the core. Nonuniform void coefficients were measured in cylindrical regions which extended radially outward from the test space to include 6.3, 15, and 20 percent of the core. Table II gives the results of the core void coefficient measurements.

TABLE II

UNIFORM AND NONUNIFORM VOID COEFFICIENTS IN THE CDC

Fraction of Core Included as Voided Region	Void Volume in Voided Region (liters)	Void Fraction of Voided Region	Reactivity Loss (¢)	Void Worth	
				¢/% Void	¢/cc
100%	22.7	1.19%	38.5	-32	-0.0017
6.3%	5.56	5.05%	15.0	- 3.0	-0.0027
15%	12.1	5.05%	34.0	- 6.7	-0.0028
20%	16.2	5.05%	45.5	- 9.0	-0.0028

Measurements were performed to evaluate the worth of various void configurations in the test space. Radial void coefficients (averaged over the core length) were measured using void simulants which consisted of aluminum tubes placed in the test space extending the full length of the core. Nine radial configurations were used and the coefficients, so measured, ranged between + 0.012 ¢/cc to + 0.018 ¢/cc, indicating a weak radial dependence. The void coefficient was, however, strongly dependent upon the axial location of the voids, as shown in Figure 7, and ranged from near zero to a maximum of 0.06 ¢/cc near the axial flux peak. These data were obtained with a cylinder of polystyrene which filled the test space over a 4 inch length and which was secured at various axial positions for the measurements.

1.32 Temperature Coefficient. The isothermal temperature coefficient for the CDC was determined from the change in excess reactivity of the core, as a function of reactor water temperature, over the temperature range from 17.4 to 45.5°C. Figure 8 shows the experimental data together with a solid curve which is a quadratic least-squares fit to the experimental data. Differentation of the quadratic equation, which defines the solid curve shown in Figure 8, gave the temperature coefficient shown in Figure 9, which indicates a variation in the coefficient from about -0.46¢/°C at 17°C to -1.52¢/°C at 45°C.

1.4 Flux Distributions

Steady state neutron flux distributions in the core and test space were determined from the irradiation of 0.040-inch-diameter cobalt wire and also

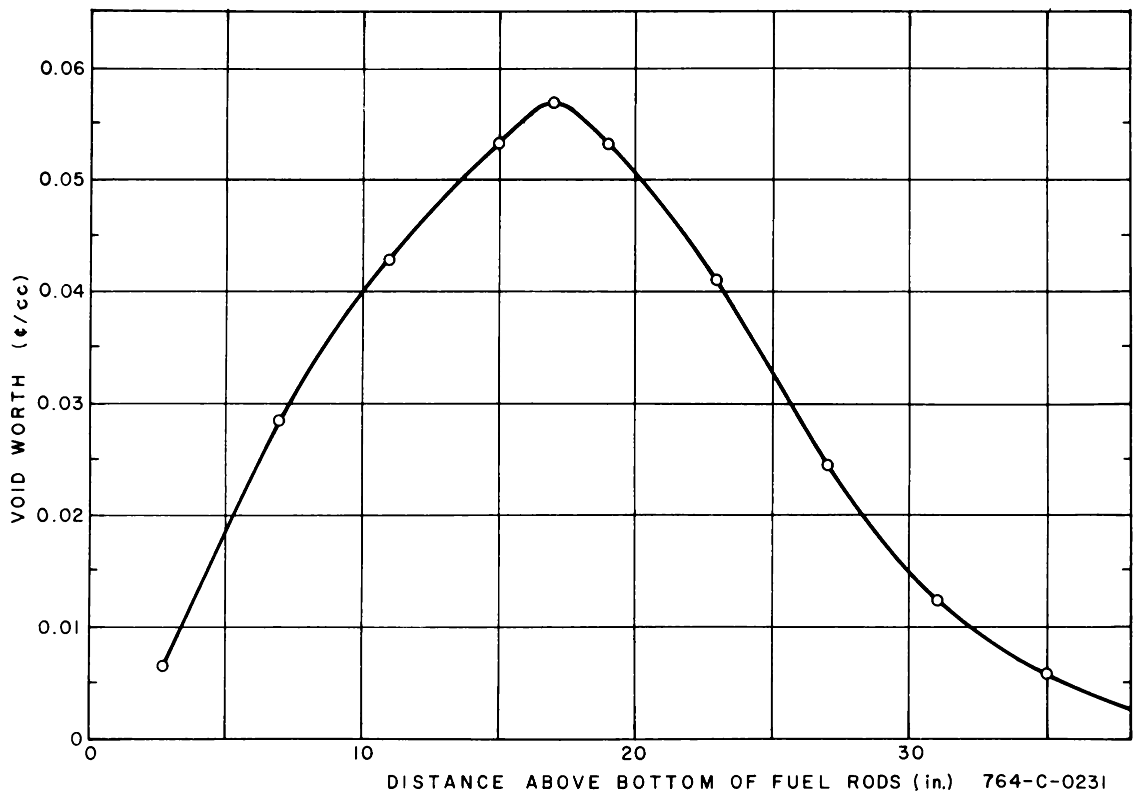

Fig. 7 Void coefficient of the CDC test space.

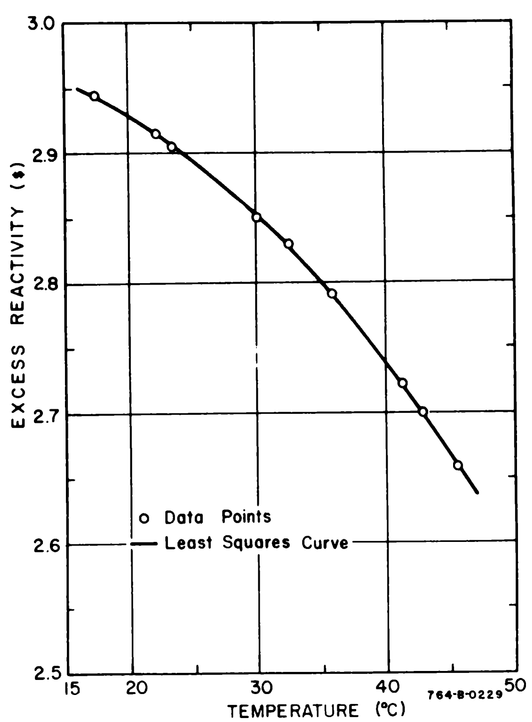

Fig. 8 CDC excess reactivity versus system temperature.

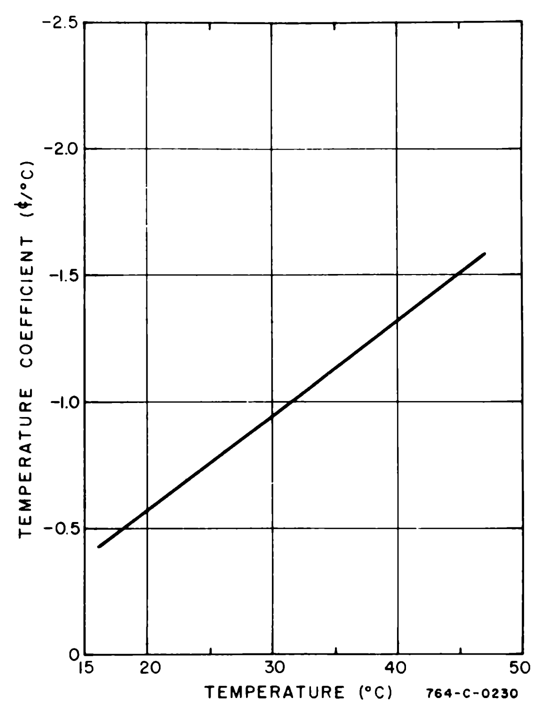

Fig. 9 CDC temperature coefficient.

by 0.156-inch-diameter by 0.005-inch-thick gold foils positioned at selected locations in the core.

Axial distributions in the core were obtained by the irradiation of cobalt wires located in 39 different core positions. Figure 10 shows typical axial flux distributions at three of these core positions together with the calculated thermal flux distribution. The precision of the measurements does not allow exact determination of the axial flux peak; however, the results indicate that the intermediate grid is very near the static axial flux maximum. The radial flux distribution in the core was also obtained from irradiation of a cobalt wire. The results of this measurement is shown in Figure 11 together with a calculated thermal flux distribution.

Radial and axial flux profiles in the experiment test space were determined from irradiation of gold foils for three test configurations: (a) test space water-filled; (b) test space water-filled and containing a water-filled 2.375-inch-OD aluminum capsule having a 0.154 inch wall thickness; and (c) test space water-filled and containing the water-filled capsule and a CDC fuel rod.

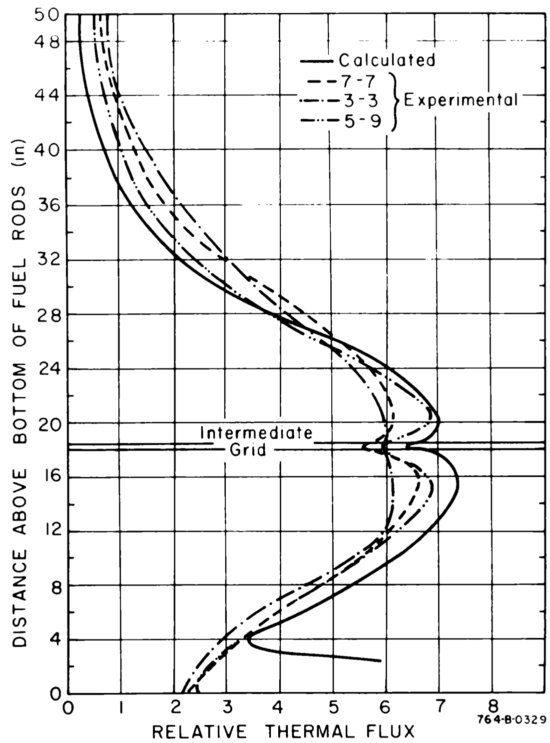

Fig. 10 Axial flux distribution in the core. Numbers with experimental data identify fuel rod locations referenced to the core center. Locations 7-7 and 3-3 are on a core diagonal (See Figure 5).

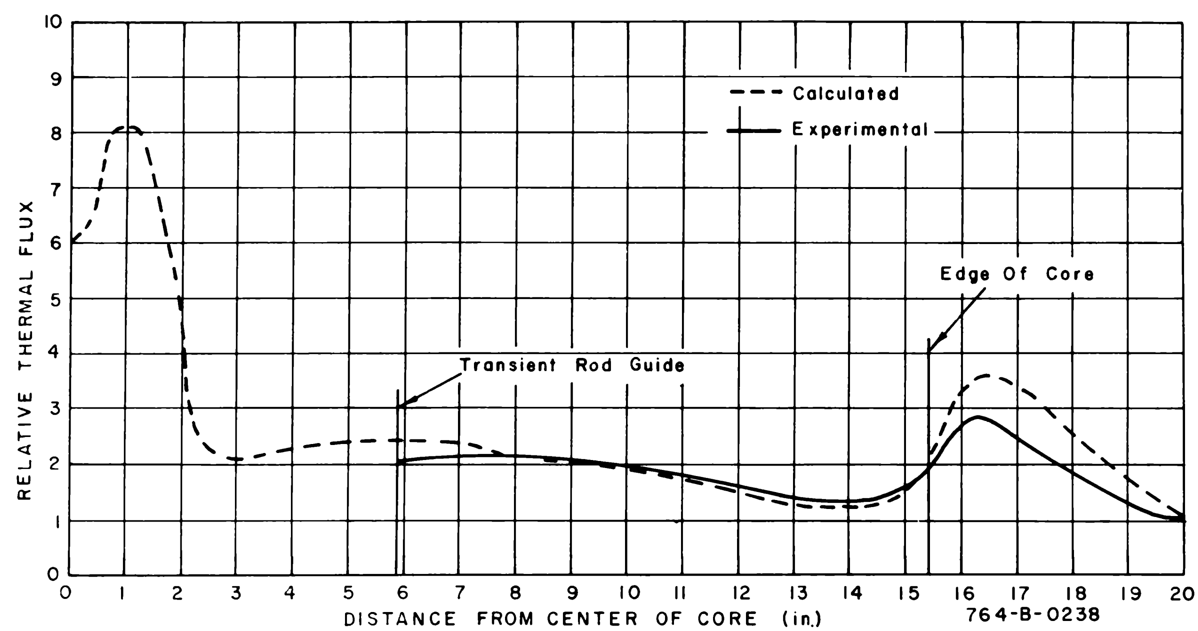

Fig. 11 Radial flux distribution in the core.

Both the capsule and fuel rod were centered in the test space. Figures 12 and 13 show the measured axial and radial profiles, respectively, for the three configurations. The insertion of a CDC fuel rod in the test space caused a flux depression of about 35 percent and about a 1/2-inch change in the critical position of the control rods. Axial flux distributions, shown in Figure 10, were measured at the center of the test space for configurations 1 and 2 and were measured on the outside of the fuel rod (0.25 inch from the test space center) for configuration 3.

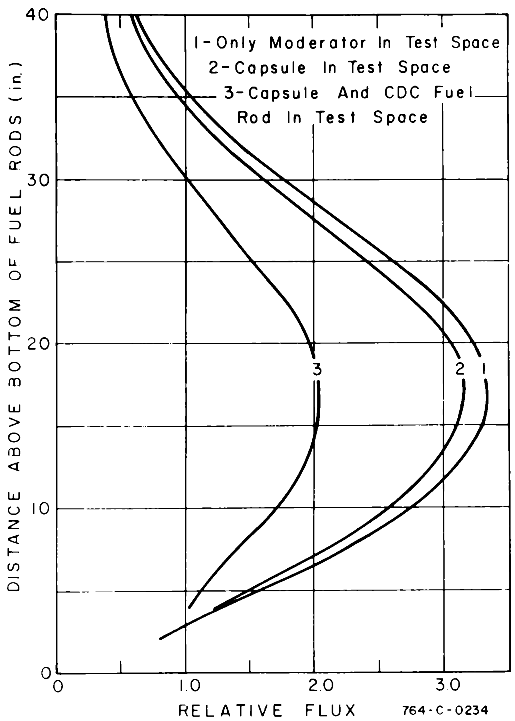

Fig. 12 Axial flux distributions in the CDC test space. A relative flux of 1.0 corresponds to the maximum flux measured in the core.

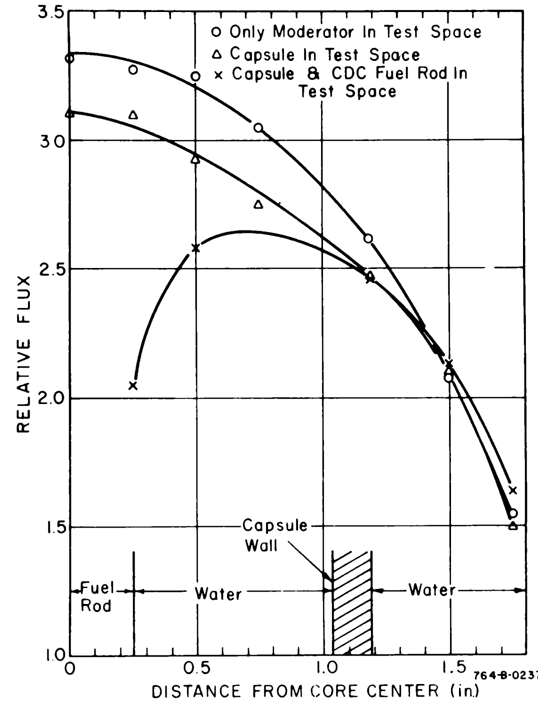

Fig. 13 Radial flux distributions in the CDC test space. A relative flux of 1.0 corresponds to the maximum flux measured in the core.

Flux distributions, shown in Figures 10 through 13, are in general agreement with calculated distributions obtained using the one- and two-dimensional diffusion theory codes, FOG and PDQ04. For example, the ratio of the maximum thermal flux measured on the outside of a CDC fuel rod in the test space to the maximum thermal flux measured in the core was 3.3 as compared to a calculated value of 3.2.

1.5 Reduced Prompt Neutron Lifetime, ℓ/β_{eff}

The experimental method employed in the determination of ℓ/β_{eff} was the power spectral density technique [12] which has been successfully used previously at Spert [13]. Analysis of the experimental data yielded a value of 4.45 msec for the reduced prompt neutron lifetime of the CDC.

1.6 Figure-of-Merit

Figure-of-merit experiments were performed for a CDC fuel rod and for a 4.8 percent-enriched PL-2 [14] fuel rod. In each experiment, a CDC energy release of about 1 MW-sec ($\approx 0.4 \frac{\text{watt-sec}}{\text{g of } UO_2}$) was provided for fuel rod irradiation. Figures-of-merit were determined from isotope analysis of segments from both the test fuel rod and a CDC fuel rod which was located near the driver core hot spot. Four isotopes were analyzed: Ba-140, La-140, Mo-99, and Ru-103. Experimental results obtained from these isotopes were averaged and are listed in Table III together with calculated values for both single fuel elements and fuel element clusters. Positive reactivity additions to the CDC, caused by the insertion into the test space of a CDC and a PL-2 fuel rod, were measured to be 24 and 33¢, respectively.

TABLE III

FIGURES-OF-MERIT FOR VARIOUS TEST FUELS

Type of Test Fuel	Number of Components	Calculated Reactivity Addition to the CDC (¢)	Figure-of-Merit	
			Calculated	Measured
PL-2 Fuel Rod	1	39 (33)[a]	3.6	4.0
PL-2 Fuel Rod	3	89	2.6	
CDC Fuel Rod	1	30 (24)[a]	2.6	2.9
CDC Fuel Rod	3	71	2.0	
Small "D" Core Plate [15]	1	17	12	
Small "D" Core Plate	3	53	12	
PBF Fuel Rod[16]	1	35	2.7	
PBF Fuel Rod	3	81	1.9	

[a] Measured values

1.7 Summary of CDC Static Nuclear Characteristics

The principal calculated [13] and experimental [17] static nuclear characteristics of the CDC are summarized in Table IV. In general, the calculated values were in good agreement with experimental results.

TABLE IV

STATIC NUCLEAR CHARACTERISTICS OF THE CDC

	Calculated	Measured
Critical mass (control rods at upper limit, 36.4 in. above bottom fuel rods)	2.2×10^3 kg of UO_2 (≈ 1350 fuel rods)	2.29×10^3 kg of UO_2 (1431 fuel rods)
Final operational loading	2.6×10^3 kg of UO_2 (≈ 1620 fuel rods)	2.66×10^3 kg of UO_2 (1660 fuel rods)
Excess reactivity with control rods at 36.4 inches upper limit	2.9$	2.9$
Shutdown margin with control rods at lower limit	9.6$	9.3$
Critical position of control rods	23 in. above bottom of the fuel rods	27.6 in. above bottom of the fuel rods
Reactivity with control rods at 27.6 inches and transient rod at upper limit	-5.6	more than 3$ subcritical
Reduced prompt neutron lifetime, ℓ/β_{eff}	3.8 msec	4.45 msec
Reactivity worth of 100% void in test space	+2.4$	2.0$
Uniform void coefficient of parent core	---	-0.0017¢/cc
Isothermal temperature coefficient (20°C)	---	-0.57¢/°C
Peak-to-average thermal flux ratio with control rods at critical position	4.0	3.6
Excess reactivity with rods completely removed from core	+5.5$	(not measured)

2. KINETIC PROPERTIES OF THE CDC

This section contains calculational results of the expected kinetic response of the driver core and of various types of experiment test fuels.

2.1 Kinetic Responses of the Driver Core

The Capsule Driver Core has a number of properties which are common to those of the Spert I Oxide Core [1, 2, 3] previously used in Spert I, and this similarity permits, to a certain degree, the extrapolation of the available oxide core data and analytical techniques to the CDC. These reactors have the same metal-to-water ratio and identical fuel rod construction. The CDC differs only in the enrichment (and also slightly in the UO_2 density) and in the overall size of the core which contains about 2.8 times as many fuel rods as did the Spert I Oxide Core (ie, 1660 versus 600). A calculational model derived by Spano [10] for the reactivity feedback in a low-enrichment oxide core has been applied via the space-independent kinetics equation to the self-limiting power excursion tests conducted in the previous Spert I Oxide Core. Calculated values of peak powers, burst energies, and burst shapes have shown good agreement with measured values when only Doppler effects are considered in reactivity feedback. That is, other known feedback mechanisms, such as moderator heating and boiling, have not been considered due to their apparent lesser role in the shutdown process. Therefore, these calculations may be considered somewhat as overestimates of energy releases and power levels to be reached. The Doppler reactivity feedback model together with the computer code IREKIN [18] was used in the determination of expected burst shapes, peak powers, and burst energies for the CDC.

2.11 The CDC Inhour Relation. The inhour curve for the CDC, shown in Figure 14, was developed using computed values of ℓ/β_{eff}. Since the present core design limits the maximum control rod withdrawal of the CDC to 36.4 inches above the bottom of the fuel rods, the maximum reactivity insertion will be about 2.9$ which, as can be observed from the inhour curve, will yield the minimum reactor period of about 2 msec.

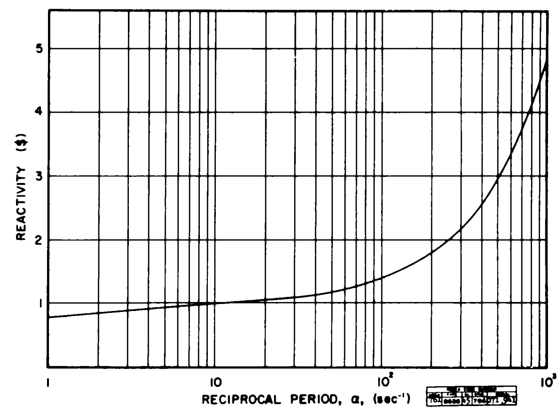

Fig. 14 Computed reactivity insertion versus reciprocal period for the CDC.

2.12 Power. Figure 15 shows the relation of peak power versus reciprocal period for the CDC obtained from the Doppler reactivity feedback model. Peak power from about 1.6 GW at a 10-msec-period transient to about 43 GW at a 2.2-msec-period transient are predicted. The peak power density in the CDC for periods of 10 msec and 2.2 msec is expected to be about 617 W/g of UO_2 and 16,600 W/g of UO_2, respectively. These values may be compared to 770 W/g of UO_2 for 9.9 msec and 18,400 W/g of UO_2 for 2.2-msec-period transients in the Spert I Oxide Core. Calculated CDC burst shapes for transients having initial periods of 10 msec, 7.5 msec, 3.0 msec, and 2.0 msec are shown in Figures 16 and 17.

2.13 Energy Release. Figure 18 shows computed curves of total energy release and energy release at the time of peak power for the CDC. The energy release at the time of peak power varies from about 31 MW-sec for a 10 msec period transient to a value of about 247 MW-sec at a 2 msec period transient. As can be observed in Figures 16 and 17, the total energy release is dependent upon the time at which the reactor is scrammed, since post-burst power levels are relatively high. The energy release was computed to 0.50 sec after peak power to obtain the total energy release curve shown in Figure 18.

2.14 Core Fuel Response Figure 19 is a plot of the computed average enthalpy at the time of peak power for the CDC along with Spert I Oxide Core data. The Spert I data points, shown at reciprocal periods of 455 and 645 sec^{-1}, are the experimental results of planned potentially destructive tests.

During the Spert I Oxide Core test, which had a period of 2.2-msec ($\alpha = 455\ sec^{-1}$), two fuel rods ruptured about 0.5 msec after peak power. The only other significant damage incurred during this test was the bowing and/or discoloration of about 150 fuel rods. Damage which occurred during another Spert I Oxide Core test, with a period of 1.55-msec ($\alpha = 645\ sec^{-1}$), included

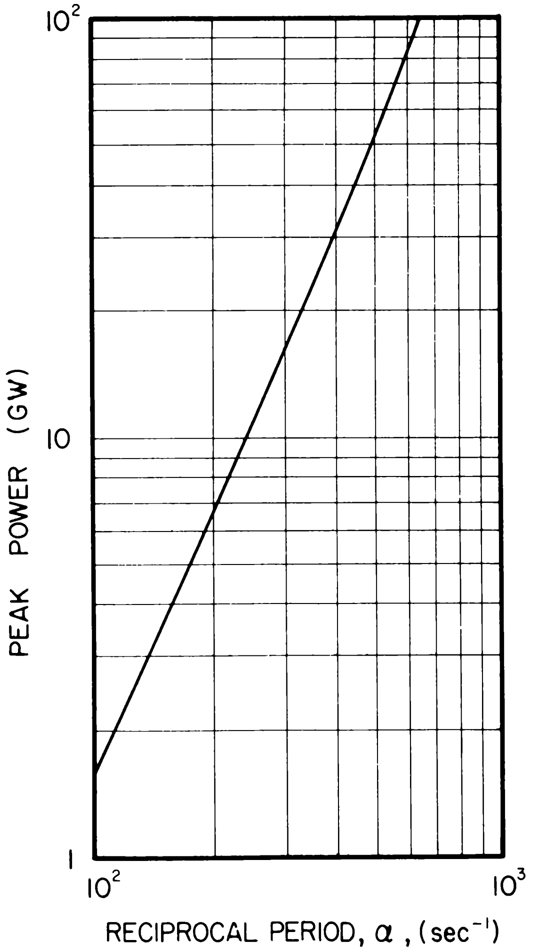

Fig. 15 Computed CDC peak power versus reciprocal period.

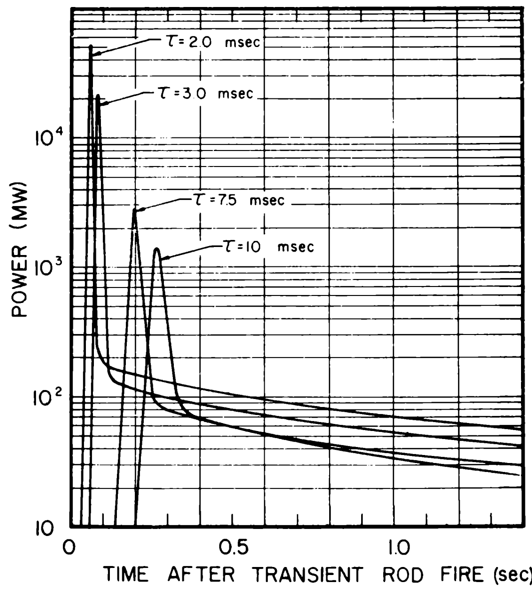

Fig. 16 Computed CDC power plots for various period, τ, transients.

the rupture of two fuel rods about 0.6 msec before peak power, along with bowing and/or discoloration of about 175 fuel rods. The rupture of fuel rods is suspected to have been caused by "waterlogging" that resulted from defects [3]. Rupture of fuel rods causes reactivity compensation, in addition to Doppler broadening, because of sudden moderator heating, boiling, and expulsion. This increase in reactivity compensation resulted in a decrease in the expected enthalpy at peak power for the 1.55-msec-period test as evidenced by the data of Figure 19. Whereas the average enthalpy at peak power was predictable for the 2.2-msec-period test in which fuel rod rupture occurred after peak power, it was much less than expected for the 1.55-msec-period in which fuel rod rupture occurred before peak power.

It has been established [3] for these Spert I tests that the reactivity effect arising from the expulsion and redistribution of fuel after the fuel rods ruptured was of negligible magnitude and probably did not significantly affect any of the observed kinetic properties.

Figure 20 shows computed values, after completion of power excursions, of the average and total enthalpy for the CDC as a function of reciprocal period along with Spert I Oxide Core data. Energy release was computed to 0.05 sec after peak power for comparison with Spert I Oxide Core data. Maximum enthalpy is defined as the volume-weighted, peak-to-average thermal flux ratio multiplied by the average enthalpy and represents the enthalpy at the core hot spot. Enthalpies obtained for the two short-period Spert I tests were less than expected due to fuel rod ruptures which caused premature shutdown of the reactor. The expected kinetic response of the CDC in terms of power and enthalpy is nearly the same as the measured response of the Spert I Oxide Core for power excursions having periods of 3 msec or longer. Therefore, since the

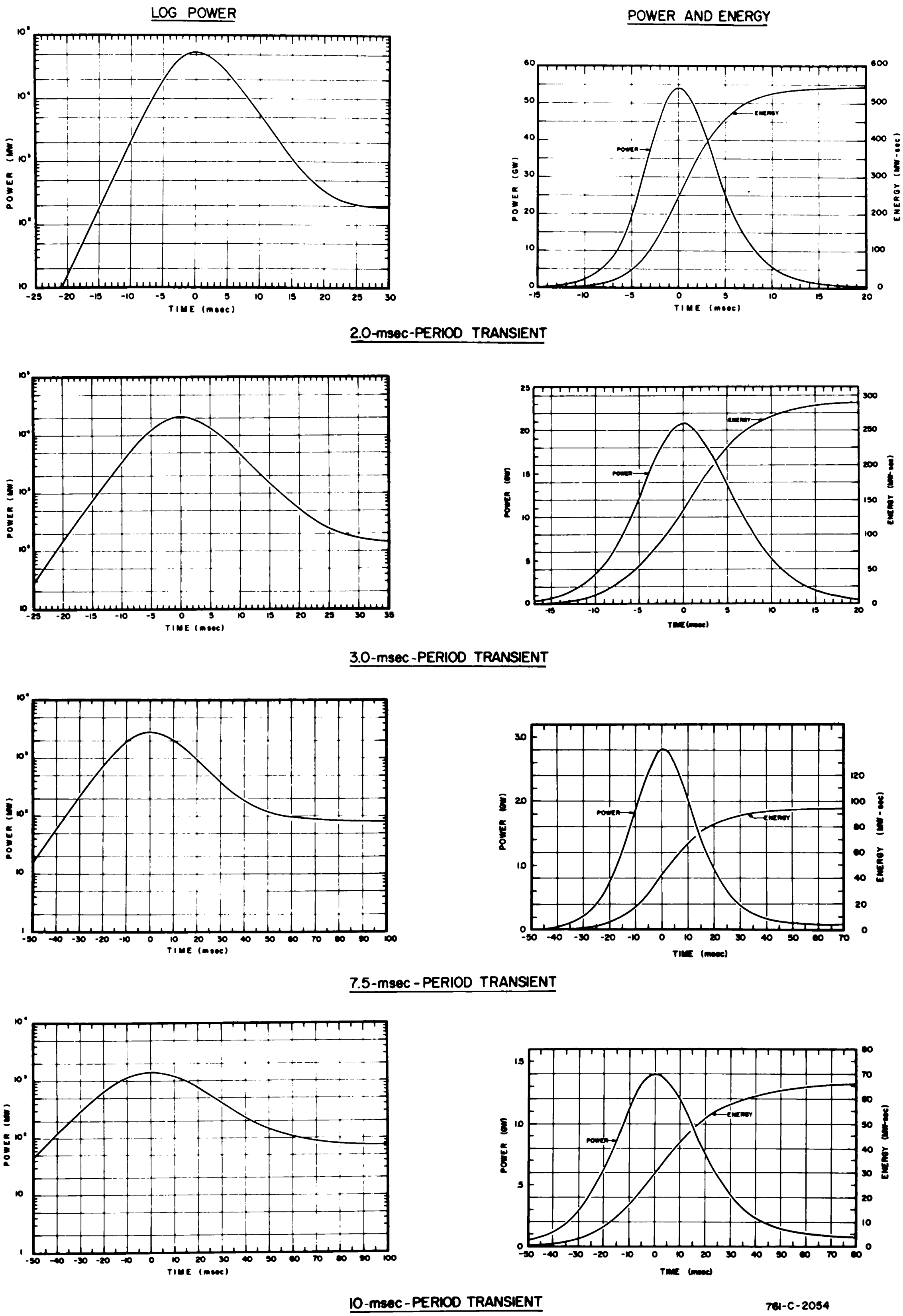

Fig. 17 Computed CDC power and energy as a function of time.

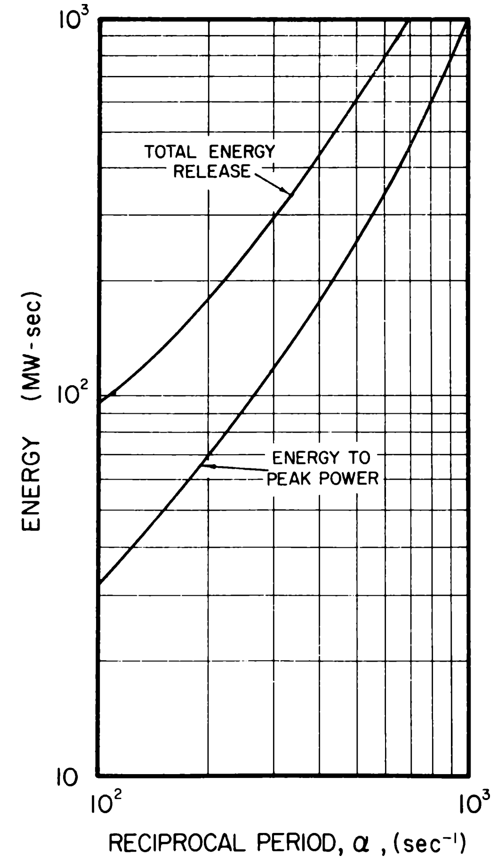

Fig. 18 Computed CDC energy release versus reciprocal period.

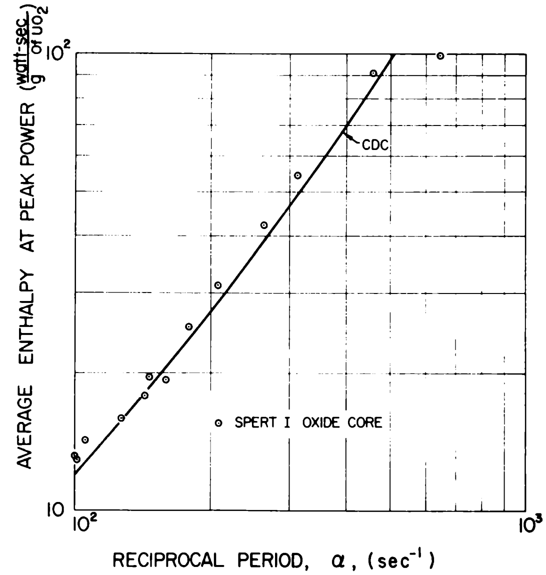

Fig. 19 Computed CDC and experimental Spert I Oxide Core average fuel enthalpy at peak power.

CDC fuel is nearly identical in all physical and thermal properties to that of the Spert I Oxide Core, thermal responses such as temperature distribution, rod bowing, and discoloration are expected to be similar to those observed in the previous core.

2.2 Test Fuel Responses

2.21 Test Fuel Types. Calculations have been performed to evaluate the performance capabilities of the CDC with respect to several types of test fuels which may be tested as part of the CDC program. The fuels considered here include the following:

(1) CDC fuel rods, 3 percent enriched, compressed UO_2 powder in 0.028-inch stainless steel cladding.

(2) Proposed fuel rods for the Power Burst Facility (PBF) [16] which are 0.75 inch OD,

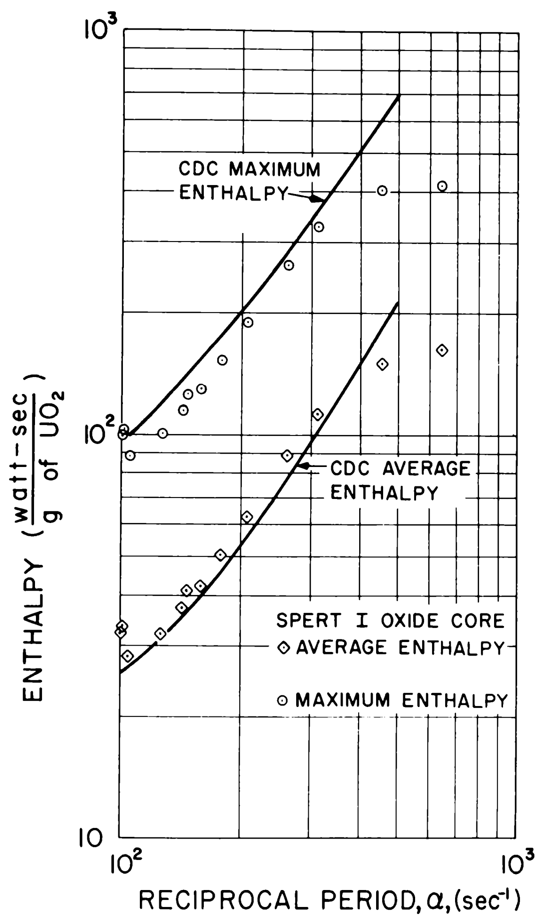

Fig. 20 Computed CDC and experimental Spert I Oxide Core enthalpy after power excursions.

clad with 0.028-inch-thick type-304 stainless steel and contain 0.56-inch-OD fuel pellets, composed of a mixture of 60 percent, by volume, UO_2 (7 percent-enriched) and 40 percent Ca stabilized ZrO_2 (the PBF fuel pellets are surrounded by a ZrO_2 insulator, 0.69 inch OD)

(3) PL-2 fuel rods [14] which are composed of 4.8 percent-enriched UO_2 fuel pellets, 0.420 inch OD, contained in 0.466-inch OD, type-304 stainless steel tubes, 0.020 inch thick

(4) Spert "D" [15] core-type fuel plates that are 93 percent-enriched UAl meat, 0.020 inch thick by 0.75 inch wide, clad with aluminum 0.020 inch thick by 1 inch wide, and having 0.060-inch-thick moderator channels.

Other fuel types will, of course, be encountered during the CDC testing program, and these will be analyzed as they arise prior to being approved for testing.

2.22 Temperature Response of Test Fuels. Based upon the figures-of-merit presented in Section III-1.6, Table III, and the highest expected enthalpy possible for the CDC, expected maximum enthalpies of the selected test fuels were calculated as a function of the CDC excursion period. These are shown in Figure 19.

Expected adiabatic enthalpies and temperatures of the various test-fuel types were computed using the data of Figure 19 and computed specific-heat capacities. Figures 21 and 22 show these quantities for the various test fuels as functions of reciprocal period for a 2.5-msec-period transient, the expected temperature responses of various test fuels are listed in Table V.

Although the physical consequences of short-period, high temperature excursions in fuel samples are not well known, it is reasonable to assume that transient pressure pulses may arise around the fuel sample as a consequence of several possible processes including: (a) fuel rod bursting caused by high internal pressures; (b) rapid dispersal of finely divided, hot, UO_2-or UAl-fuel into the water, leading to steam explosions [a], and, in the case of UAl-fuel, a possible metal-water reaction; and (c) steam explosions or metal-water reactions caused by the molten clad.

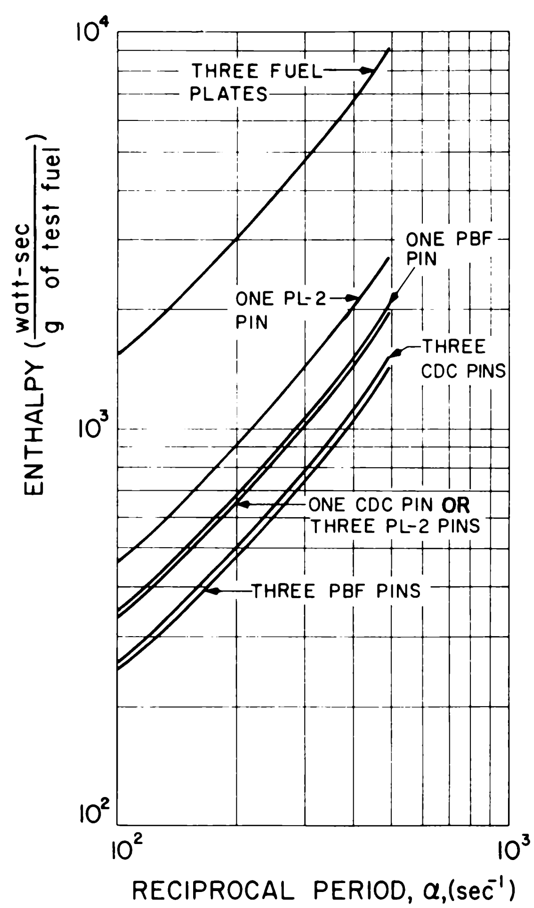

Fig. 21 Computed average energy in test fuels.

[a] This is a speculative result since there is no evidence that such explosions will occur. In fact there is evidence [3] that this process may give rise only to small nondamaging pressures.

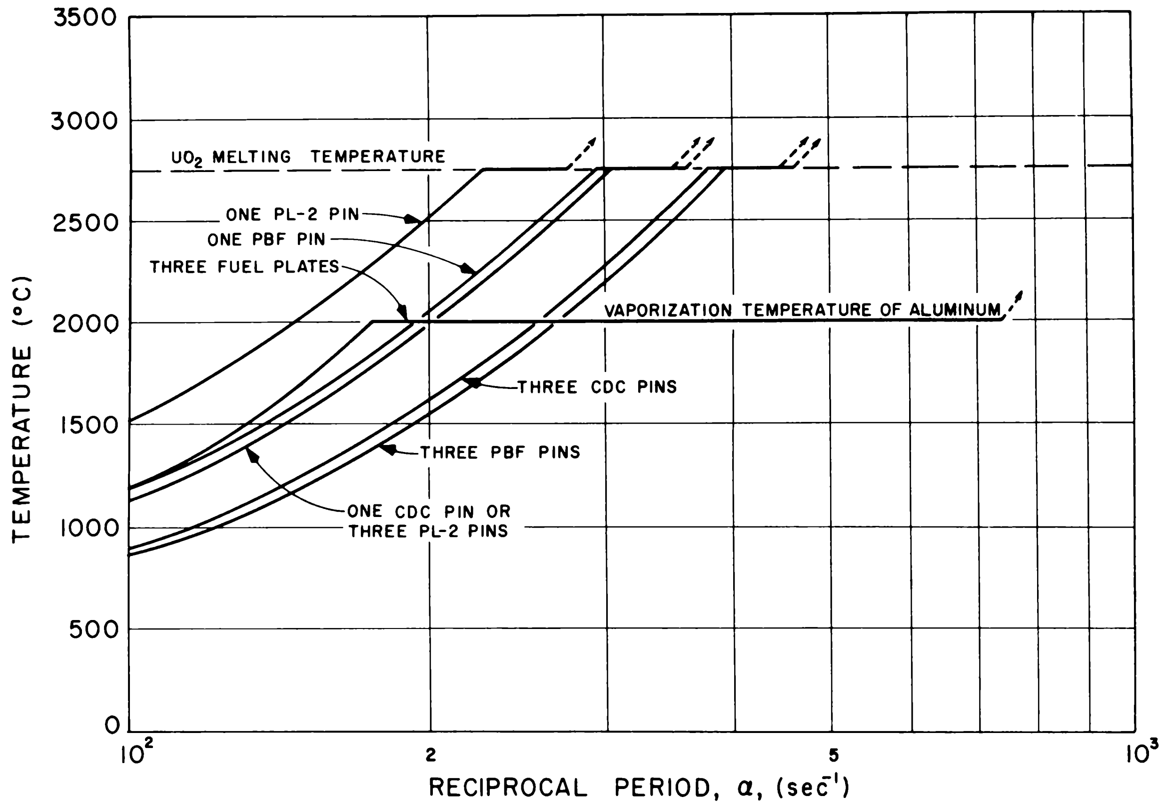

Fig. 22 Computed maximum adiabatic temperature of test fuels.

TABLE V

EXPECTED TEST FUEL RESPONSE FOR A 2.5 MSEC PERIOD

Type of Test Fuel	Number of Fuel Elements	Maximum Enthalpy in Test Fuel (cal/g)	Expected Adiabatic Temperature Response
PBF Fuel Rod	1	360	Complete melting
	3	250	10% melting
PL-2 Fuel Rod	1	480	Complete melting
	3	345	Complete melting
CDC Fuel Rod	1	345	Complete melting
	3	260	25% melting
Small Spert "D" Core Type Plate	3	1600	50% vaporization

IV. EXPERIMENTAL PROGRAM

The Capsule Driver Core will constitute one of the most useful tools available for the development of information pertinent to reactor safety. Eventually, the CDC program will include studies not only of oxide fuels and metallic fuel plates, but also prototype or developmental fuels, so that safety information can precede the full-scale test-reactor stage of fuels development. The CDC is also expected to play an important part in the reduction of hazards in presently existing fuels by possibly indicating beneficial changes in materials and/or design. The CDC also will become involved in materials testing, fuels development, instrument development, and other areas wherein needs exist for a high-nvt burst facility.

It is due to this diversity of the CDC capsule program that long range predictions of the detailed nature of the capsules and of other capsule programs are not possible. Safety analysis of each of these programs, obviously, is outside the scope of this report. Nevertheless, one general feature of the safety analysis to be conducted for each capsule program can be stated here, (viz, each individual capsule program in the CDC will be analyzed in all important safety aspects and reviewed by both Phillips management and the Spert-Step Safeguards Committee before being approved for testing in the CDC). The analysis which each experiment must undergo, and the review and approvals necessary, are set forth in the Standard Practices Manual which is discussed in Appendix B. Because of these reviews and the other established practices at Spert, the CDC program should never involve hazardous procedures or constitute hazards in excess of those to be reviewed in Section V of this report.

1. NUCLEAR START-UP AND STATICS EXPERIMENTS

A series of tests has already been performed on the Capsule Driver Core. These tests (designated the Static Test Series) included the loading of the core to critical, loading to operational size, measurement of control-rod and transient-rod worths, determination of the shutdown margin, and determination of the total excess reactivity, various flux distributions, void and temperature coefficients, and power calibration factors. The tests provided information which is useful for both instrumenting the core and initiating the next testing phase, which constitutes an investigation of the kinetic properties of the core. During the execution of the Static Test Series, the core was found to respond largely as predicted by previous computer calculations, and no unique procedures or problems were involved.

2. FIDUCIAL TRANSIENT TESTING

The CDC will be subjected next to a series of power excursions designated, the Fiducial Transient Tests. The general nature of this test series is that of progressively decreasing the period of each test until reaching a region of periods wherein possible core damage can occur or wherein the response of the core is not considered to be predictable. The fiducial tests, typically, include measurements of the nuclear power, energy release, dynamic flux profiles, moderator dynamic pressure, temperature and strain. These measurements are made for the purpose of gaining sufficient information about the core to allow:

(1) Definition of an operating region; that is, a region of periods in which safely conducted kinetics can be performed.

(2) An understanding of the kinetic behavior of the core which will be compared with the behavior observed in other Spert cores.

(3) Extrapolation to shorter period tests.

(4) Proof-testing of the operational instrumentation and procedures.

(5) Verification of the theoretical calculations predicting the kinetic responses of the core.

The fiducial series will begin with a transient which has a period of the order of 1 sec. Such long period transients can be observed in "real" time at the Control Center and can easily be controlled in the event of unexpected behavior. A subsequent transient, also with a long period, is then run so that, with the two sets of data, extrapolations complemented with theory become possible of peak power, total energy release, fuel temperature, and other factors which aid in the safe and successive increase of reactivity to achieve shorter periods.

During the fiducial series, careful monitoring of fuel and clad temperatures will be conducted both by direct (thermocouple) measurements and by calculations of temperature based upon measured energy releases. Physical responses of the fuel rods to these temperature excursions generally take the form of small stresses in the clad, which may produce elastic swelling or bowing of the fuel rods. At higher temperatures, bowing may become large and inelastic, and "scorching" or discoloration of the clad may occur when central fuel temperatures range between 1200 to 1800°C.

Fuel rod responses described here have not been directly observed with the CDC fuel (which is 3 percent enriched) but are inferred from the observed responses of similar oxide fuels used in the Spert I Oxide Core programs (which were enriched to 4 percent). The two fuels are identical in all known physical and thermal characteristics with the exception of enrichment and are therefore expected to perform alike. The fiducial transient program to be conducted with the CDC will, nevertheless, be conducted cautiously as though little or nothing is known about the fuel response to thermal excursions. Post-test visual inspections of the fuel will be performed periodically, and measurements will be taken to determine any physical deformations.

The most valuable information to be obtained during the fiducial transients, however, will be that obtained from a capsule. A regular CDC fuel rod sample will be encapsulated and placed in the flux peak of the experiment tube so that full advantage can be taken of the figure-of-merit to anticipate fuel rod responses. A CDC fuel sample in the experiment tube will receive nearly three times the energy release of the hottest fuel rod in the core. By post-test inspection of a test sample after CDC transients, it will be possible to predict more accurately the responses to be encountered by the core, per se, as the period is gradually reduced. It is possible, by this technique, to reduce the period of the CDC core until the maximum core temperatures are in the range of 500 to 600°C and at the same time, study fuel responses at temperatures which approach and exceed those reached during the shorter periods of the previous test program using this type fuel. Not only can the problem of high temperature response be anticipated or studied, but also other problems of repeated thermal excursions at high temperature can be evaluated. In the Spert I Oxide Program, many tests on this type fuel were conducted at temperatures approximately 1000°C and below, and a degree of confidence has been gained from these tests which indicates that repeated tests can be conducted on the CDC fuel without undue concern about fuel responses. However, in the temperature range from 1500°C on up to about 2000°C, little information is yet available, and this region can be most conveniently studied by this "leading rod" capsule technique for both single and repeated tests to determine whether the CDC core itself can safely operate in this region.

During the "leading rod" experiments emphasis will be placed also upon evaluating the consequences of fuel rod waterlogging and the failure mode of such waterlogged fuel rods under transient conditions.

With the aid of experience gained during the Spert I Oxide Core tests, theoretical predictions of transient behavior, and with the results of the "leading rod" capsule studies, successively shorter period transients will be executed in the CDC until the period has been reduced to the lowest value possible without damaging the reactor fuel or producing other undue hazards. For the CDC, this lower limit is presently expected to be in the range of 2 to 3 msec.

3. CAPSULE TESTING

After the nuclear characteristics and kinetic behavior of the CDC have been established by completion of the Statics and Fiducial Transient Test Series, the core will then begin serving in its function as a burst irradiation facility for encapsulated fuels and other materials. Periods (and therefore energy releases) used for capsule irradiations will lie within the range which has been shown to be nondamaging to the core as established in previous fiducial test series, with the exception, however, that modifications of the established range of safe periods may be effected as a result of new findings arising from the capsule program itself.

As mentioned earlier, specific criteria for individual tests are difficult to establish at this time due to the varied nature of the objectives. The Spert IV Operating Limits, however, set forth general criteria that no in-pile experiment will be permitted that indicates any reasonable expectation of major damage to the facility. The Operating Limits also set forth the requirements for safety analysis, review, and approvals of each specific test series proposal. Also, although the steel experiment tube is expected to protect the core against all reasonable pressure disturbances, capsules will be designed to provide containment of all experiments involving the possibility of either an explosion or a fisside release, and the experiment tube will be given credit only as a secondary containment. These general limits and the analysis and review of each specific test series proposal are believed to provide assurance that adequate safeguards will be exercised before any capsule experiment is permitted in the CDC.

V. RADIOLOGICAL HAZARDS ANALYSIS

In this section, the types of accidents which could conceivably occur during the course of the CDC program are discussed. For each case, the discussion includes first a review of the circumstances required for initiation of such an accident; next, a discussion of the measures which are believed to essentially negate the possibility of the accident; and finally, the course and consequences of the accident postulated. In general, the course of events postulated tends to maximize the consequences in terms of the radiological hazards.

Other accident situations, no doubt, exist; however, none can be conceived which, from considerations of radiological consequences or probability of occurrence, are any greater than those discussed. The intent here is to explore the types of accidents which can be postulated and to select from these the worst accident, as a basis, for a quantitative radiological hazards analysis.

It is believed that the accidents to be discussed categorize the worst accidents which can be postulated for the CDC, and that all other accidents either produce less fisside burden, less release potential, or converge to the same hazard as the worst accident discussed here.

1. OUTLINE OF POSTULATED ACCIDENTS

The accidents to be discussed are diagrammed in Figure 23 to illustrate the course of each accident and the cause-and-effect relationships. A common feature of all accidents is that fisside release from the CDC core necessarily involves the rupture of at least one fuel rod. Additional fuel rods may also rupture for the same reason that the first rod ruptured or as a consequence of the first rod rupture. In other words, if a fisside release from the core is to occur, the causitive agent(s) of the accident must be sufficient to produce at least one fuel rod rupture. The probability of more than one rupture is, of course, dependent upon the nature of the causitive agent. In two cases, the accidents lead to a condition wherein a large fraction of the core may be propitious toward fuel rod failure from both internal and external causes, and a cascading rod-rupture process may be postulated. In most cases, the causitive factors would likely produce, at most, only a few rod-failures; however a cascading rupture process cannot always be eliminated.

2. THE DEFECTIVE ROD ACCIDENT (I)

2.1 Cause

For this accident, it is assumed that either as a consequence of repeated usage of the core or as a consequence of an original defect, one or more rods attain a condition whereby they are prone to rupture during a power excursion. An example of such a condition (and one which has been experienced before) is a cladding defect which allows water to enter and saturate the fuel and constitute a source of internal pressure. The "accident" finally occurs during a normal CDC power excursion, when the fuel rod is caused to rupture.

2.2 Preventive Measures

All of the fuel rods in the CDC were visually inspected for physical defects, and those which were found containing such defects have been rejected. During use of the CDC, fuel rods will periodically be removed and inspected for evidence of possible damage. Also, by the "leading rod" capsule experiments described in Section IV, systematic types of fuel rod damage will be discovered and accounted for in establishing the limit of period or of energy release to which the CDC will be subjected. It is nevertheless possible for original defects or nonsystematic defects arising from operation to escape detection and cause incidental fuel rod ruptures.

2.3 Consequences

It is believed that the causitive factors involved in this accident would reasonably give rise only to one, or at most, a few rod ruptures since, for the most part, the other rods in the core are intact and can be expected to be immune to all but the most violent external disturbances. Thus, after the first rod rupture (Figure 23, page 33), this accident would likely follow path, "A", leading to a final conclusion of no major damage and negligible fisside release. This conclusion is also supported by experience gained with burst fuel rods in the previous destructive test program in Spert I[3]. Nevertheless, neither theoretical nor experimental data presently available justify complete exclusion of path "B" leading to more severe consequences.

3. CONTROL ROD WITHDRAWAL ACCIDENT (II)

3.1 Cause

It is assumed that while preparing for a power excursion the operators of the reactor inadvertently raise the control rods to their upper limit before raising the poison section of the transient rod into the core. Other possible causes include: a defective control system, malicious behavior on the part of personnel, and accidental manual lifting of the rods.

3.2 Preventive Measures

Recognizing the possibility of such causitive circumstances, standard practices and administrative controls have been established to control all nuclear operations. It is the intent of these to prevent such occurrences by the exercise of close supervision, redundant operator responsibility, pretest control system check-out, etc. In view of these regulations, the accident is considered to be extremely improbable. Also, it is unreasonable to expect that the accident would not be aborted early in its course by any one of several personnel in the control area when the alarms and neutron indicators were activated. Spert regulations require two certified operators in attendance during all nuclear operations, and it is also a standard regulation that anyone may, and, in fact, is directed to scram the reactor in the event he feels that nuclear operation is leading to an unsafe situation. A rod-reversal or manual scram within the "reaction time" of either of the two official operators can be expected when the alarms or neutron sensors are activated. It would require deliberate inaction on the part of all concerned for extended time, since about two minutes are required to remove the rods from the critical position to the upper limit. It is concluded that after neutron indicators and alarms are activated, a duration of this accident in excess of the order of seconds is not credible. The probability that personnel would be in the reactor building and therefore exposed to radiation is also extremely improbable since once the rods are raised from the "~~lower limit~~ SEAT" an alarm is sounded, and a manual "scram" is expected to originate from either the reactor building, as personnel egress from the building, or from the control room. An accident originating from the manual lifting of control rods is considered to be extremely improbable in view of the existing administrative control over personnel and, also, to the general personnel awareness of this problem. Also, it would require the concerted effort of two or three people to effect this action since three control rods must be removed to cause criticality.

~~3.3 Consequences~~

3.3 Consequences

The accident constitutes a type known as "ramp initiated" since the reactivity would be added as a function of time as the rods are extracted. If the control rods were pulled continuously from the critical position to the upper limit at the maximum withdrawal rate of 12 inches/minute, the minimum period would be about 85 msec with a peak power of about 55 MW occurring 13 seconds after the start of rod withdrawal. Even if no operator action were taken until the control rods reached upper limit (about 45 seconds), the maximum fuel temperature would be less than 1000°C, and no fuel damage or fission product release would be expected.

4. THREE-DOLLAR CONTROL-ROD-POSITIONING ACCIDENT (III)

4.1 Cause

It is assumed that an error is made in the actual positioning of the control rods while preparing for a power excursion. Such an error could arise either from a faulty calculation, an inattentive reactor operator, or for various mechanical reasons such as a faulty rod-position indicator. The worst of such errors would place the rods at the upper limit amounting to about 3$ excess reactivity. It is assumed, contrary to expectation, that a 3$ power excursion would cause fuel temperatures (or other responses of the fuel) which are destructive.

4.2 Preventive Measures

Regulations require that two certified operators be in attendance during all nuclear operations so that calculations and rod-positioning actions are always double checked and supervised. The upper limit of control rod motion has been deliberately set at a reactivity level (3$) just adequate for the CDC program, and an excursion involving this amount of reactivity is expected to produce fuel temperatures several hundred degrees below the melting point of UO_2. Such temperatures, it is believed, would not cause immediate fuel rod failure, unless the fuel rods were in some way defective (discussed in Section V-3).

Mechanical failures which could lead to an undetected over-insertion of reactivity cannot be totally denied; but, in view of the routine inspection procedures, visual readouts and television monitoring, these are believed to be extremely improbable. The critical position of the control rods is always determined in the course of preparing for a transient, and malfunctions would almost certainly become apparent at that time, if not before.

4.3 Consequences

Specific empirical data on the response of CDC fuel at temperatures approaching the melting point are not available. Thus, the assumption is valid that, at such temperatures, fuel rod damage, including possible rupture, may occur.

The consequences, as shown in the block diagram of Figure 23, page 33, include the probable route that the energy release would lead to fuel rod damage (ie, warping, expansion, and scorching) but, not specifically, fuel rod failure. In this event, there would be no radiological hazard produced since the fissides all remain contained.

Another route (considered less probable) suggests that, for various possible reasons, the cladding fails. These reasons might include either severe thermal stress loads on the cladding or cladding meltdown, so that fuel rod failure is a possible result. The temperatures reached in this accident, however, offer no reason to suspect that such unlikely fuel rod failures could in any way affect neighboring fuel rods. That is, the failure modes conceived here (which do not include the defective fuel rod previously covered) do not appear violent and would not produce disturbances in the core sufficient to disrupt fuel rods in other regions of the core. The fisside release is expected to be confined to the few rods which are initially ruptured due to either cladding meltdown or spill-out from a rod which fractures due to thermal stress. If the release of UO_2 is not accompanied by rather severe pressure disturbances, the large body of water surrounding the reactor would nearly immobilize all but the gaseous fissides and render this accident to a small- or negligible-hazard category.

5. FIVE-DOLLAR CONTROL-ROD-POSITIONING ACCIDENT(IV)

5.1 Cause

It is assumed that, by virtue of the particular capsule experiment being conducted in the CDC, an additional 2$ of excess reactivity is available over and above the 3$ reactivity (considered in Section V-4) which is held in the control rods. Thus, this accident is similar to the last, except that now 5$ reactivity is available. It is possible for the additional 2$ reactivity to be made available if nearly all of the water moderator in the experiment tube is removed as, for example, it might be for an experiment involving a gas-cooled reactor element meltdown test.

The accidental upper-limit positioning of the control rods is assumed to take place as explained in Section V-4.1 above.

5.2 Preventive Measures

On the premise that the additional reactivity could be added to the system inadvertantly, it is reasonable to expect an immediate cognizance of this fact during a pretest criticality check. Such a check would reveal a displacement of the critical position; and, consequently, the operators would be alerted to the situation prior to setting the control rods for a test.

If, on the other hand, the reactivity addition is known and deliberate (such as with an experiment) then, in addition to the fact that the change in reactivity status of the core would be noted during the pretest critical, there will also exist administrative limits on the control rod positions to be set for the experiments. That is, new control rod limitations will always be made which are dictated by the reactivity worth of the experiment. Preventive measures will always primarily consist of the standard practices and administrative controls developed for this purpose and discussed above in Section V-4.2. Finally, in the event that the experiment hole is to be intentionally voided for a series of capsule experiments, the electrical and mechanical control rod stops will be moved to limit the available excess reactivity to three dollars.

5.3 Consequences

An excursion involving 5$ excess reactivity is calculated to release approximately 2500 MW-sec of energy leading to core hot-spot energy densities sufficient to cause vaporization of UO_2 and pressure buildup inside of the fuel rod cladding. The average fuel temperature is predicted to lie below but close to the melting point of UO_2. It is reasonable to expect fuel rod bursting due to vapor pressure in a small fraction of the core. Also, due to the bursting pressures and the probable high velocity of hot UO_2 particles emanating from burst fuel rods, the initial fuel rod failures can conceivable give rise to a cascading type of fuel rod failure. That is, there may be a disturbance of sufficient magnitude from a bursting vaporized fuel rod to cause adjacent rods to fracture also. The cascading type of rupture probably would not include all of the fuel rods in the core, but it is nevertheless not unreasonable to assume, due simply to the net force on grid structures, etc, that peripheral fuel rods, not directly ruptured, would be bent, twisted, and otherwise caused to crack and spill.

Thus, from this accident, there is a distinct possibility of a major fisside release combined with a disturbance of the water surrounding the core which could encourage the fissides to become airborne. (This consequence is path "B" as illustrated in the block diagram.) Since the radiological hazards of this accident appear to be as large or larger than any other accident, the consequences are the subject of a quantitative analyses appearing in later sections.

6. EXPLODING CAPSULE ACCIDENT (V)

6.1 Cause

In contrast to the 5$ control-rod-positioning accident wherein the experiment tube was assumed to be voided before the accident, it is assumed, in this case, that the voiding of the experiment tube is a consequence of an explosion of materials in the capsule. Implicit in this assumption are the requirements on peak pressure, energy, total expansion, and time-of-occurrence of the explosion which are required to cause complete voiding of the experiment tube during the power excursion so that the 2$ worth of the void is added to the initial reactivity of the excursion, which is taken at the maximum of 3$. The accident, then, assumes the magnitude of a 5$ step excursion.

6.2 Preventive Measures

The general criteria established for capsule design include maximizing the containment capabilities of each capsule to obviate the possibility of capsule rupture. Design burst pressures of the capsules will usually be of the order of 8000 psi and only in the most trivial experiments will they be less than 3000 psi.

Capsule experiments with potential explosive reactions will be preceded by tests with small samples, such that the explosive energy available will be minimized and thus the potential for capsule rupture mitigated until the particular explosive phenomenon has first been observed.

Whenever explosions are predictable in capsule experiments, and such explosions appear potentially to contain a significant amount of energy, then the capsules will be provided with a free-volume or "expansion" space within the capsule to allow immediate relief of internally developed pressures. Although it is credible to assume a pressure pulse with a rise-time sufficiently short to cause fracturing of the capsule, the path of least resistance to subsequent mass flow would be inside of the capsule into the expansion space where pressure relief would occur. Also, a "relief tube" may be attached to one end of the capsule which would allow the capsule to vent freely into a hold-up tank outside of the reactor. Thus, if such an explosion were to occur which ruptures the capsule, the ability of such an explosion to void the experiment tube is greatly reduced since high internal gas pressures could not be maintained longer than a few milliseconds before venting and pressure relief would be realized. Total voiding of the experiment tube would require an unrealistically high-volume source of gas to effect a significant voiding of the experiment tube.

6.3 Consequences

The Exploding Capsule Accident (V in the block diagram of Figure 23, page 33), if assumed to insert the full 2$ worth of a completely voided experiment tube, could conceivable lead to an energy release in the core equivalent to a 5$ step excursion. As discussed in regard to the previous accident (see Section V-5.3), a 5$ excursion would release approximately 2500 MW-sec of nuclear energy with the result that a large fraction of the core (perhaps a third) would be raised into or above the meltingpoint of UO_2. A smaller fraction would undergo various degrees of vaporization, and it is reasonable to expect that multiple fuel rod failures would occur. Thus, route "B", as shown in the block diagram,appears to be possible with its attendant high fraction of fisside release.

7. REVIEW OF POSTULATED ACCIDENTS

In the foregoing, a number of accidents have been postulated each arising from causitive factors which, although considered extremely unlikely, nonetheless are possible by some exercise of the imagination. It was the intent in this review to include the categories of accidents which create the worst radiological hazards considered to be credible with the CDC system. Possibly there are many other "accidents" or unexpected occurrences which can also happen to the CDC which are not discussed here, but it is believed that there are none which are credible and which can give rise to radiological hazards greater than those already discussed.

Accidents IV and V in Figure 23 each involve 5$ reactivity (the maximum reactivity available) and produce not only the highest fisside burden with the CDC, but combine this with the highest probability for atmospheric release of this burden by virtue of the temperatures reached, the magnitude of the core included in the accident, and the probability of explosive dispersion of the fissides. In these accidents, it is assumed that a fuel rod whose fuel becomes vaporized will indeed rupture. It is then assumed, with less confidence, that other fuel rods at lower temperatures (and lower internal pressure) also rupture as a consequence of the first ruptures. It is finally assumed that the violence possible with a large scale sequence of rod ruptures is sufficient to cause even the lowest temperature fuel rods to become broken even though much of this fuel will be well below the melting point of the oxide, and the cladding would not be necessarily stressed from internal causes.

These accidents give rise to hazards greater than any other credible accident situations. Other accidents of the "reactivity" type and the fuel-rod "explosion" type either produce less fisside burden or less release potential or else, finally, converge on the same accident "route" followed in the "5$" accidents.

8. CALCULATION OF ACCIDENTAL EXPOSURE DOSAGES

The "5$" accidents have been selected as the "Maximum Referent Accident" for a quantitative radiological hazards analysis and the following section sets forth the results of calculations of dosages to human beings who may inadvertently be exposed to the radioactive material released by the accident.

The following assumptions and procedures are used:

8.1 Fisside Inventory

It is assumed that the Maximum Referent Accident takes place at a time when the fisside inventory of the core is at a high value as established by long and frequent usage. The fisside inventory is assumed to develop from a 100-day series of five transient tests per day, each of which releases a nominal amount of energy, taken to be 300 MW-sec. This rate of testing, prolonged over a 100-day period is a reasonable estimate of the maximum actual duty to be expected with the CDC and leads to an asymptotic upper limit of the important fisside inventories to be found in the core. (Radioiodines are essentially at an asymptotic concentration although certain long-lived isotope concentrations are still increasing slightly.)

8.2 The Release Fraction

Experience with several burst fuel rods in the Spert I Oxide Core program indicated only small fractional release of fissides (no solids, no iodines, and less than 15 percent of the noble gases); however, the use of these data here would not be considered conservative for the purposes of this document. Instead, the guideline of 10-CFR-100 (vis, a release of 1 percent of the solids, 50 percent of the halogens, and 100 percent of the noble gases) will be used here as it was used in TID-14844.

8.3 Building Containment

Although the Spert IV building, with its approximate inside free volume of 1.7 x 10^5 cubic feet, would likely contain all or at least a large fraction of the airborne particles, no credit is taken for this containment since it was not constructed for containment purposes and since the amount of fisside hold-up to be used can not be adequately defined. Therefore, the airborne contaminants are assumed to be released to the atmosphere in an instantaneous, ground level cloud.

8.4 Usage of a Computer Program

The inventory of each important isotope in the core at the moment of the release is the result of both the operating history of the reactor (described previously) and the energy release of the final transient. These inventories are computed by a machine code known as CURIE [19]. In addition, the code provides the total curies of activity available and computes the time dependency of this total for long times after the accident.

On the basis of the total inventory of the core, together with the fraction which is released to the atmosphere, the following sections present calculations of personnel dosages using calculational methods presented in Appendix C.

8.5 Cloud Dose

External radiation dosages received by persons at the Control Center and at the nearest site boundary (about 6.4 miles distant) are calculated assuming in both cases that the cloud passes directly over the person and that the person remains still until the cloud has completely passed. Wind velocities are taken to maximize the dosages and are 7 meters/sec and 2 meters/sec, respectively, for "lapse" and "inversion" conditions. Deposition velocity is taken as "zero".

8.6 Inhalation (ingestion) Dose

During cloud passage at either the Control Center or the site boundary, the exposed person breathes in radioisotopes and, therefore, suffers ingestion exposures to various body organs. Specific isotopes are considered and dosages calculated for weather conditions described above. In addition, the thyroid dose, so calculated, is adjusted to account for possible ingestion of contaminated milk. This adjustment consists of multiplying the I-131 inhalation dose by 120 and the I-133 dose by 10. The corresponding child dose is a factor of 10 greater [20]. Again, wind velocities and the deposition velocity are as indicated above. Thus, both the inhalation and the ingestion doses calculated presume that no action is taken either to evacuate personnel from the path of the cloud or to dispose of any milk which is produced and possibly contaminated.

8.7 Deposition Dose

During cloud passage, solid particles are assumed to precipitate on the ground so that a dose is received by any person remaining on the contaminated ground. At the Control Center, the time of such exposure can be controlled and is taken to be no longer than 15 minutes; although, from actual evacuation tests, the evacuation time has been shown to be less than 5 minutes. Beyond the site boundary, it is assumed, again, that no preventive action is taken, so that the exposure is given both for the first 24 hours and for a "lifetime". Wind velocities are assumed to be 7 meters/sec and 2 meters/sec, respectively, for lapse and inversion conditions. Particle deposition velocity is assumed to be 1 cm/sec.

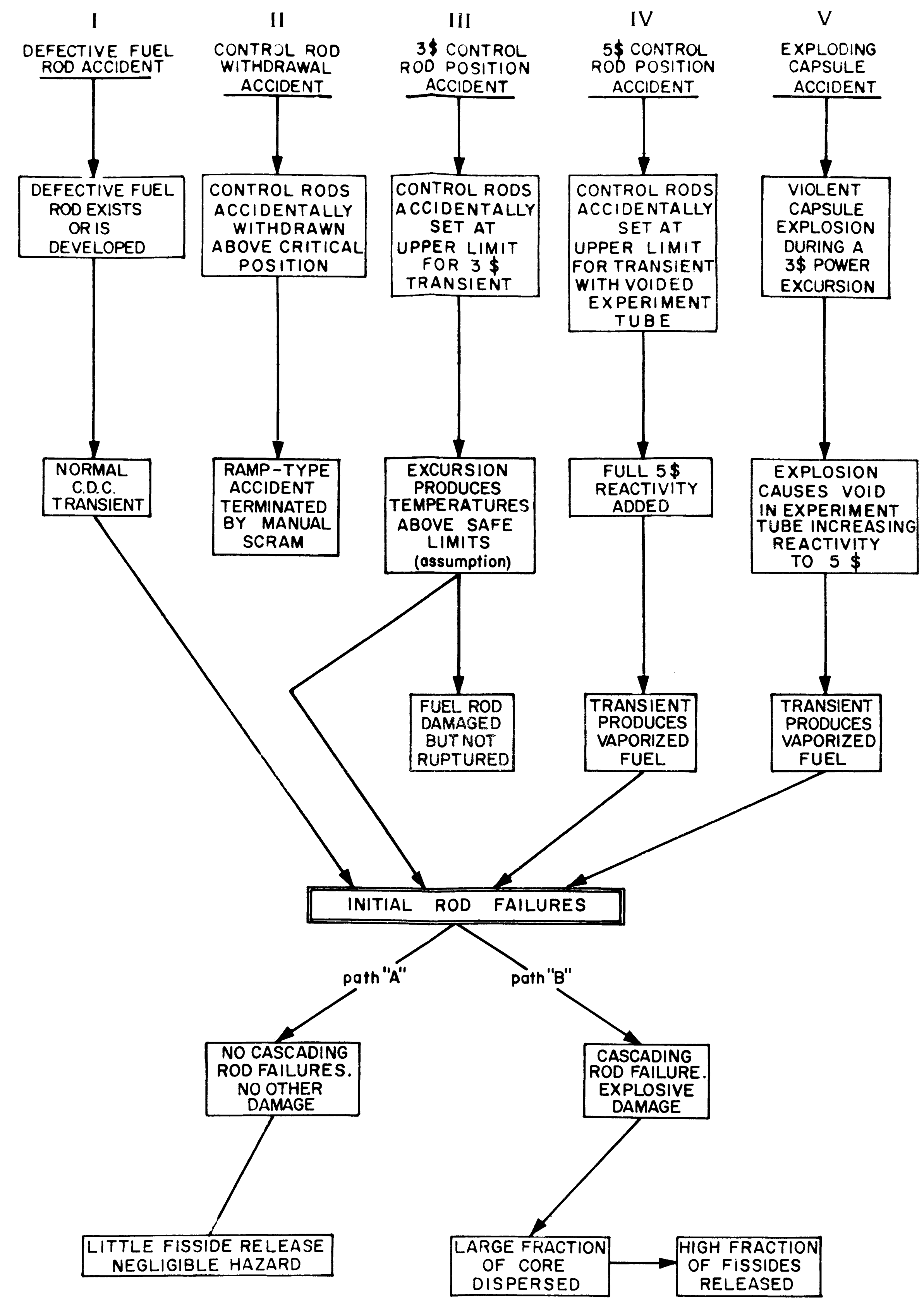

Fig. 23 Block diagram of several conceivable accidents.

8.8 Whole Body Dose

From the dosages described above (viz, cloud dose, inhalation dose, and deposition dose), the whole body doses are obtained for exposures at both the Control Center and the nearest site boundary which is about 6.4 miles distant from the Spert IV facility [a].

9. TABLES OF RADIOLOGICAL DOSES FROM THE MAXIMUM REFERENT ACCIDENT

The following tabulations show the results of these calculations. Table VI shows whole body dosages, and Table VII shows thyroid dosages. The inhalation dosages to six other body organs are shown in Table VIII.

TABLE VI

WHOLE BODY DOSES (REM)

	CDC -- Maximum Referent Accident		
Weather	Control Center (15 minutes)	Site Boundary (24 hours)	Site Boundary (lifetime)
Lapse	2.6	2.4×10^{-3}	3.6×10^{-3}
Inversion	2.1×10^{3}	8.9	13

TABLE VII

THYROID DOSES (REM)

	CDC -- Maximum Referent Accident			
		Site Boundary (6.4 miles)		
Weather	Control Center (inhalation)	Inhalation	Ingestion (adult)	Ingestion (child)
Lapse	1.30	1.34×10^{-2}	0.69	6.9
Inversion	4.56×10^{3}	94	5.2×10^{3}	5.2×10^{4}

[a] It should be noted that except for Atomic City, a village of less than 150 people located close to the nearest site boundary from Spert, the area next to the site boundary proximal to Spert is unoccupied lava bed and desert range for several miles.

TABLE VIII

TOTAL DOSES TO SELECTED BODY ORGANS (REM)
CDC -- MAXIMUM REFERENT ACCIDENT

Organ	Weather	Control Center	Site Boundary
Kidney	Lapse	1.59×10^{-3}	1.84×10^{-5}
	Inversion	5.87	0.138
Liver	Lapse	1.77×10^{-3}	1.96×10^{-5}
	Inversion	6.39	0.150
Lung	Lapse	5.19×10^{-2}	6.82×10^{-4}
	Inversion	2.19×10^{2}	5.04
Testes	Lapse	6.55×10^{-5}	7.85×10^{-7}
	Inversion	0.251	5.90×10^{-3}
Bone	Lapse	4.08×10^{-2}	6.10×10^{-4}
	Inversion	1.86×10^{2}	4.67
Muscle	Lapse	1.02×10^{-5}	1.71×10^{-7}
	Inversion	5.04×10^{-2}	1.30×10^{-3}

VI. CONCLUSIONS

It is concluded, from considerations discussed in this document, that the Capsule Driver Core operation and subassembly testing program can be conducted in a manner consistent with the general policy of the AEC of protecting government and contractor personnel and the general public against undue exposure to radiation or other health and safety hazards which may arise in the execution of nuclear safety research activities.

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APPENDIX A

CONTROL-ROD AND TRANSIENT-ROD-DRIVE SYSTEMS

APPENDIX A
CONTROL-ROD AND TRANSIENT-ROD-DRIVE SYSTEMS

1. GENERAL MODIFICATIONS

The Spert IV control-rod drive system was designed and installed in 1961 as a part of the original facility [5]. Although several modifications have been made to various parts of the drive system in order to accommodate the requirements for the CDC control rods, the system is basically unchanged. The control-rod-drive system consists of an inverted screw jack, driven by a single variable-speed motor-transmission combination. The drive unit is mounted on a base plate positioned on a movable control bridge spanning the reactor vessel and independent of the core structure.

Changes in the existing Spert IV control-rod-drive system were those necessary to accommodate the longer CDC core and to match the CDC rod configuration. The drive base plate has been positioned on a newly constructed control-rod-drive support bridge which rests about six feet above the original control bridge. Essentially this has meant that the entire Spert IV control-rod-drive assembly and guide structure has been raised about six feet. Figures A-1 and -2 illustrate the CDC control-rod-drive system.

The eight control rods are coupled in pairs to the rod shafts by means of four individual electromagnets and armatures suitable for underwater service. Although the control rods operate from the same motor-transmission combination, the individual magnets permit raising or scramming individual or various combinations of the control rod pairs. De-energizing the magnets allows the control rods to fall, and they are accelerated through the initial portion of their downward travel by means of an adjustable spring. Scram time, as measured from initiation of scram signal to shock absorber contact, is about 300 msec. Both upper and lower limit switches are provided on the control rod drive to prevent overtravel.

The nuclear operational requirements of the Capsule Driver Core have necessitated the use of a fast transient-rod system which will allow all of the programmed excess reactivity to be added before peak power. Consequently, the existing Spert IV transient-rod unit has been replaced by an air-operated system somewhat similar to the system used during the Spert I Oxide Core Destructive Test Program [21].

The transient rod is raised or lowered entirely by an air piston arrangement without any assistance from a motor-transmission combination. It is held at the upper limit position by an air-operated latch which is mounted on the bottom of the rod-drive base plate. As a transient is initiated the transient-rod blades and shafting below the transient-rod latch are accelerated by this air piston arrangement. This accelerated drop permits reducing the travel time between upper rod limit and the transient-rod-shock-absorber contact (a distance of about 28 inches) to a minimum of about 80 msec.

The transient-rod-drive system consists of four basic parts. The spring shock, shock absorber, air cylinder, and piston sections are all included in one assembly, while the air supply system is a separate assembly. Figures A-3 and -4 show details of the transient-rod-drive system and latch.

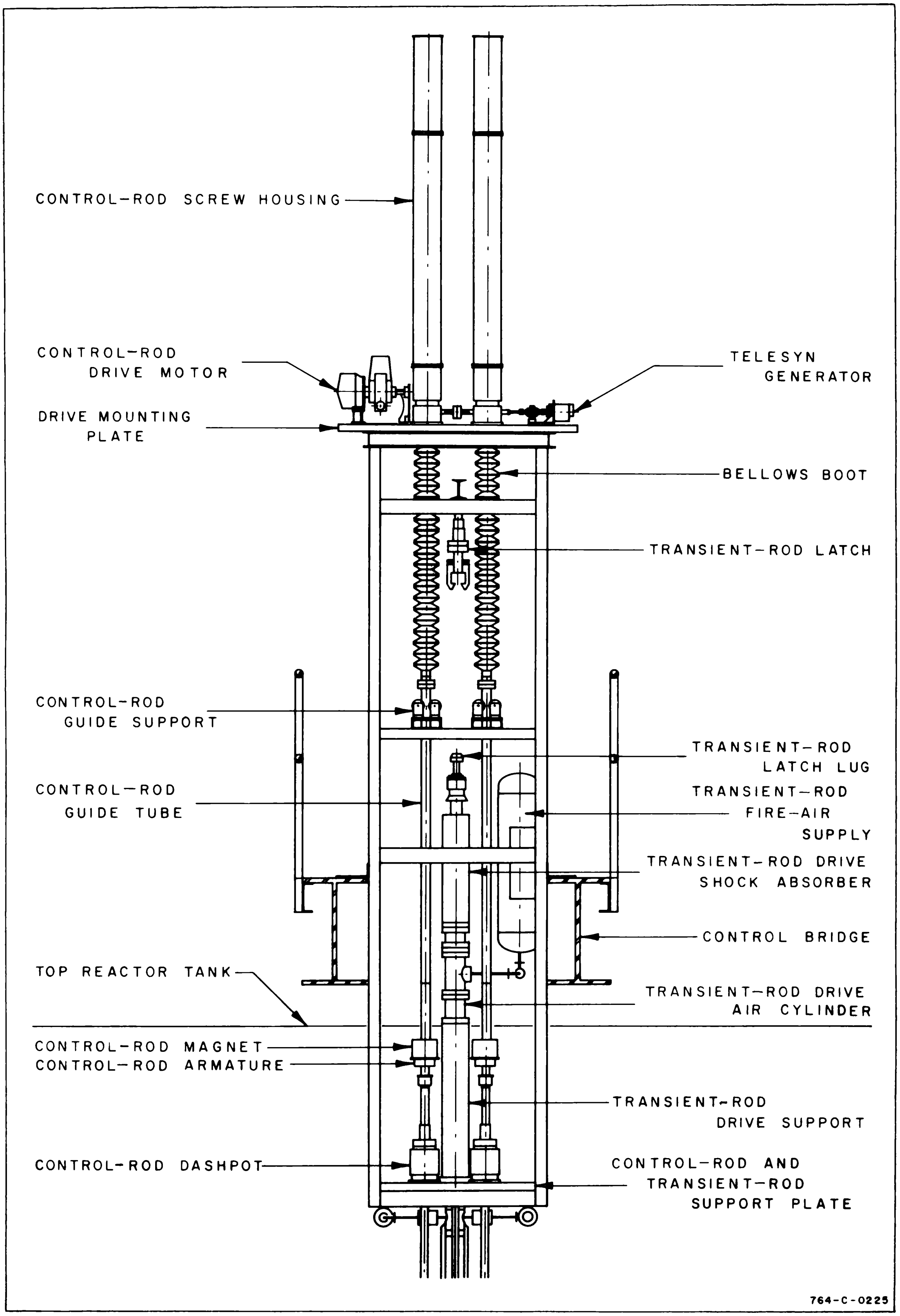

Fig. A-1 CDC control- and transient-rod-drive system.

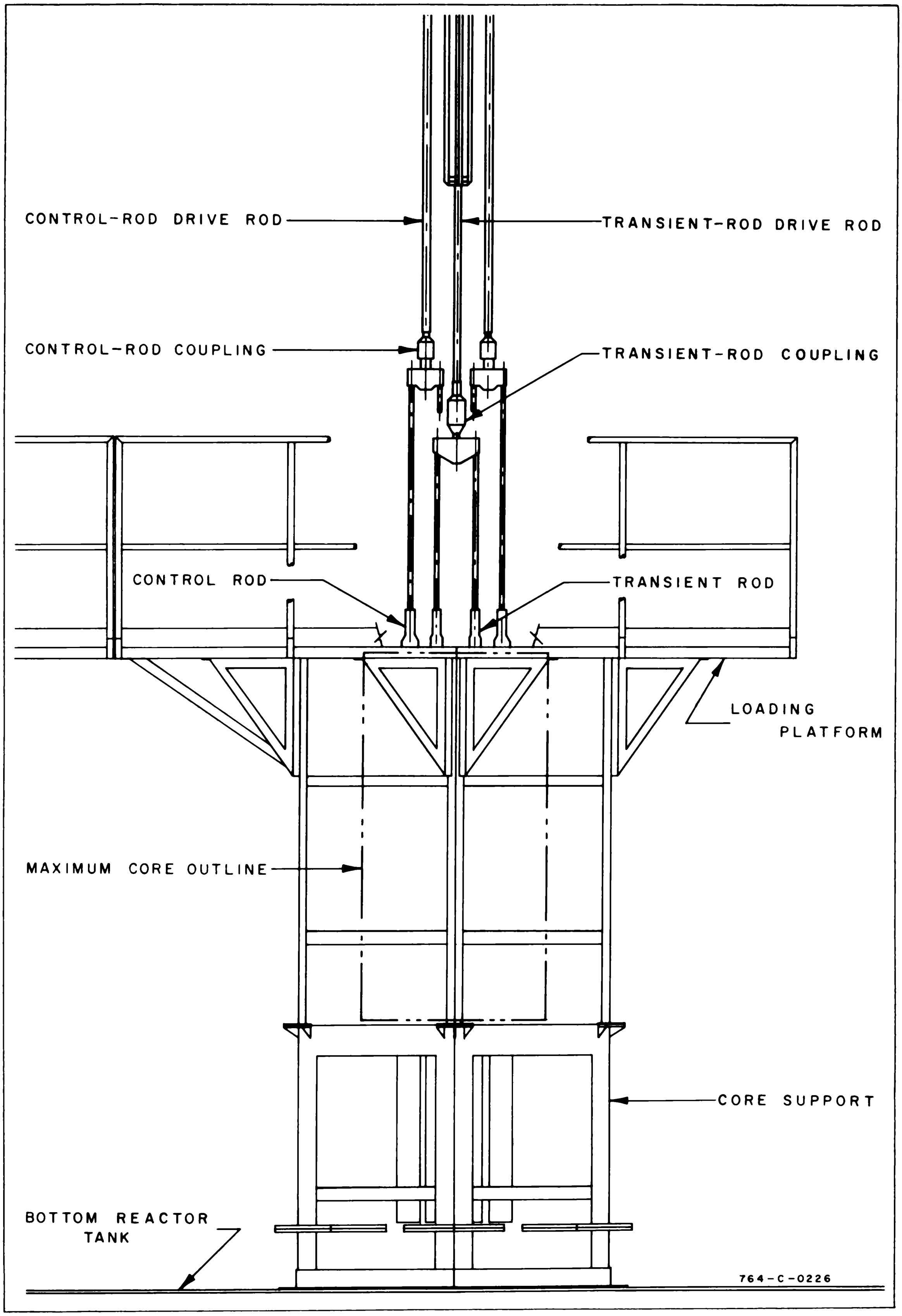

Fig. A-2 CDC core, core support, and rods.

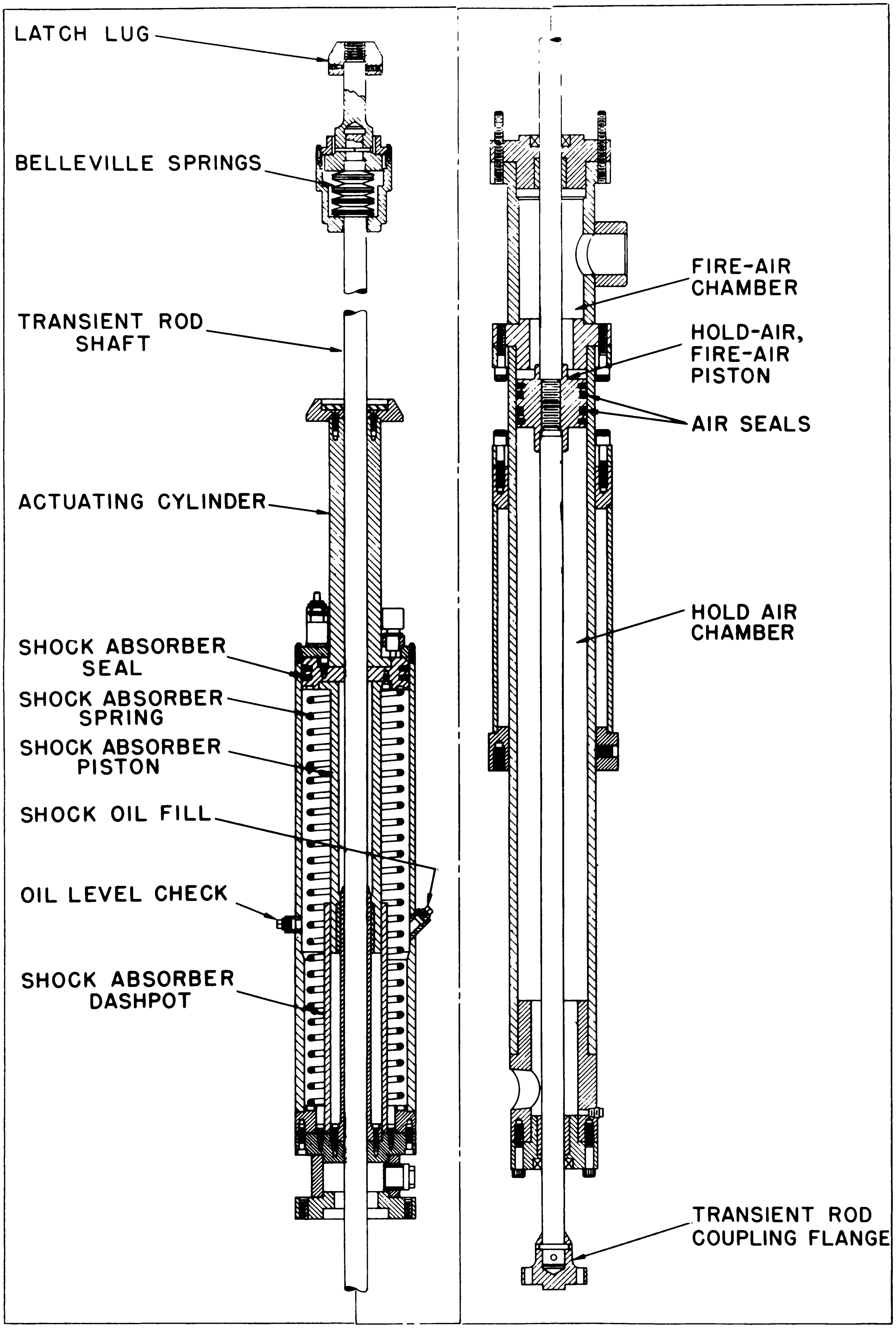

Fig. A-3 Transient-rod piston and shock absorber mechanism.

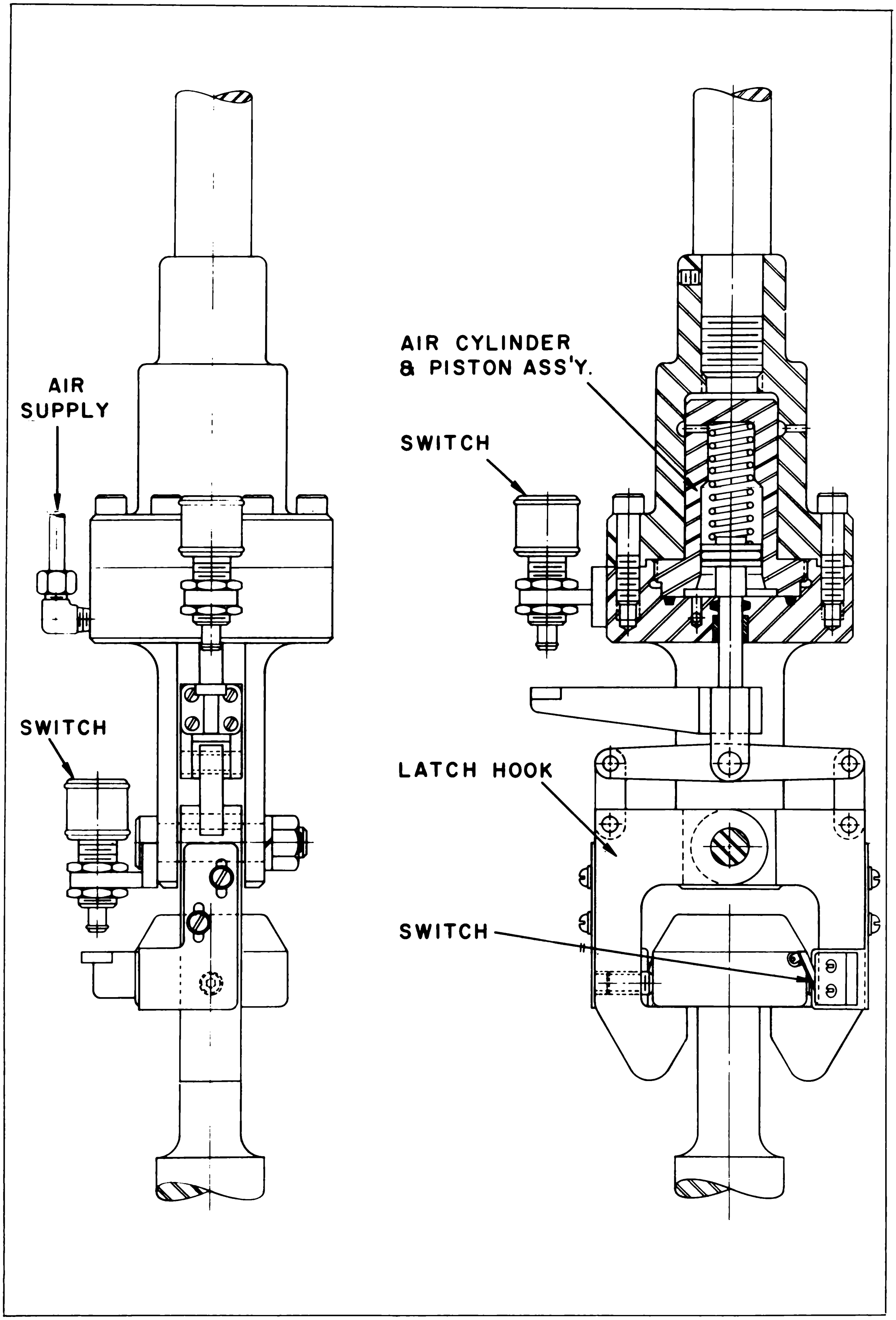

Fig. A-4 Transient-rod latch.

The air piston is a two-way piston with hold-air on the lower side of the piston and fire-air on the upper side of the piston. The transient rod is raised to its upper limit position by means of a differential air pressure applied across the piston with the pressure in the hold-air chamber being greater than the pressure in the fire-air chamber. An air-operated latch mounted below the control-rod-drive mounting plate holds the transient rod in the upper limit position. The latch is in the locked position until just prior to firing time. In this manner, an inadvertent rod drop, which would increase the reactivity of the system, is avoided.

In operation, the hold-air will be set somewhat higher than the fire-air to assure correct positioning of the transient rod prior to initiation of a transient. Initiation of a transient is accomplished by opening the latch then quickly expelling the hold-air from the system. The transient rod is accelerated downward by the fire-air. This downward acceleration is active over the first 28 inches of travel of the transient rod. After the 28 inches of downward acceleration, the transient rod is decelerated for an additional 8 inches, providing 36 inches of total rod travel.

The fire-air pressure and hold-air pressure are provided by high pressure air from a 600 psig source. Parallel-mounted bottles of oil-pumped nitrogen may be used in place of the air if desired. Both hold-air and fire-air pressure are regulated through manually adjusted air valves. The actual condition of the pressures may be monitored remotely at the control center panel through a system of preset limit switches indicating high or low pressures. The fire-air system incorporates an air reservoir to provide an adequate supply of high pressure air to the air piston and to allow for expansion of the air as the piston is displaced.

The major deceleration of the transient rod is accomplished through an oil-filled piston and sleeve shock absorber. The initial deceleration shock is dissipated through series-mounted springs in the latch lug assembly.

Each of the four transient rod blades is made up of two sections. The aluminum upper or "follower" section which is about 80 inches long, as measured from the top of the core to the lower poison section. The poison section is 28 inches long and consists of three strips of the same B_4C-Al material used in the control rods formed into a solid section cruciform, 2-1/8 inches by 3-3/8 inches by 0.245 inch thick.

2. CONTROL SYSTEM DESIGN

This section is written as a replacement for Chapter VII, Control System Design of the Spert IV Facility report (IDO-16745) to indicate the control system changes which have taken place in order to replace the original motor-driven transient-rod system with the new air-operated system.

2.1 General Discussion

This section of the report is devoted to a discussion of the various components of the Spert IV reactor control system with particular emphasis on the functional operation of the items discussed. In order to establish a framework for such a descriptive discussion, consideration is first given to the various requirements which the control system must fulfill.

From a general viewpoint, the primary design requirements are that no hazard to personnel shall stem from system operation and that known risks to equipment shall be minimized, including those risks demanded by the experimental program. The control system must provide proper manipulation of control units and must furnish information on all operations performed and indications of equipment failures or improper operations. All functions should be performed in such a manner that any component failure which constitutes loss of control shall shut down the system automatically.

These control system requirements, which are a consequence of the purpose of the facility and therefore of its mode of operation, also must reflect, somewhat, the philosophy of operation of the facility. The purpose of Spert IV is to provide a facility in which experimental programs can be carried out to develop information on the kinetics of a variety of reactor systems and on the inherent physical mechanisms which affect the neutronic behavior and, thus, the safety of these reactors. The experiments which will be performed include transient power excursions, initiated by programmed reactivity perturbations. The Spert IV control system provides two primary means of rapid reactivity addition: ejection of a "transient" rod and fast withdrawal of the control rods, both of which are discussed in more detail below. Control rods in the existing Spert reactors are designed in such a manner that withdrawal of rods removes neutron-absorbing material. In some core designs, the rods also include a "fuel follower" so that control rod withdrawal also adds fuel to the core. The transient rod is essentially an inverted control rod of the first type and is used for the initiation of step-wise reactivity perturbations. Raising the transient rod draws neutron-absorbing material into the core and reduces reactivity of the system.

The philosophy of operation of the Spert reactors provides that no nuclear operation of the facilities be conducted with any personnel within one-third mile of the reactor. The control system design provides for operation of the facility from the control center building, which is approximately one-half mile from the reactor.

The variety of tests to be performed and the short test-time interval for most of the experiments performed led to the selection of a simple control system in which operation is strictly manual with no servo or feedback loops in the control system. Because of the short time scale for the tests, the individual functions required during a transient test (such as ejecting the transient rod, starting data recording and photographic equipment, and scramming control rods) are programmed on a sequence timer, and the test is initiated by starting the timer. The reactor operator is always under the direct surveillance of at least one other certified reactor operator who provides backup and, together with all other persons in the reactor room, has the authority and responsibility to scram the reactor in the event of any unanticipated situation.

Because the action of conventional power level or period scram circuits would, in many instances, compromise the acquisition of information for which the experiment is conducted, such scram circuits are not used in the control system design. Provision is made, however, for the inclusion of special scram circuits for specific experiments where operator fatigue might become a factor of the experiment. Permanent incorporation of such scram circuits in Spert IV would not only, in many cases, compromise the acquisition of information for which the experiment is conducted, but also would compromise the development of a proper operator attitude. The type of tests performed in the Spert reactors requires that the operator be cognizant of the safety implications of each individual act in the performance of a test. This attitude must carry over beyond the test to all activities such as fuel manipulations and changes in the reactor core, components, and control systems, as necessitated by the experimental programs. The development of this attitude can be inhibited severely if an individual believes that he can err and have his error automatically compensated for by an automatic-trip circuit. A reliance of protective devices, which frequently must be bypassed or for which the set points must be specified and adjusted prior to each test in order not to interfere with the test, actually would result in a less safe operation. The required attention span of the operator is very brief for most of the experiments performed. Thus, the need for feedback control and safety scram circuits, because of the possibility of operator inattention or fatigue, is obviated.

2.2 Description of Control System

In the following sections, 2.21 through 2.210, the control-rod design and operational systems are discussed in detail. Details of the transient rod and transient rod operation are discussed in sections 2.211 through 2.214.

2.21 Coupling Magnets. The Spert IV control rods are coupled to the drives by means of electromagnets. To scram the reactor, the magnets are de-energized, and the control rods are allowed to fall into the core by gravity after a spring-assisted breakaway. The electromagnets used for coupling the rods to the drives can be energized individually so that all rods need not be withdrawn at once. The transient rod is integral with a piston-cylinder drive system and therefore has no coupling magnet.

The electromagnets of the Spert IV drives are cylindrical in design, with an outer diameter of five inches. An axial section of the core and armature is of conventional E-1 appearance. Twelve radial slots are milled partially through the inner and outer shells to impede circulating currents induced upon de-energization, thereby decreasing the release time. Saturation current for the 18-ohm coil is about 0.6 A, which provides a lifting force of about 900 pounds. Normal operating current is expected to be about 0.1 A, which will lift 300 pounds. Release times under operating conditions are expected to be less than 50 msec. The mechanical description of the magnets and performance curves are given in Section VIII of Reference 5.

2.22 Screw Jack. The basic element of a Spert IV control rod drive unit is an acme-thread screw jack, the mate of which is a worm wheel driven by a right angle worm. The outside diameter and pitch of the screw jack are 1-1/2 inches and 3/8 inch, respectively, resulting in a mechanical efficiency of roughtly 35 percent for the jack, and an over-all efficiency of 15 to 20 percent for the drive. A screw or worm which is less than 50 percent efficient is "nonoverhauling" or self-locking. This "inefficient" mechanism was chosen for its self-locking feature. The drive is simple, compact, and rugged. The speed ratio of the worm and wheel is 6:1. Thus, 16 revolutions of the input shaft are required for 1 inch of linear motion of the screw jack.

2.23 Mechanical Power. The mechanical power for the control rod drive is obtained from a 1-horsepower, 4-pole, 3-phase induction motor, operating on 208-volt, 60-cycle power. Constancy of speed, overload capacity, ease of control, and low cost are among the characteristics which make induction motors eminently suitable for the rod drives. Ease of reversing and the absence of auxiliary starting windings and switches dictated the choice of polyphase over single-phase motors. This motor is mounted integrally with a Graham variable speed transmission, with an output speed range of 0 to 200 rpm. The driving sprocket is identical in number of teeth with the driven sprockets. Both the motor shaft and the transmission output shaft are equipped with spring-set, magnetically released disc brakes to limit coasting. The maximum rate of 200 rpm produces rod motion at the rate of 12-1/2 inches per minute.

2.24 Rod Position and Speed Indication. A Telesyn self-synchronous motion-transmitting system is geared to the worm shaft of control rod screw number four. This operates a Veeder-Root digital counter at the control console which is calibrated to indicate the rod screw position in hundreths of inches. The speed ratio control of the Graham transmission is operated and monitored by a similar Telesyn system with a pilot motor driving a Telesyn transmitter at the control relay rack which, in turn, drives Telesyn receivers on the Graham transmission and on a monitoring Veeder-Root counter at the control console.

2.25 Control System Power Circuit. The NRTS electrical power standard, for applications up to 100 horse-power, is 480-volt, 60-cycle, 3-phase. Delta-connected ungrounded secondaries are in general use because a double fault is required to disable such a system. However, ungrounded secondaries have the disadvantage of being able to accumulate static charges. Experience with unusual equipment failures at Spert III, for which such charges seem to be the only plausible explanation, led to the choice of a Y-connected secondary with a grounded neutral for the Spert IV facility.

Experience with damage to 480-volt wiring which occurred in connection with a steam leak at Spert III lead to the choice of 208-volt, Y-connected power for the Spert IV control system because the Spert IV building can be expected to provide a high humidity environment on occasion. Most of the Spert IV control system operates directly from the 120/208-volt power, furnished from a single 6-kVA bank of transformers, with small 6- and 28-V transformers being required only for panel indicator lights and annunciator buzzers. System voltages are limited to 120 V above ground. Full load currents of the drive motors are less than 4 A each.

The control system 3-phase, 480-V primary power (Figure A-5) is obtained from the main bus of the reactor building motor control center through a 40-A trip, 100-A frame, air circuit breaker in cubicle D4. This bus is fed directly from the Spert IV substation through an 800-A air circuit breaker with 50,000-A interrupting capacity. Three leads from the input terminals run to a 600-V barrier type terminal strip with 24 terminals which serves as a junction block for the 3-phase transformer bank. Each transformer is a dry-type, four-winding, 2-kVA unit with two 240-V windings and two 120-V windings, connected in series and in parallel, respectively, to provide step-down from 480 to 120 volts. The three transformers are connected delta-Y to provide 120/208-V power for control relays and drive motors. The neutral and 3-phase power leads from the transformer secondary are designated ϕ_0, ϕ_{1a}, ϕ_{2a}, and ϕ_{3a}, respectively. The neutral is grounded solidly at the

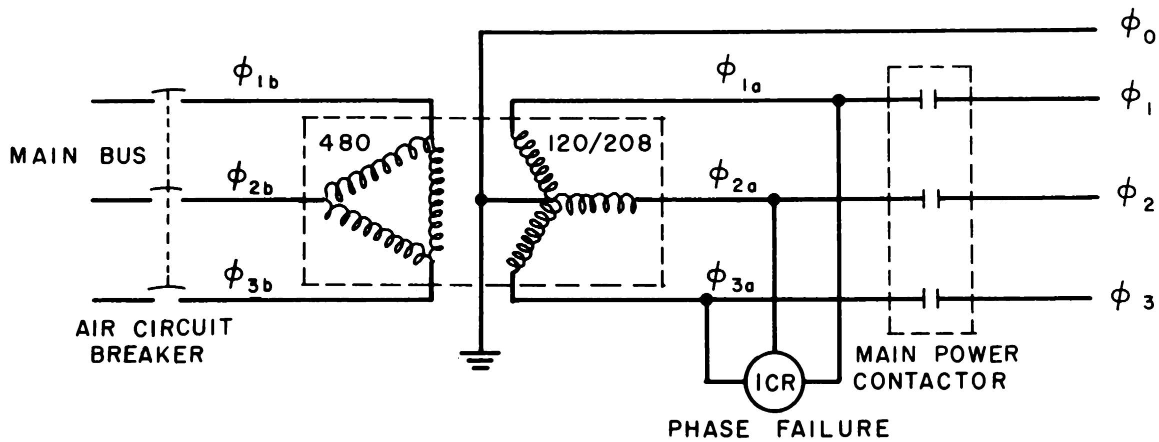

Fig. A-5 Control system power circuit.

relay rack and at the control console but is otherwise insulated and carried throughout the system, providing the option of a floating neutral in case this should ever be desired.

An induction disc-type undervoltage and phase failure relay, set to open at about 200 volts, monitors the transformer secondary. The control system is energized from the transformer secondary through an NEMA size 1 open-type magnetic contactor, designated the main power contactor. Downstream from the contactor, $\phi 1a$, $\phi 2a$, and $\phi 3a$, become $\phi 1$, $\phi 2$, and $\phi 3$, respectively.

As shown in Figure A-6, the main power keyswitch on the control console controls the main power contactor, subject to interlocks provided by contacts

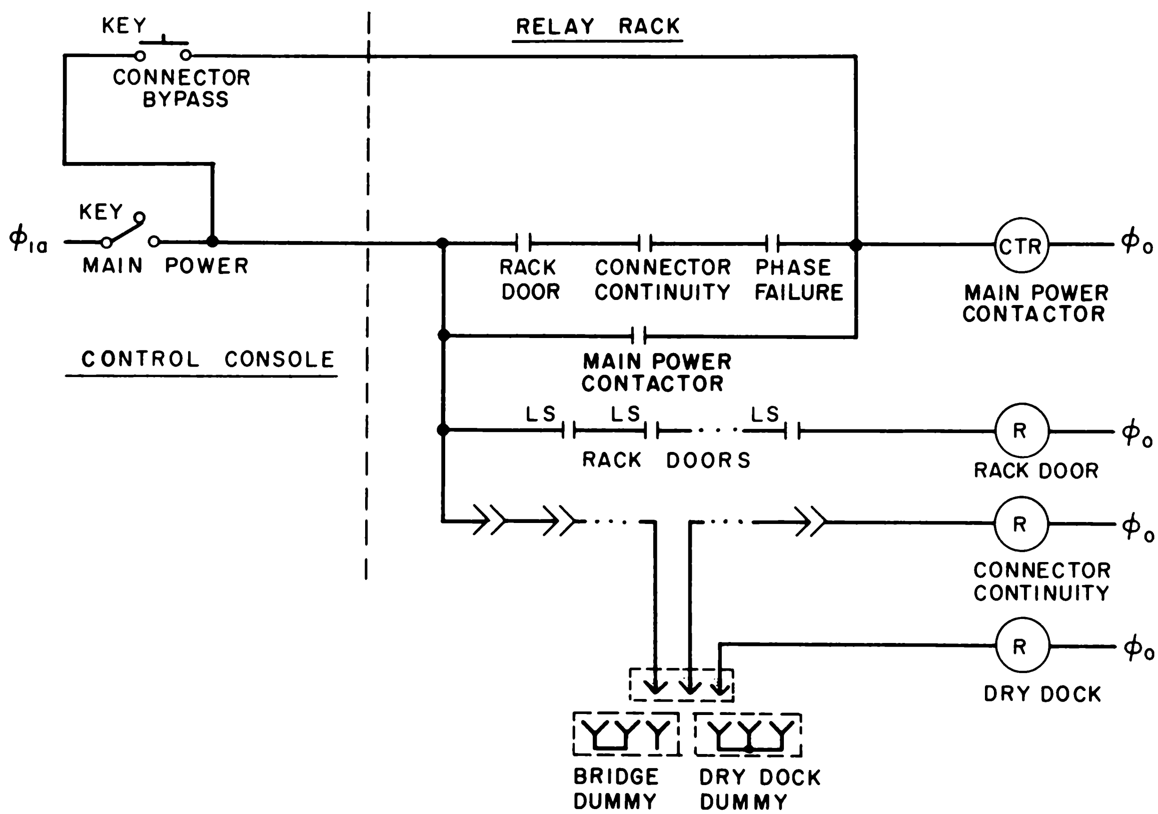

Fig. A-6 Control system power control circuit.

of a relay-rack door relay, a connector continuity relay, and the phase failure relay. The door relay is controlled by a series of switches actuated to close the circuit when the doors of the control system relay racks are closed. This is for safety of personnel working on the equipment in the racks. A red light, to indicate "power on", is also placed by each door handle. Similarly, the connector continuity relay is controlled by a circuit which traverses all cable connectors in the control system in such a manner that disengaging or mis-engaging any connector in the system interrupts the circuit. Those connectors in the system which are mechanically interchangeable are rendered electrically noninterchangeable by variation in the choice of pins used for this circuit.

Presence of any of these inhibitory conditions prevents turning main power on, but an auxiliary pole of the main power contactor is wired to prevent the loss of main control power by subsequent occurrence of any of these conditions. The console annunciator sounds a buzzer alarm and lights an identifying light if the main power keyswitch is at "on", and any of these inhibitory conditions occur. A momentary contact keyswitch is provided, designated the "connector bypass switch", to allow an operator to turn main power on despite the interlocks if it becomes necessary to check control circuitry with drive motor cables disconnected.

The dry dock relay, shown in Figure A-6, is energized by the connector continuity circuit through the dry dock dummy cable receptables when the drives are in dry dock. This relay is used in connection with the warning light and horn system described in subsection 2.215 beginning on page 59.

2.26 Control-Rod-Drive Insert -- Withdraw Circuit. A standard NEMA size-O reversing motor starter is used to control the 1-horsepower, 208-volt, 3-phase induction control-rod-drive motor. Electrically, the starter is comprised of two units, designated the control-rod-insert contactor and the control-rod-withdraw contactor. Basic control of these contactors is from a pistol-grip, insert-withdraw switch on the control console. The "off" position and insert position are maintained by detents. The withdraw position is spring-returned and must be maintained by the operator. No inhibitions are included in the insert circuit except for the lower limit indication which prevents mechanical damage to the drive if it were to be driven to its lower extreme. The electrical interlock, normally included in the reversing starter to prevent simultaneous energizing of both coils, has been altered as shown in Figure A-7, so that only the withdraw contactor is interlocked. The starter is mechanically interlocked in a standard fashion to prevent short circuits should the electrical interlocks malfunction. Each contactor has an auxiliary control relay (eg, control-rod-insert relay) operating in parallel with it. These are necessary for other control functions to be discussed elsewhere, but the control-rod-insert relay also serves an additional function here. In case of a malfunction of the control-rod-withdraw contactor, operation of the control-rod-insert relay by means of the control-rod-insert switch increases the certainty of interruption of control current to the withdraw contactor. Then, since the withdraw interlock has not been included in the insert circuit, the insert coil is free to operate, causing the drive motor to insert and, by means of the mechanical interlock, positively opening the withdraw circuit.

Contacts of the scram relay are provided in the control-rod-insert circuit to initiate automatic rundown of the drives following a scram. For power excursion tests, the sequence timer can be used to program either control-rod-

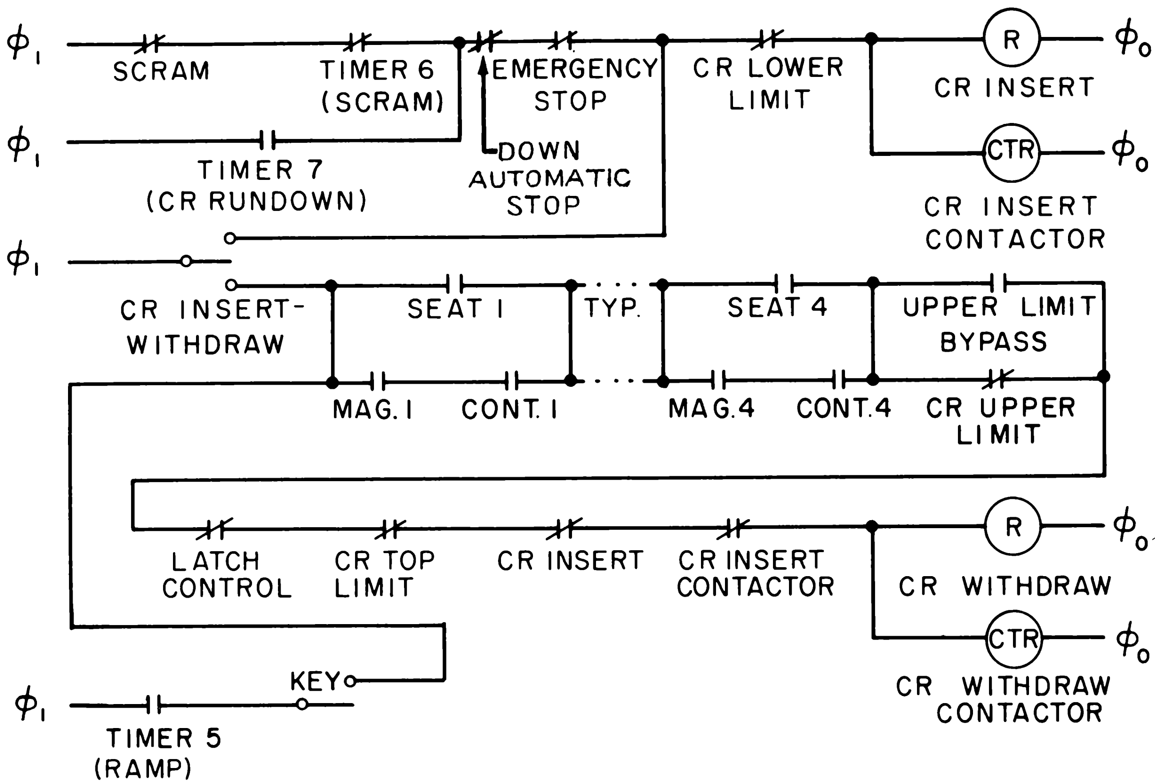

Fig. A-7 Control-rod system, insert and withdraw circuit.

drive insertion (timer 7 contacts in Figure A-7) or a scram to terminate the test. If the test is terminated by such a programmed scram, the automatic drive rundown is inhibited by the timer 6 contacts (Figure A-7). This feature is provided to prevent electrical noise, from the motor drives, from possibly reducing the quality of the data being obtained at this time from the various transient instrumentation channels.

Contacts of an "emergency stop" relay which is operated by a spring-return switch on the control console are provided to permit the operator to override an automatic control-rod-drive insertion and assume manual control of the insertion in the event of an obvious malfunction in the drive mechanism.

Limit-switch relay contacts are included in the withdraw circuit along with the electrical interlocks with the insert contactors. The "upper limit" switches indicate drive positions corresponding to complete withdrawal of the poison section of the control rods from the reactor core. The "top limit" switches indicate the extreme position of drive withdrawal, which is required in order to clear the bridge for removal of the drive system to dry dock.

The drive is not equipped with mechanical latches. Contact with energized magnets, as indicated by contact switch relays and magnet current relays, is required for control rod withdrawal. Selective withdrawal of individual control rods is possible since this requirement is bypassed for each rod individually by contacts of its seat switch relay.

In order to facilitate withdrawal of the drives to their mechanical limits for maintenance purposes, etc, an "upper limit bypass" relay has been provided which is operated by a maintained contact keyswitch on the control console. This upper limit bypass switch also deactivates the Klaxon horns and flashing red lights which normally operate whenever drives are raised from lower limit. It also prevents the scram which normally would occur should loss of transient rod contact be indicated without the fire relay being energized. The key for this upper limit bypass keyswitch is removed from the switch for all normal operations and is kept under the administrative control of the Nuclear Test Section group leader. The annunciator on the control console alarms whenever the bypass switch is in the "on" position at the time the main power switch is turned on.

Because reactor control always must be maintained, even at the risk of individual component damage, the drive motor-overload relays serve only to actuate an annunciator alarm to warn of motor overheat. Thus, in emergencies, the motors may be operated even though the overheat alarm indicates risk of damage to the motors.

2.27 Control-Rod-Magnet Circuits. A full-wave, 3-phase bridge rectifier provides current at about 8 volts for the four rod magnets, as shown in Figure A-8. The rectifier is constructed from 6.3-volt, 1.2-A filament transformers and six 1.2-A germanium diodes. No filtering is necessary. Current to each magnet can be monitored by a shunted galvanometer and adjusted by a rheostat. If, at a later date, power level and/or period scrams are required, a suitable power amplifier can be inserted in the circuit as shown at "A" in the magnet 1 circuit of Figure A-8.

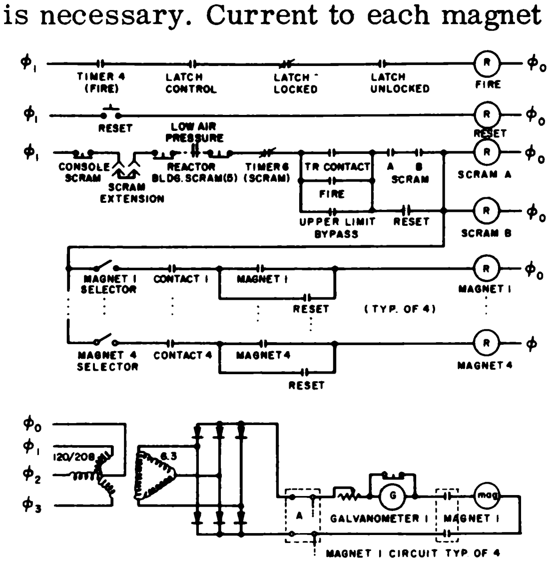

Fig. A-8 Control system magnet-control circuit.

The scram circuit is similar to an ordinary motor starter circuit with a multiple stop-button station, except that two parallel relays are used and holding contacts of each are in series. Either relay is able to scram the reactor despite malfunction of the other. Manual scram buttons are permanently installed at the control console and at five locations in the reactor building. Extension cord jacks are provided at the control console for additional hand-held scram buttons in the console room. Contacts of timer relay 6, operated by the sequence timer, provide for programmed scrams. Automatic scram is provided in case of a reduction of plant air pressure to less than 20 psig.

Control rods are selected for withdrawal by closing appropriate "magnet selector" switches on the control console. Actual energization requires the rod to be in contact with the magnet, as indicated by the magnet contact switches, and is accomplished by operating the "scram reset" button after magnet selection. Thus, selector switches can retain a given configuration through successive runs of the reactor, but resetting of the magnets requires deliberate action by the operator on each occasion.

2.28 Control-Rod-Drive-Speed-Control Circuits. The variable-speed transmission in the Spert IV rod drive is equipped with a remotely controlled electrical Shaftrol speed-ratio-changing mechanism. As supplied by the manufacturer, this Shaftrol unit is driven by a reversible-induction stall motor, and its action is monitored by a remote indicating voltmeter which receives its signal from a variable-resistance potentiometer, mounted with and actuated by the Shaftrol gear train. This monitor is not sufficiently precise for Spert purposes. The voltmeter and potentiometer have been replaced by a Telesyn synchronous-motion transmitter driving a Veeder-Root digital counter. To minimize mechanical modification of the Shaftrol unit, a Telesyn receiver was substituted for the stall motor in each Shaftrol, and a Telesyn transmitter, driven by a pilot motor, was mounted in the relay rack, with an additional Telesyn receiver at the control console driving the monitoring digital counter. The stall motor no longer suffices for pilot duty, because the Shaftrol gear train is likely to be damaged at stall by the additional torque required to drive the digital counter at high speed. Therefore, an ordinary shaded-pole-induction motor is now used for pilot duty, and a limit switch is installed in the Shaftrol to prevent damage. Reversing is accomplished by switching transmitter Telesyn stator leads, with a delay being incorporated in the pilot motor starting circuit to allow time for the transmitter Telesyn to stabilize before assuming load. The speed-control circuit is shown in Figure A-9.

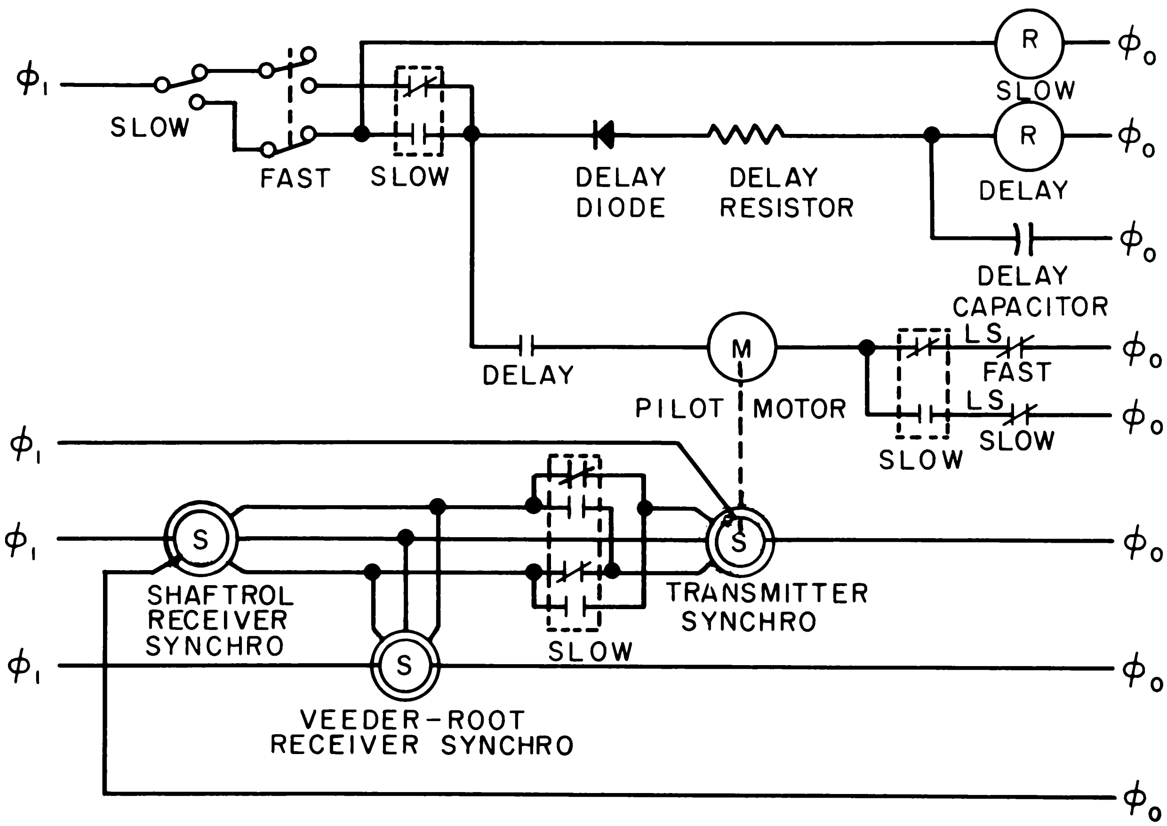

Fig. A-9 Control-rod system, rod-drive speed-control circuit.

Because the transmission speed ratio is not a linear function of Shaftrol rotation, no attempt has been made to choose gear ratios to make the digital counters direct reading. A calibration curve is required to interpret counter readings.

2.29 Control-Rod-Drive Sensing-Switch Circuits. Circuits for magnet contact switches and rod seat switches are shown in Figure A-10.

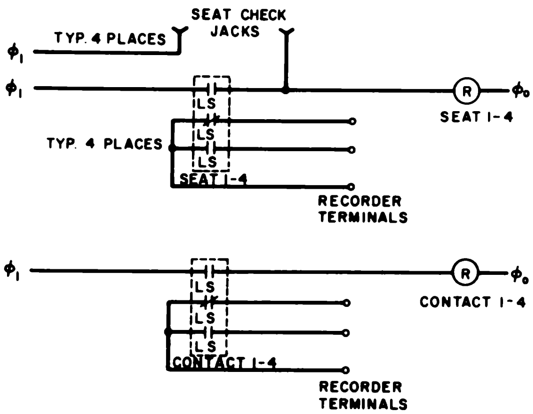

Fig. A-10 Control-rod system, rod-drive sensing-switch circuits.

The seat switches operate control relays, which operate control console indicator lights and function in various interlock circuits. Both normally "open" and normally "closed" circuits are brought directly from the seat switches for use in the recording of time references on the recording oscillographs. In addition, plug-in jacks at the control console are wired directly to the seat-switch relay circuit in such a manner (as shown in Figure A-10) that, in event of a power failure, the position of the seat switch on each rod can be checked by an ohmmeter or continuity tester to determine whether the rods are seated in the core. A reading of 900 ohms, the resistance of a seat-switch relay coil, indicates the seat switch is open and the rod is not seated. A line resistance reading of about 2 ohms indicates the seat switch is closed.

Magnet contact switch circuits are identical to seat switch circuits, except for the continuity-testing feature. Rod-drive-limit switches (not shown) merely operate control relays, no provision being necessary for the recording of limit switch signals.

2.210 Control-Rod-Jam Alarm Circuit. It is conceivable that a rod might possibly stick and not fall to seat position when released for scram, and the action taken in such a case depends on the particular circumstances at the time. A circuit is provided (as shown in Figure A-11) to sound an annunciator alarm if any rod fails to reach seat within 2 sec after being released from its magnet. The rod-jam relay is a thermal relay which "picks up" after receiving a steady signal for 2 sec. If any rod sticks between magnet and seat so that either its magnet relay or its seat relay is energized, the rod-jam relay receives a signal.

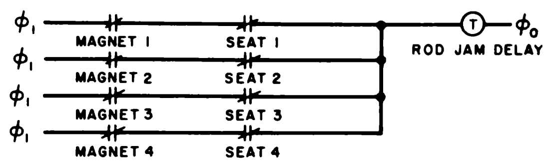

Fig. A-11 Control system, rod-jam alarm circuit.

2.211 Transient Rod Components. The operation of the transient rod differs from the control rod operation in that there is no motor-driven mechical drive system associated with it. The transient rod is raised and lowered by air pressure and has only two static, or positive, locations (viz, the upper and lower limits). Thus no Veeder-Root type position indicator is incorporated in the transient rod circuitry. Indications of upper and lower limit are provided on the control console.

The system is comprised of four basic parts. The spring shock, shock absorber, air cylinder, and piston sections are all included in one assembly previously shown in Figure A-1. The air supply system is a separate assembly.

The air piston is a two-way piston with hold-air on the lower side of the piston and fire-air on the upper side of the piston. In operation, the hold-air is set 20 psi higher than the fire-air. This assures correct positioning of the transient rod prior to initiation of a transient.

The transient rod is raised by applying regulated plant "lift" air to the hold-air side of the piston. The rod is then lifted until the latch lug comes into contact with the latch (Figure A-2) which is permanently fastened to the upper bridge structure. Latching is automatic, since the latch itself is normally closed. Once the rod is in the position, both hold-air and fire-air are applied to the sytem. The lift-air pressure is set at about 40 psi, just sufficient to gently raise the rod.

As shown in Figure A-2, the latch is opened by the application of air pressure. The latch is maintained in the locked position until just prior to initiation of the transient. Initiation is accomplished by opening the latch and dumping the hold-air from the system into the atmosphere. This allows the transient rod to be accelerated downward by the fire-air. The overall travel of the rod is 36 inches, with the first 28 inches of travel being accelerated, and the last 8 inches of travel being decelerated in the shock absorber.

The initial deceleration shock is dissipated through 10 Belville-type springs, mounted in series. Additional deceleration of the transient rod is accomplished through an oil-filler piston and sleeve shock absorber.

The fire-air pressure and hold-air pressure are provided by two parallel-mounted bottles of oil-pumped nitrogen and are regulated through manually adjusted air valves. The pressures may be monitored remotely at the control panel through a system of preset limit switches indicating high or low pressures. The fire-air system incorporates an air reservior, located close to the piston, to provide an adequate supply of high pressure nitrogen to the air piston and allow for expansion of the nitrogen as the piston is displaced.

2.212 Transient-Rod Insert-Withdraw Circuit. Since the transient rod is raised and lowered by air pressure, the associated electronic circuitry is quite simple and is shown in Figure A-12. A pistol-grip switch, similar to the one previously described, is used for control.

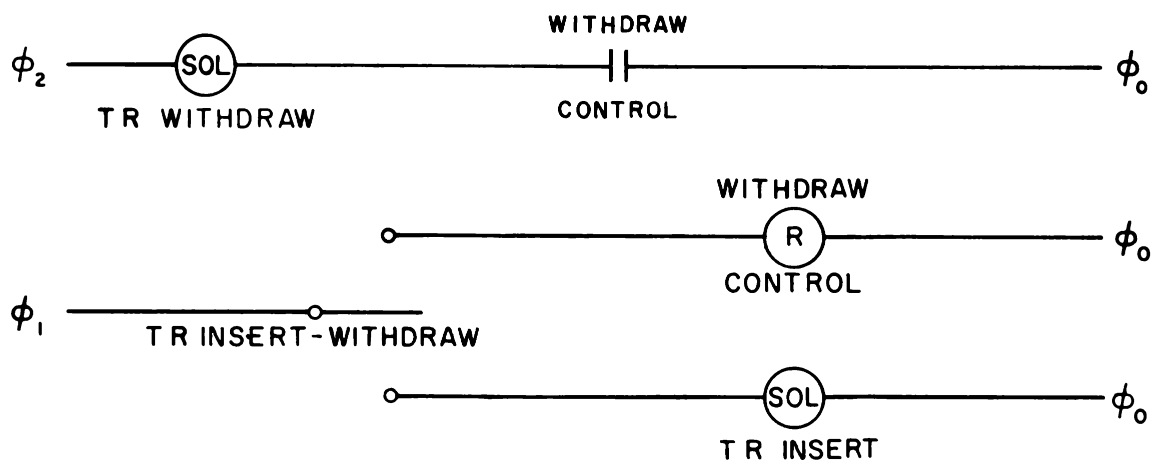

Fig. A-12 Transient-rod insert and withdraw circuits.

The transient rod is withdrawn by energizing the withdraw control relay, which, in turn, energizes the solenoid-controlled air valve. This allows "lift-air" to be applied to the hold-air side of the air piston.

The transient rod is inserted by energizing a solenoid-controlled air valve which allows the "lift-air" to slowly vent (bleed) from the hold-air side of the air piston.

2.213 Transient-Rod Air-Control Circuits. The transient-rod air-control circuits are shown in Figure A-13. Air pressure to the system is controlled

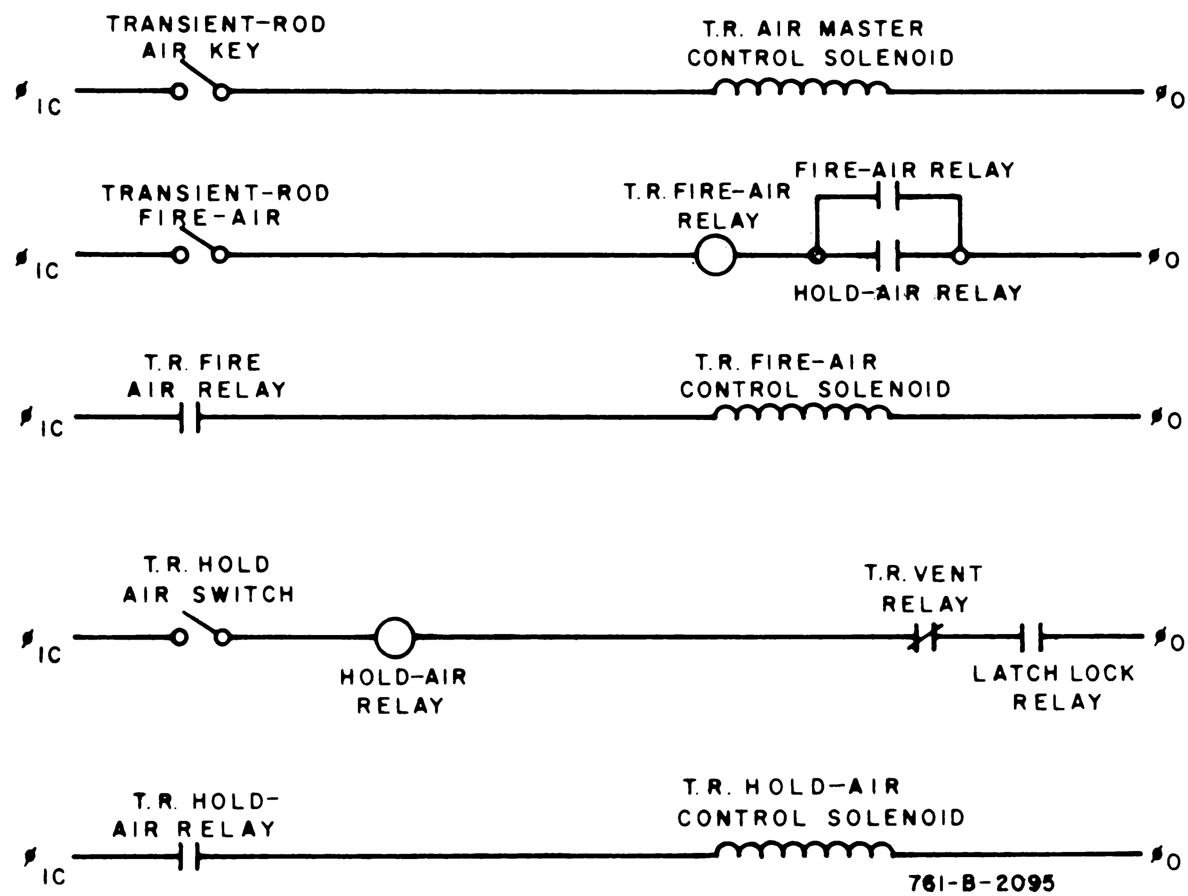

Fig. A-13 Transient-rod piston air-control circuits.

by a key-operated switch which actuates the transient-rod air-master-control solenoid. To prevent damage to the system, an interlock prevents the hold-air from being applied to the system until there is indication that the latch lug is securely locked in the transient rod latch. Then, hold-air may be applied to the bottom side of the piston by a push button on the console which actuates a relay and the hold-air control solenoid valve. Fire-air is applied in the same manner, except an interlock prevents turning on the fire-air until the hold-air is on. This prevents the fire-air from driving the rod down into the latch hooks and applying an unnecessary load to the latch.

The air pressure may be vented simultaneously from the hold-air and fire-air side of the piston by means of a vent switch installed on the console. As can be observed in Figure A-14, this switch energizes the transient-rod hold-air and fire-air vent solenoids. The fire-air vent solenoid is normally open to prevent an inadvertent firing of the transient rod in the event of a power failure. The hold-air and fire-air may be vented independent of each other by means of the hold-air and fire-air bleed switches.

The transient-rod-fire circuit, shown in Figure A-15, incorporates an interlock to prevent firing the rod unless the latch is open. Firing occurs by opening two solenoid dump valves in parallel, which rapidly vent the hold-air; in case of malfunction, either of these valves is capable of dumping the hold-air.

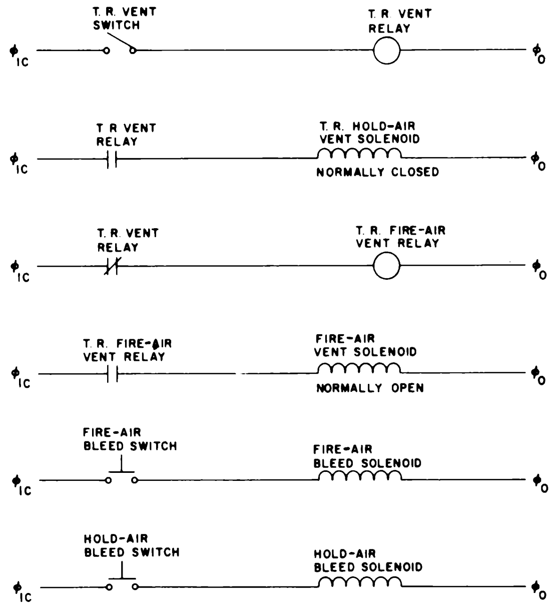

Fig. A-14 Transient-rod vent and bleed circuits.

2.214 <u>Transient-Rod Latch-Control Circuit</u>. The transient-rod latch-control circuit is shown in Figure A-16. The latch is normally closed, and it is opened by the application of air to the piston assembly. The air supply is controlled from the console and may be operated by three switches: (a) the latch-open switch, (b) the sequence timer, and

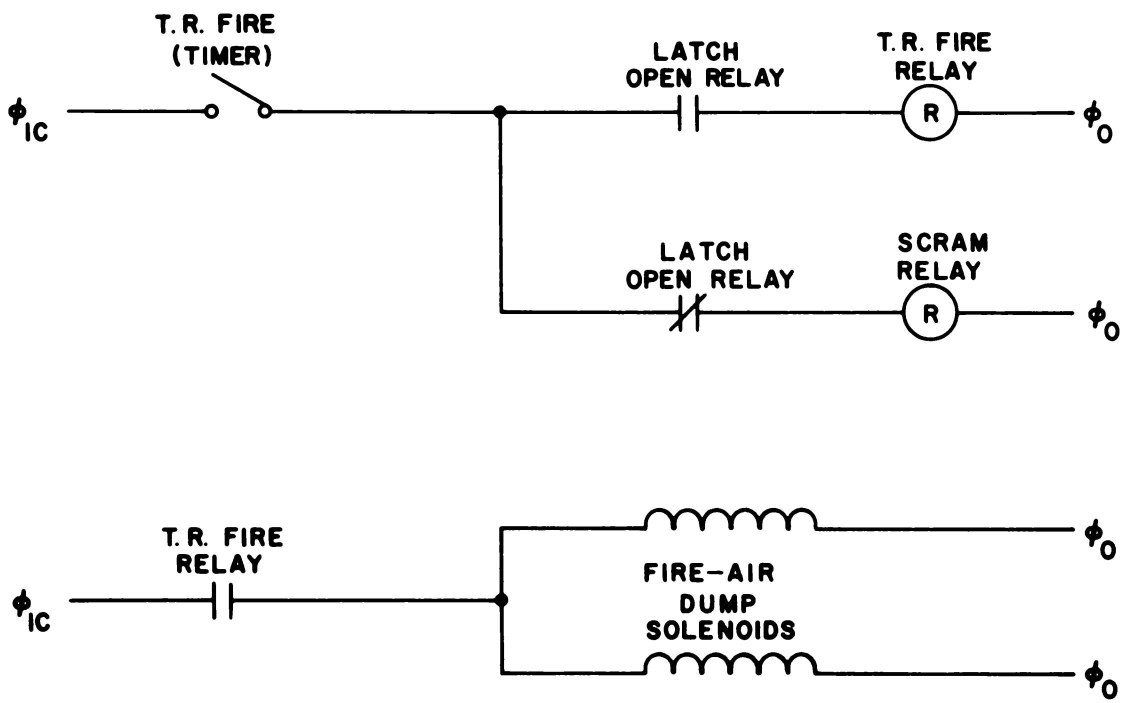

Fig. A-15 Transient-rod-fire circuit.

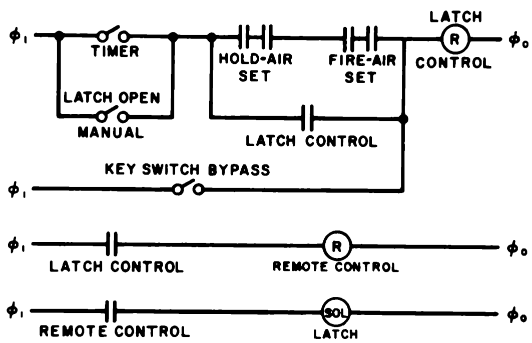

Fig. A-16 Transient-rod-latch control circuit.

(c) the latch-open bypass keyswitch. For safety purposes, indications of both hold-air and fire-air pressure must be obtained before the latch can be opened by either the latch-open switch or the sequence timer. The latch-open bypass key bypasses the hold-air and fire-air air-pressure-set interlocks and is used to rlease the rod from the latch when only lift-air is applied to the system.

2.215 Warning Lights and Horns. The reactor area has been provided with warning lights and horns which are part of the reactor control system.

Whenever control power is on, as it must be to operate the rod drives, if any rod is raised from seat position or if the control rod drive is raised from lower limit position without the use of the upper-limit bypass keyswitch, the warning system becomes active. The circuitry is shown in Figure A-17.

The horns, one interior and one exterior, are operated with building-lighting power by means of the horn relay. The horn relay may be operated by a manual horn button on the control console or by the red warning-light contactor. When operated by the red light circuit, the horn is limited to 15 sec of operation by a time-delay relay.

The horn relay is one of 18 relays used to control 120-volt reactor-building power circuits and is not an integral part of the reactor control system. The operating coil of these relays have very much higher impedance than the control relays used elsewhere in the system. Because of this, a bleeder-resistance shunt is used to prevent spurious operation by capacitive leakage in the long cable leading to the console horn button.

During operation of the reactor, a signal from any seat or lower-limit switch, indicating drive motion, energizes the horn relay and the red warning-light contactor. Provisions are made such that the warning-light contactors are not energized by the console manual horn button; and, as previously mentioned, the horn relay operates from the red warning-light contactor for only 15 sec. For servicing operations, in which the drives must be operated with personnel at the reactor, the upper-limit bypass switch is used to prevent activation of the red lights and horns. Actuation of the dry dock relay causes the seat switch or limit signal to activate yellow warning lights rather than the red warning lights and horns when the drives are operated in dry dock.

Phase-one power is used for most single-phase requirements in the control system. Because it is desirable not to have 208 volts present in the system wiring except where necessary, phase unbalance is tolerated. This is partially compensated by using phase-three power through the warning-light contactors to the light flasher circuits. The flashers are Reynolds units having motor-driven, cam-operated contacts. Because these contacts may be in closed position when the unit stops, it is not sufficient merely to control the flasher drive motors. The contactors had to be provided to cut off flasher power completely.

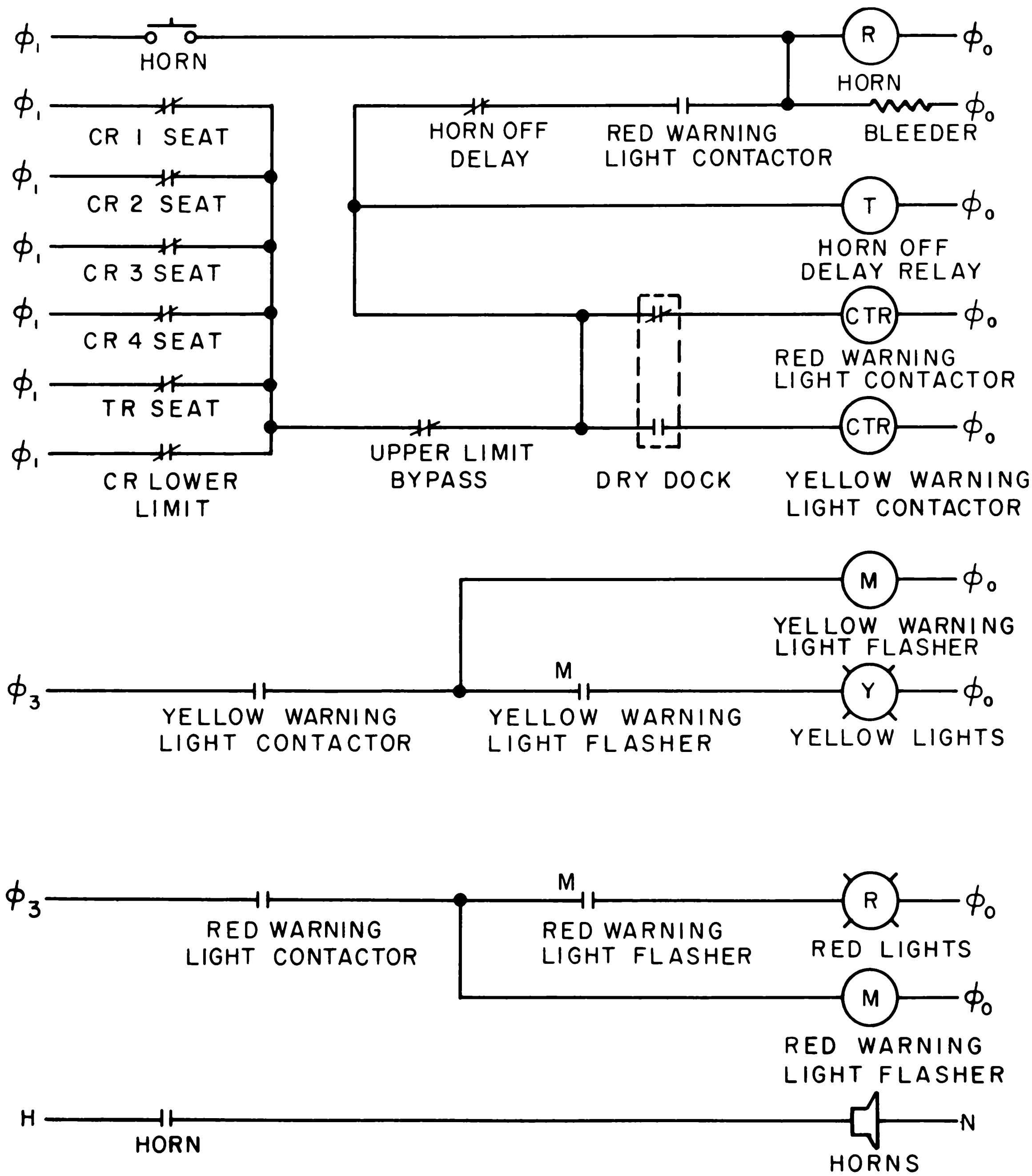

Fig. A-17 Control system warning lights and horn circuits.

2.216 Control System Electropane Annunciator Plug-In Unit. The control system alarm annunciator is an eight-unit Electropane assembly manufactured by Electro Devices, Inc. The schematic of a typical plug-in unit is shown in Figure A-18. The window of each unit contains three lamps, colored red, white, and green. Normally, the bulbs are series-connected to line voltage, giving a dim light indicating no bulb is burned out. When alarm occurs, all contacts (Figure A-18) close, applying full voltage to all bulbs and activating the external audible alarm. Closing the reset button opens the two contact pairs, designated reset, silencing the alarm and leaving only the red bulb lit. Clearing-the-alarm condition then opens the signal contacts and recloses one pair of reset contacts, lighting the green bulb. Closing the reset button again then returns the unit to normal.

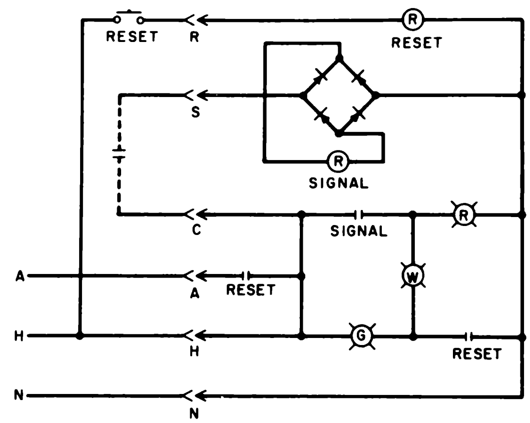

Fig. A-18 Control system annunciator plug-in unit circuit.

The operating of the signal coil can be reversed for use with normally closed external alarm sensing contacts.

2.217 Control System Annunciator Alarm Sensing Circuits. Alarm circuits included in the Spert IV annunciator are shown in Figure A-19. The annunciator is powered by control-center-building power through a separate pole of the control-console main-power switch, and a second delay relay is incorporated to allow control relays to assume correct indications before alarming the annunciator when main power is turned on.

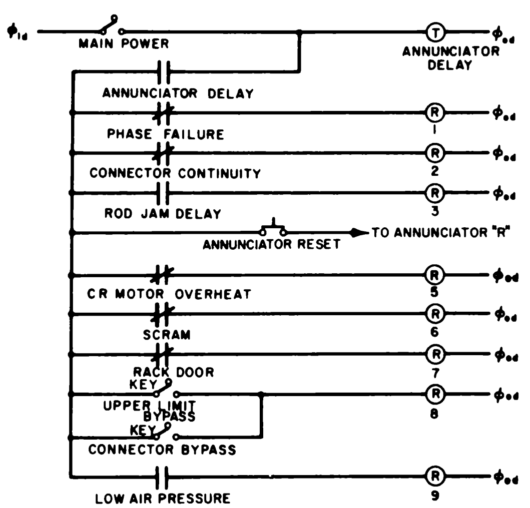

Fig. A-19 Control system annunciator alarm circuit.

The power-failure alarm sounds when low voltage or reversed-phase sequence de-energizes the phase-failure relay, closing a pair of contacts.

The connector-continuity alarm sounds when the connector-continuity relay is de-energized by opening the connector-continuity circuit.

The rod-jam alarm sounds when the rod-jam relays picks up, after any rod registers neither contact nor seat for more than 2 sec.

When the overheat relays in the drive-motor starters open, units four and five alarm.

Scram can be detected by watching seat, contact, or magnet-selector indicator lights on the console, but an annunciator unit monitors the scram relay to give immediate, unmistakable information in event of its being de-energized for scram.

Opening any relay-rack door drops out the rack door relay, causing alarm.

Use of either bypass keyswitch alarms the annunciator, to ensure against inadvertent use of these switches, and, in the case of the upper-limit bypass switch, to provide alarm when main power is turned on, if the upper-limit bypass switch has been left on from previous operation.

2.218 Sequence-Timer Circuits. The Spert IV control system is provided with Multiflex timers, manufactured by Eagle Signal Corp., for sequence programming of experiments. Two basically similar units, differing in range capacity by a factor of ten, are used together. Very precise timing is available for sequences requiring no more than 30 sec, while sequences requiring up to 5 min also can be programmed but with less precision. Settings can be made with accuracy within 1/4 percent of full scale.

The basic timer circuit is shown at the top of Figure A-20. This is an elaboration of what is referred to by the manufacturer as the "no voltage reset arrangement", protected against automatic restarting. The 30-sec timer and the 300-sec, or 5-min, timer are wired to operate from the same starting keyswitch and stop pushbutton. Energizing the clutch solenoid of each timer by means of the "start" switch engages a clutch and lowers the contact trip bars to ride on a sliding plate. The synchronous motors then drive the sliding plates downward. The contacts close and open as the trip bars drop off the downward-moving plates in accordance with their time settings. De-energizing the clutch solenoid disengages the clutch and raises the trip bars, allowing the sliding plate to reset, by spring action, to its original position. Section one timer contacts are used as a holding circuit, enabling a momentary switch to be used for starting so that the timer does not repeat its cycle automatically after resetting. Used in this manner, section one closing contact always must operate when the clutch is energized, and section one contact determines the length of the timer cycle.

Apparatus is not operated by the timer directly, but by seven timer-controlled relays at the control console and seven other functionally identical relays in the reactor building relay rack. These relays are designated timer 1 relay, timer 2 relay, etc and are controlled through selector relays by correspondingly numbered contact pairs of the timers. Timer relays 2 through 7 are shown near the bottom of Figure A-20. Timer 1 relay is shown with the basic timer circuit because of the internal use of section one contacts. The selector relays, as the figure shows, select whether the 30-sec or 300-sec timer contacts shall control a given timer relay. The timer-selector relays are in holding circuits with momentary selector switches. As is evident from the middle portion of Figure A-20, each selector switch, in picking up its relay, drops out the relay of its opposite member. To drop both of any pair of selector relays out at once, it is necessary to turn off timer-panel power. A timer-panel switch is provided so that the timers, with associated relays and indicator lights, may be left inoperative if desired when the control system is being used in nonprogrammed operations such as core loading. Contacts of timer 4 and timer 5 relays are used to bypass, and thus prevent, accidental function of both the timer panel

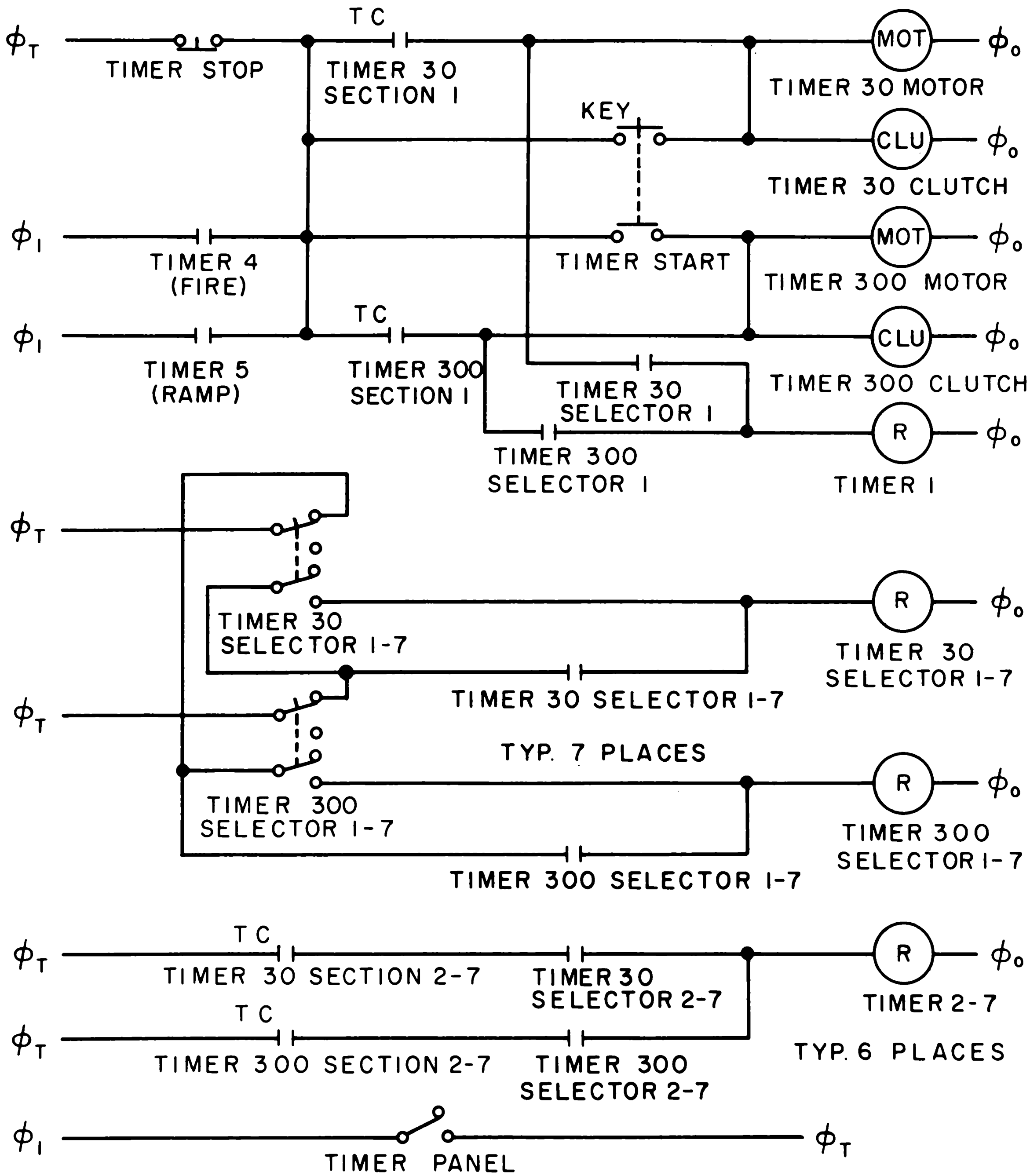

Fig. A-20 Control system sequence-timer circuit.

switch and the timer stop button during programmed transient power excursions, when actuated, by connecting the basic timer circuit directly to phase-one power as shown in the upper left portion of Figure A-20. Timer 4 relay is permanently assigned to control step transients and timer 5 relay is permanently assigned to control ramp transients.

3. CONTROL CONSOLE

The Spert IV control console is built from eight standard prefabricated 22-inch metal sections and two 45° "pie" sections. The two pie sections are inserted between the third and fourth and the fifth and sixth rectangular sections, so that the overall appearance is roughly that of a quadrant of a circle. The operator is seated before the two center sections which contain intercom control switches and the switches and indicating lights which, with the sequence-timer panel in section six, constitute the controls of the reactor. The remaining panels are occupied by television monitors, oscillographs, and nuclear instrumentation. Figure A-21 is a photograph of the complete reactor control console.

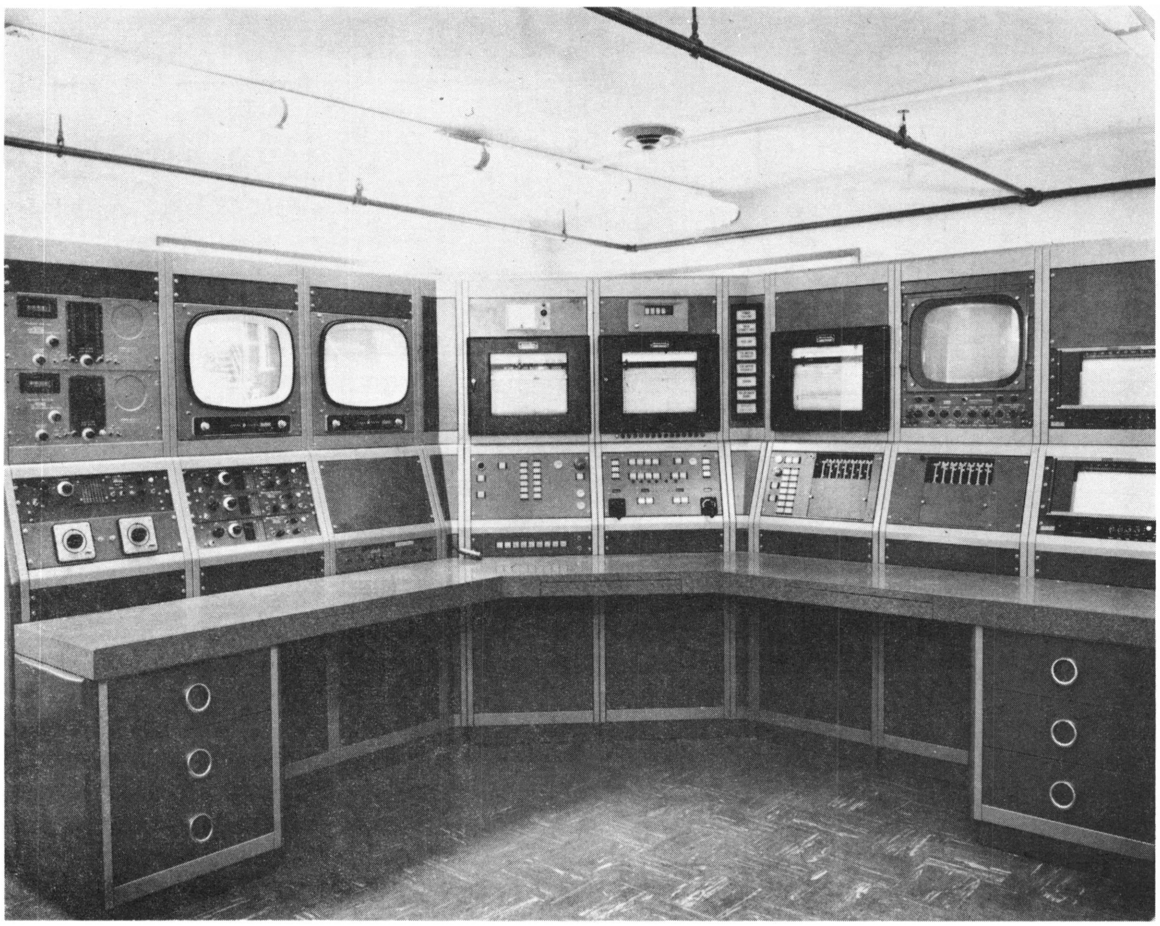

Fig. A-21 Photograph of reactor control console.

Twelve 19-conductor control cables and one Number 2 AWG, stranded, neoprene-covered, bonding cable are laid from the control console to the reactor building relay rack for reactor control. Sixty RG8A/ coaxial cables and a number of 19-conductor cables are used for instrumentation, process control, and intercommunication.

The reactor control panels are occupied by illuminated push buttons which serve for operation and indication for most of the controls. Two piston-grip switches, control-drive-insertion and -withdrawal, and six locking-key switches control critical functions such as main power, sequence timer, transient rod latch, and interlock bypasses. A red mushroom-head push button and a black flush-mounted push button are for scram and manual evacuation horn, respectively. Fixed indicator lights, identical in appearance to the illuminated push buttons, monitor the contact, seat, and limit switches.

Two indicator light colors are used: red and white. Fixed indicators use only white and indicate by illuminating at "on" and "off". Limit switches, contact and seat switches, and the ramp-selector keyswitch are monitored in this manner. Momentary push buttons, used for drive-speed control, are continuously illuminated in white only, merely for the sake of appearance and ease of identification.

The annunciator reset momentary push button is left unilluminated except when the annunciator alarms, at which time it is illuminated with a flashing white light.

The scram reset button (also momentary) is normally lighted white. When scram occurs, it changes to red. It is located just beneath the mushroom-head scram button.

Two sequence timer units are used, having ranges of 30 sec and 300 sec, respectively. Each timer has six sections with fully adjustable "on-off" times and a seventh which is not adjustable but is capable of controlling equipment desired to run coincidentally with the timer itself. The timer panel has its own power "on" switch, a maintained-contact push button, which allows the timers to be left inoperative if desired. This switch is continuously illuminated white when turned on, and the timer stop button, adjacent to it, is illuminated simultaneously, also white. Each section of the timer is provided with a selector push button which is illuminated white when selected and turns red when it becomes active during a run.

The selector switches are momentary push buttons which energize relay-holding circuits. Once picked up, a selector relay can be dropped out only by picking up its opposite member or by turning off timer-panel power. That is to say, for example, if timer 30, section two is picked up, the selection can be changed to timer 300, section two by actuating the corresponding selector switch, but to drop out both sections it is necessary to turn off timer power.

The timer panel is provided with a stop button so that a program sequence can be stopped if circumstances require preventing the initiation of a transient. However, if a program has proceeded far enough that a transient has been initiated, timer relays bypass both the timer "stop" button and the timer power switch so that impulsive or accidental action cannot then interfere with programmed data recording the scram.

Both timers operate from the single timer start keyswitch, but corresponding sections of both timers cannot be used in the same run. However, different sections of the two timers can be used in the same run. This arrangement was chosen so that the various sections could be permanently assigned to certain functions without eliminating the choice of the 30-sec or the 300-sec range for any given function. For example, section four of both timers is assigned to timing the firing of the transient rod. The selector switches determine which timer actually is used in a given run, but regardless of which timer is chosen to fire the transient, section six of either timer still may be selected to terminate the run with a programmed scram.

Magnet-selector switches are of the maintained-contact variety so that they can be left unchanged from run to run. These switches are normally unilluminated, but are lighted red when actuated for magnet selection and then turn white when the magnets are actually energized by the operation of the scram reset button. If any magnet loses contact with the rod for any reason, it is de-energized automatically, and its selector switch reverts to red.

Twelve auxiliary relays are provided in the reactor building relay rack which are operated by 12 maintained-contact "on-off" switches on the control console. These switches, normally unlighted, are white when in "on" position. The switches and relays are used for miscellaneous functions, such as remote control of high voltage to ion chambers and remote control of 120-volt and 480-volt reactor building power circuits.

A patch panel is provided in the relay rack at which any combination of circuits may be interconnected, involving the auxiliary relays, the timer relays, the remotely controlled building-power receptacles, and the instrument-room control cable. Eighteen 120-volt relays and one 480-volt relay are provided in the reactor building for controlling the evacuation horns, seventeen 120-volt power receptacles, and one 480-volt, 3-phase power receptacle, respectively. All of these may be controlled from the patch panel.

Two 4-digit Veeder-Root counters are mounted above the insert-withdraw switch and the drive-speed-control switch on the control console. These indicate withdrawal position and Graham transmission-speed setting of the control rod drive. The position indicator reads directly in hundredths of inches. The speed indicator reads in arbitrary numbers which must be referred to a calibration chart to give actual speed in inches per minute. This could not be made direct-reading by mere choice of gear ratios because the relationship between the drive speed and the speed control shaft angle is not linear.

APPENDIX B

NUCLEAR OPERATION REGULATIONS

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All nuclear operations conducted in the Spert IV facility are subject to two bodies of regulations consisting of the Spert IV Operating Limits and a manual of Standard Practices which, in general, establish additional, more restrictive limits than those set forth in the Spert IV Operating Limits. Both the Operating Limits and the Standard Practices set forth the minimum rules and instructions which must be followed by all Spert personnel in the performance of their duties in order to promote the safe and efficient operation of the Spert facilities and to provide the high degree of administrative control and cognizance which this requires.

Included in the Standard Practices Manual are specific detailed requirements concerning the following:

(1) Review and approval by Phillips Petroleum Company personnel of specific experimental test series proposals

(2) Safety analysis and test limits for each experimental test series proposal

(3) Nature and frequency of inspections of system components

(4) Instrumentation requirements for operation and maintenance activities

(5) Personnel requirements, certifications, and methods of certification for facility operation

(6) Formulation, review, and approval for all operational procedures

(7) Any other standard practices which Phillips Petroleum Company deems necessary to comply with the requirements of the Nuclear Safety Article of the operating contract.

APPENDIX C

RADIOLOGICAL CALCULATIONS

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In the radiological hazards analysis, an operating history was assumed in order to estimate a reasonable upper limit of radiological doses which would be incurred as a consequence of fission-product release during the subassembly testing program. Specifically, it was assumed that five transients per eight-hour day were conducted for a 100-day period. Each transient released 300 MW-sec of energy, except for the last which released 2500 MW-sec. This long operating history results in an estimate of the iodine and long-lived fission-product inventories which are reasonable upper limits.

1. INHALATION DOSE

The radiological analysis of the above operating history was performed with the aid of an IBM-7040 modified CURIE computer program[19]. This program calculated the isotopic inventory of fission products as a function of time after the last transient and also calculated the inhalation doses received at various downwind detector positions using Sutton's diffusion equations. The meteorological parameters used in these calculations were taken from published meteorological data applicable to the NRTS[22].

The inhalation dose calculations assumed instantaneous fission-product release at ground level, and doses were calculated along the center-line path of the cloud.

2. CLOUD-GAMMA DOSE

Considering the reactor as a point source, the total gamma dose received by an exposed person downwind from a fission product cloud released at ground level is given as[23]

$$D_\gamma = \frac{2\ Q(t)\ C_f}{\pi C^2\ \bar{u}(x)^{2-n}} \quad \text{rem} \tag{1}$$

where

$Q(t)$ = source strength, curies

C_f = conversion factor = $0.26 \ \frac{\text{rem-m}^3}{\text{sec-curie}}$

n = Sutton's stability parameter

C = diffusion coefficient for isotropic turbulence (meters n/2)

$\bar{u}$ = mean wind speed (meters/sec)

x = downwind distance (meters) = $\bar{u}t$.

The diffusion and stability parameters used in all dose calculations are listed below:

Lapse: $\bar{u}$ = 7 m/sec
C = 0.35
n = 0.2

Inversion: $\bar{u}$ = 2 m/sec
C = 0.03
n = 0.5

3. DEPOSITION GAMMA DOSE

The deposition dose received by a person located downwind from the fission product release described earlier was calculated using the method outlined in References 23 and 24. Neglecting depletion of the fission-product cloud by deposition of fallout, the total deposition dose is given by the following equation:

$$D_{dep} = \frac{2V_g \, C_f'}{\pi \bar{u} \, C^2 \, x^{2-n}} \int_{t_a}^{t_u} Q(t)dt \tag{2}$$

where

V_g = Velocity of deposition (m/sec) = 1 cm/sec

C_f' = conversion factor = 2.78 x 10^{-3} $\frac{\text{rem-m}^2}{\text{sec-curies}}$

Q(t) = source strength of solids and halogens, curies

t_a = time of cloud arrival, sec

t_u = upper limit of exposure, sec.

The other terms have been previously defined. The deposition source, Q(t), in the above equation was obtained from the modified Curie code. This calculated source was factored into four exponential components and integrated over the exposure-time interval to obtain the total number of disintegrations emitted by the deposited material. The expression for this integrated deposition source is

$$\int_{t_a}^{t_u} Q(t)dt \approx \sum_{i=1}^{4} \frac{Q_i}{\lambda_i} \left(e^{-\lambda_i t_a} - e^{-\lambda_i t_u}\right) \tag{3}$$

where

$t_u = t_a$ + length of exposure (sec)

Q_i = activity of the i^{th} component at t=0 (curies)

λ_i = decay constant of the i^{th} component (sec^{-1}).

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