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SPERT I DESTRUCTIVE TEST PROGRAM SAFETY ANALYSIS REPORT

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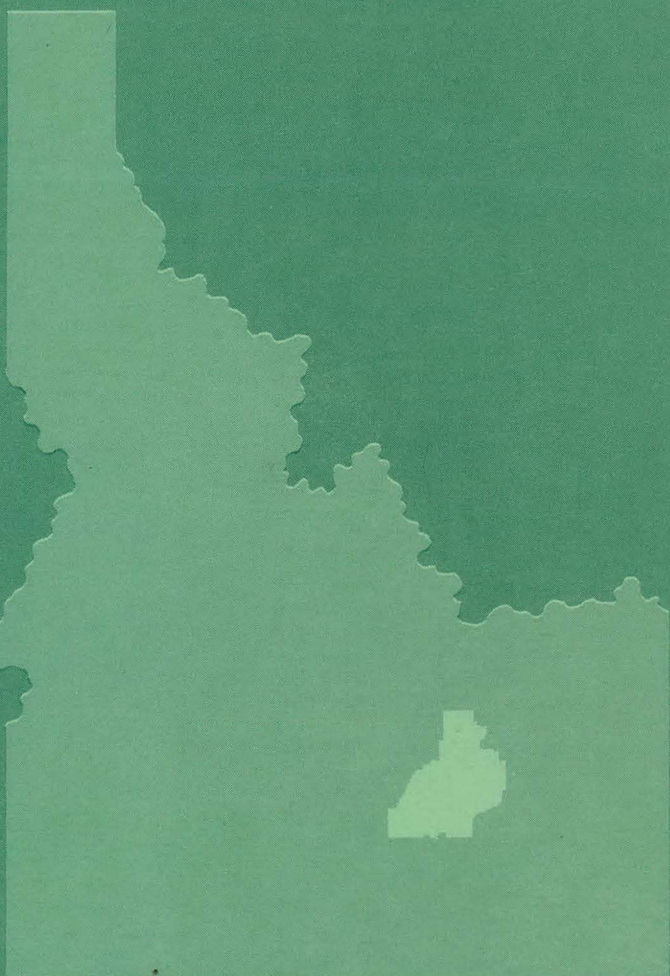

PHILLIPS PETROLEUM COMPANY

ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION

US ATOMIC ENERGY COMMISSION

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SPERT I DESTRUCTIVE TEST PROGRAM

SAFETY ANALYSIS REPORT

by

A. H. Spano and R. W. Miller

PHILLIPS
PETROLEUM
COMPANY

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SUMMARY

As an important part of the Nuclear Safety Research and Development Program being carried out by Phillips Petroleum Company for the Atomic Energy Commission, a series of core destructive tests will be initiated in the fall of 1962 at the Spert I reactor facility. This report reviews the objectives of a core destructive test program; the proposed experimental program of destructive tests on a water-moderated, plate-type core; the experimental results of non-destructive transient tests which have so far been obtained on the test core and the extrapolation of these results to the destructive case; an analysis of the hazards involved in performing such destructive tests; and a detailed description of the reactor facility and environmental conditions.

To date, the Spert experimental program has been devoted principally to investigations of the self-limiting power excursion behavior of reactors under runaway conditions. In particular, extensive power, temperature and pressure data have been obtained from many series of transient tests performed on several water-moderated, highly enriched, plate-type reactors, encompassing wide variations in core parameters and in system conditions. These test series have been carried out over a range of reactor periods, to the point where minor fuel plate damage first occurred. The proposed destructive test series represents not only an extension of the study of reactor response to large reactivity insertions beyond the threshold of core damage and into the region of limited and total core damage, but also the initiation of a systematic investigation of the consequences of reactor accidents. The core destructive program will constitute a series of tests of successively shorter period, resulting in increasing damage to the core and other reactor components. This will necessitate the replacement of fuel assemblies and instrumentation after each test.

The water-moderated core which will be used for these experiments is mounted in the Spert I open-type reactor vessel, which has no provision for pressurization or forced coolant flow. The core is a 5 x 5 array of highly enriched aluminum clad, plate-type fuel assemblies, using four blade-type, gang-operated control rods for reactor control.

Reactor transients are initiated at ambient temperature by step-insertions of reactivity, using for this purpose a special control rod which can be quickly ejected from the core.

Following an initial series of static measurements to determine the basic reactor properties of the test core, a series of non-destructive, self-limiting power excursion tests was performed, which covered a reactor period range down to the point where minor fuel plate damage first occurred - approximately for a 10-msec period test. These tests provided power, temperature and pressure data which established the kinetic behavior of the test core relative to other Spert cores and provided a basis for extrapolation to shorter period tests. Additional kinetic tests in the period region between 10 and 5 msec have been completed to explore the region of limited core damage. Fuel plate damage results included plate distortion, cladding cracking and fuel melting. These exploratory tests have been valuable in revealing unexpected changes in the dependence of pressure, temperature, burst energy, and burst shape parameters on reactor period, although the dependence of peak power on reactor period has not been significantly changed.

On the basis of the results obtained in these kinetic tests, an analysis has been made to determine the nature of the results to be expected for an assumed 2-msec period test in which total core destruction occurs. An evaluation of hazards involved inconducting the 2-msec test, based on pessimistic assumptions regarding fission product release and weather conditions, indicates that with the procedural controls normally exercised in the conduct of any transient test at Spert and the special controls to be in effect during the Destructive Test Series, no significant hazard to personnel or to the general public will be obtained. All nuclear operation is conducted remotely approximately 1/2 mile from the reactor building. Discussion is also given of the supervision and control of personnel during and after each destructive test, and of the plans for re-entry, cleanup and restoration of the facility.

SPERT I DESTRUCTIVE TEST PROGRAM
SAFETY ANALYSIS REPORT

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SPERT I DESTRUCTIVE TEST PROGRAM
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I. INTRODUCTION

A. General

Preparations are underway for the initiation of a series of core destructive tests at the Spert I facility in the fall of 1962. These tests form an integral and important part of the Nuclear Safety Program being carried out at the National Reactor Testing Station by the Phillips Petroleum Company for the Atomic Energy Commission. The program is under the jurisdiction of the Idaho Operations Office, with technical cognizance in Washington under the Nuclear Safety Research and Development Branch of the Division of Reactor Development.

This Safety Analysis Report is intended to provide a summary of the objectives of the test program; a detailed description of the reactor facility, instrumentation and operating procedures pertinent to the destructive tests; the results and analysis of non-destructive static and kinetic experiments already performed, from which predictions of results of the destructive tests can be made for comparison with experiment; and an evaluation of the hazards involved and of the cleanup problem. The report has been prepared at the request of the Atomic Energy Commission[1] and is submitted for approval prior to the commencement of the Destructive Test Series.

B. Spert Reactor Safety Program

The inherent capability of reactors in general to give rise to rapid increases in power has been recognized from the first to offer serious potential hazard to personnel and property.

Although the probability of an accident and its consequences have been largely unknown, cognizance that such hazards were possible has given rise to a necessary, but costly, technological conservatism, in order to provide protection against the hazards implicit in a "maximum credible accident". This conservatism arises in almost all phases of reactor technology, from the original reactor design involving isolation of the plant by containment and distance to the final detailed procedures adopted for operation of the reactor.

In recognization of the need to obtain an understanding of the physical phenomena involved in reactor kinetics and the implications in regard to reactor accidents and consequences, Phillips Petroleum Company was asked by the Atomic Energy Commission in 1954 to undertake a long range reactor safety program. The original instructions at the inception of the Spert program included not only basic non-destructive studies of the importance of various parameters in reactor kinetic behavior that bear on the problem of reactor safety, but also included planned integral-core destructive tests to investigate the consequences of reactor accidents[2,3,4]. The latter aspect of the Spert program has been delayed primarily because additional parametric studies on various core configurations were necessary in order to understand and evaluate more fully the operative reactivity compensating mechanisms, and because Spert I was until recently the only facility available for such tests. The completion of a substantial portion of the investigations which could be performed in Spert I and the construction of the Spert II, III and IV facilities, make it appropriate at this time to carry out a series of tests in which core meltdown and physical disassembly occur.

In previous Spert program discussions, reactor accidents have been categorized into three phases: initiations, response, and consequences. The major effort in the initial Spert work has been on the response phase, since this represents that aspect of reactor safety which is most strongly influenced by the inherent properties of the system. The proposed destructive tests will provide additional information on the response phase in the region of short-period transients, and will also constitute the first extension of the Spert experimental program into the consequences phase of reactor accidents.

C. Borax Destructive Test

The first extensive experimental kinetic studies prior to the Spert program were the Godiva tests for fast systems and, later in 1953 and 1954, the Borax experiments for thermal systems. In the Borax tests, self-limiting power transients were performed in a reactor comprised of MTR-type fuel elements, covering the period range between 100 and 5 msec at boiling temperatures and between 100 and 13 msec at ambient temperatures[5,6,7]. While only minor damage was obtained as a consequence of these tests, the results indicated that larger reactivity insertions would lead to extensive core damage. The Borax program was concluded with a 2.6 msec period, ambient-temperature test, which yielded a maximum power of about 14×10^9 watts, an estimated nuclear burst energy of about 135 Mw-sec and estimated peak pressures of the order of 10,000 psi, and resulted in nearly total destruction of the core and facility. While this test gave information representing a considerable step forward over what had previously been available for evaluating accident excursions, and constitutes the only planned destructive test performed to date, the limited information obtained was largely qualitative as a result of the single-shot aspect of the test and of the available instrumentation coverage.

In addition, the Borax program did not include ambient temperature tests in the reactor period region $13 \gtrsim \tau_0 \gtrsim 2.6$ msec, where pressure and temperature destructive effects are expected to be important. The power, energy, temperature and pressure data obtained during the Borax destructive test were relatively incomplete and uncertain, especially, in regard to the important pressure measurements. Radiation sensitivity and frequency response capabilities of the instrumentation added to the uncertainty. The instrumentation used in the Borax test was inadequate to obtain specific test data about destructive processes in the reactor, and even with the present-day improved instrumentation, it is not expected that the necessary information would be forthcoming from a single test. A series of tests, successively shorter in period and perhaps repetitive, are required to test extrapolability of the reactor response from the longer-period, non-destructive region to the short-period, total destructive region. The possible existence of mechanism thresholds, such as melting points, yield points, burnout points, critical temperatures, etc., requires performance of a series of tests in small steps, in which continuous analysis of results and improvement in instrumentation is made leading to the total destructive test.

The Borax test demonstrated the importance of destructive core tests and the feasibility of conducting such tests with negligible risk to personnel. At a point 0.5 miles from the reactor, where the nearest personnel were located, the instantaneous dose rate immediately after the test reached a peak of approximately 400 mr/hr. In 30 sec the dose rate had reduced to 25 mr/hr and to less than 1 mr/hr after 5 minutes. The test, having been conducted under meteorological control, resulted in the contaminants being swept downwind, with no exposure to personnel from a "fisside cloud".

D. Objectives of the Spert Destructive Test Program

A long range destructive test program would be intended to be a comprehensive investigation of the destructive reactor accident for a given class of reactors. It would entail a series of controlled destructive tests designed to provide information on questions relating to (a) reactor kinetics and shutdown behavior; (b) the magnitude of the pressures generated and their mechanical effect on the reactor environs as related to the general problem of containment; (c) energy partition; (d) the extent of mechanical damage, radiation exposure, fission product release, etc., resulting from a given destructive burst, and (e) identification of the ultimate shutdown mechanism in a severe accident. For the first step in this overall program, an initial test series (with which this report is concerned) will be performed in Spert I to obtain data on the response of the reactor to large reactivity insertions, as distinct from test data bearing primarily on accident consequences exterior to the reactor vessel.

This distinction is important, for example, to the accident analysis problem of the transient pressure developed in a destructive burst. In order to obtain useful pressure data, the experimental conditions which affect the interpretation of the data will be maintained as simple as possible by avoiding, to any significant extent, the additional boundary-condition complexities which would be inherent in tests directed primarily at the containment problem. The initial destructive test series will be directed principally to a study of the reactor response, the pressure source, and the damage effects occurring within the open Spert I reactor vessel.

More specifically, the objectives of the initial tests are concerned with such questions as:

(a) What is the nature of reactor shutdown and of the reactor dynamics in the relatively unexplored regions of limited and total core destruction and, in particular, to what degree are the previous Spert results (power, energy, temperature, etc.) obtained in the non-destructive region extrapolable?

(b) What is the space- and time-dependence of the pressure pulse developed within and without the reactor vessel, the impulse loading given to the reactor vessel and core-support structure, the effect of the pressure pulse in the violent disassembly of the core, and the dependence on the magnitude and rate of rise of the transient pressure on reactor period? In regard to the origin of the pressure rise, to what extent and with what time constants are such mechanisms as steam production, core expansion, radiolytic gas formation or a metal-water gas evolution involved?

(c) Is the energy source in the destructive accident entirely nuclear, or is there a significant contribution from other sources such as a metal-water reaction?

(d) What is the extent of the mechanical damage (missile sizes and speeds), the radiation exposure, and fission product release? What fraction of the core melts down or is otherwise damaged before effective shutdown by core disassembly occurs?

In the initial destructive test series the intent is to obtain full information on items (a) and (b), and to make a modest beginning on points (c) and (d).

In the light of these objectives and the previous Spert work, which has been principally directed to the study of water-moderated, plate-type cores, a small, high-enrichment, plate-type core has been selected for the initial study. The core is comprised of 25 Spert D-type fuel assemblies, arranged in a 5 x 5 array. The fuel assemblies have been designed to be of simple construction and to consist of removable fuel plates, in order to permit easy removal and replacement of damaged fuel plates during the intended series of increasingly-severe destructive transients. The fuel assemblies are described in detail in Section II-B and in Appendix A.

E. Schedule

The destructive test program to be performed in Spert I in 1962, is discussed in greater detail below. This program consists of (a) static measurements, to establish the basic core properties; (b) fiducial, non-destructive kinetic tests, to establish base-point data for kinetic behavior for extrapolation to shorter-period, destructive regions and for comparison with other Spert cores; (c) exploratory kinetic tests, to investigate the region of limited core damage; (d) final preparatory kinetic tests, for checkout of instrumentation, etc., and (e) destructive tests, conducted with meteorological controls, special operating procedures and personnel control appropriate to a violently destructive test.

Portions of the static, fiducial-kinetic and exploratory tests have been completed at the date of writing (June 1, 1962) of the Safety Analysis Report. Assuming that approval for the conduct of the Destructive Test Series is received during the summer of 1962, the series will be initiated about September 1, 1962. Barring unexpected difficulties or the onset of weather which precludes continuation of the series, the destructive series will continue until a violently destructive test occurs. "Violently destructive" is taken to mean that mechanical damage or melting is severe enough to require a period of two months or more for area cleanup and restoration of equipment.

II. BRIEF DESCRIPTION OF REACTOR FACILITY

The following is intended to provide a brief description of the Spert I reactor facility: The building, the reactor proper, reactor control, operational instrumentation, and auxiliary equipment. A more detailed description of the reactor facility is given in Appendix A.

A. Spert I Reactor Site

The Spert I reactor site is located approximately 1/2 mile from the Control Center building and approximately 1/2 mile from the nearest (Spert II) reactor facility. Spert I consists of the reactor building (shown in Fig. 1); an earth-shielded instrumentation bunker which is located adjacent to the reactor building and contains the transient electronic instrumentation used for transmitting instrumentation signals to the Control Center; and a small service building (designated the "terminal building") which is located about 400-ft south of the reactor building and is used to house the water-treatment equipment, air-compressor and other auxiliary equipment. The reactor is operated remotely with the control console located in the Control Center building.

Detailed descriptions of the Spert I site and buildings are given in Appendix A-I.

B. Reactor

Fig. 2 is a cutaway view of the Spert I reactor. The reactor vessel is an open, unpressurized, 10-ft diameter by 16-ft deep, carbon-steel tank. The core is moderated and cooled by light water, with no provision made for forced coolant circulation through the core. (A small electrical stirrer unit is available to aid in obtaining uniform bulk-water temperature conditions in the reactor.) The water level in the reactor vessel is nominally 4.5-ft above the top of the fuel plates.

The destructive test core (designated as the Spert I D-12/25 core) is comprised of 25 (nominally, 12-plate) fuel assemblies, mounted in a 5 x 5 array in a rectangular grid structure. A schematic cross section through the core is shown in Fig. 3. Four symmetrically-placed, gang-operated control rod assemblies, each consisting of a pair of poison blades with aluminum followers provide reactor control. An additional,

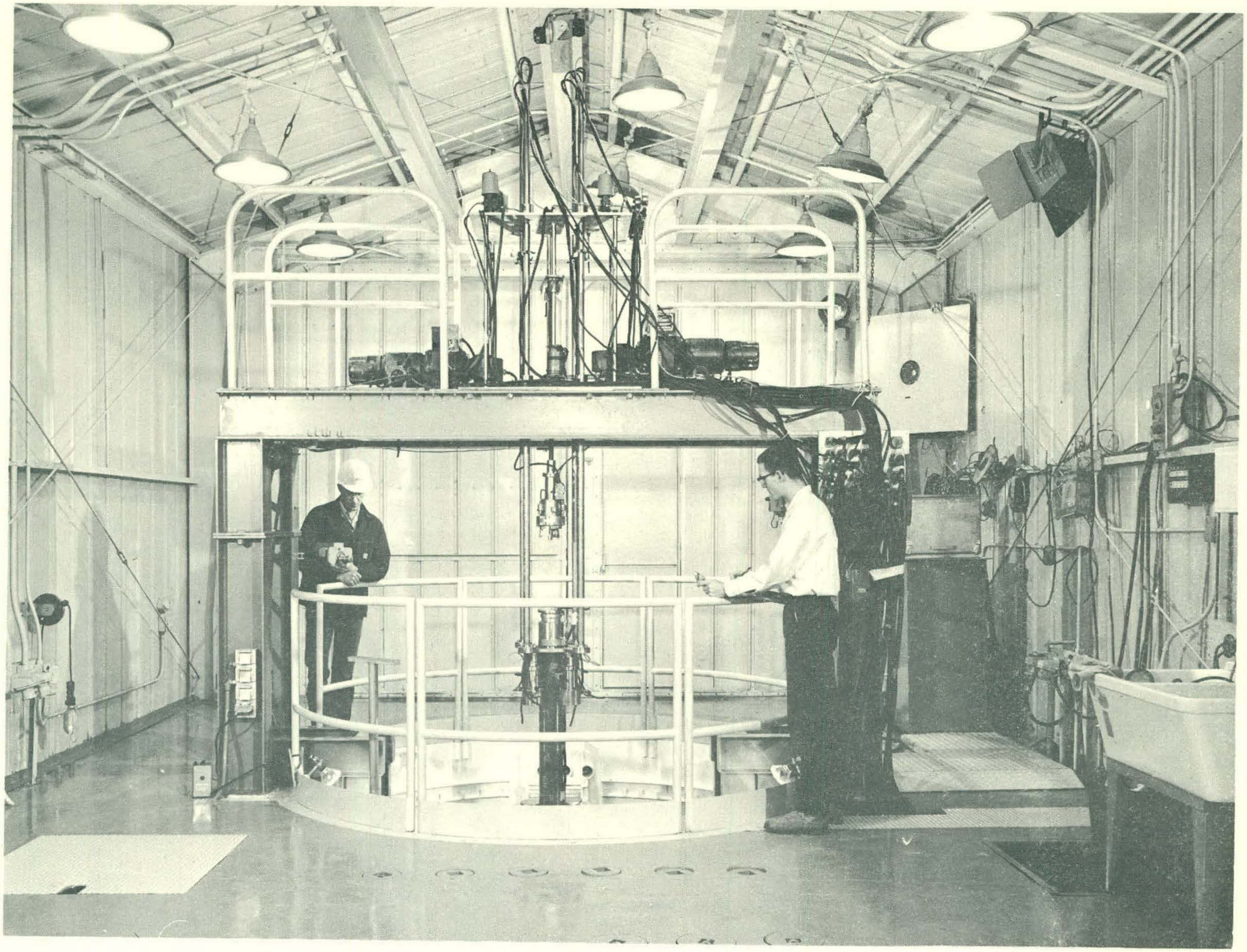


Fig. 1 View of interior of reactor building

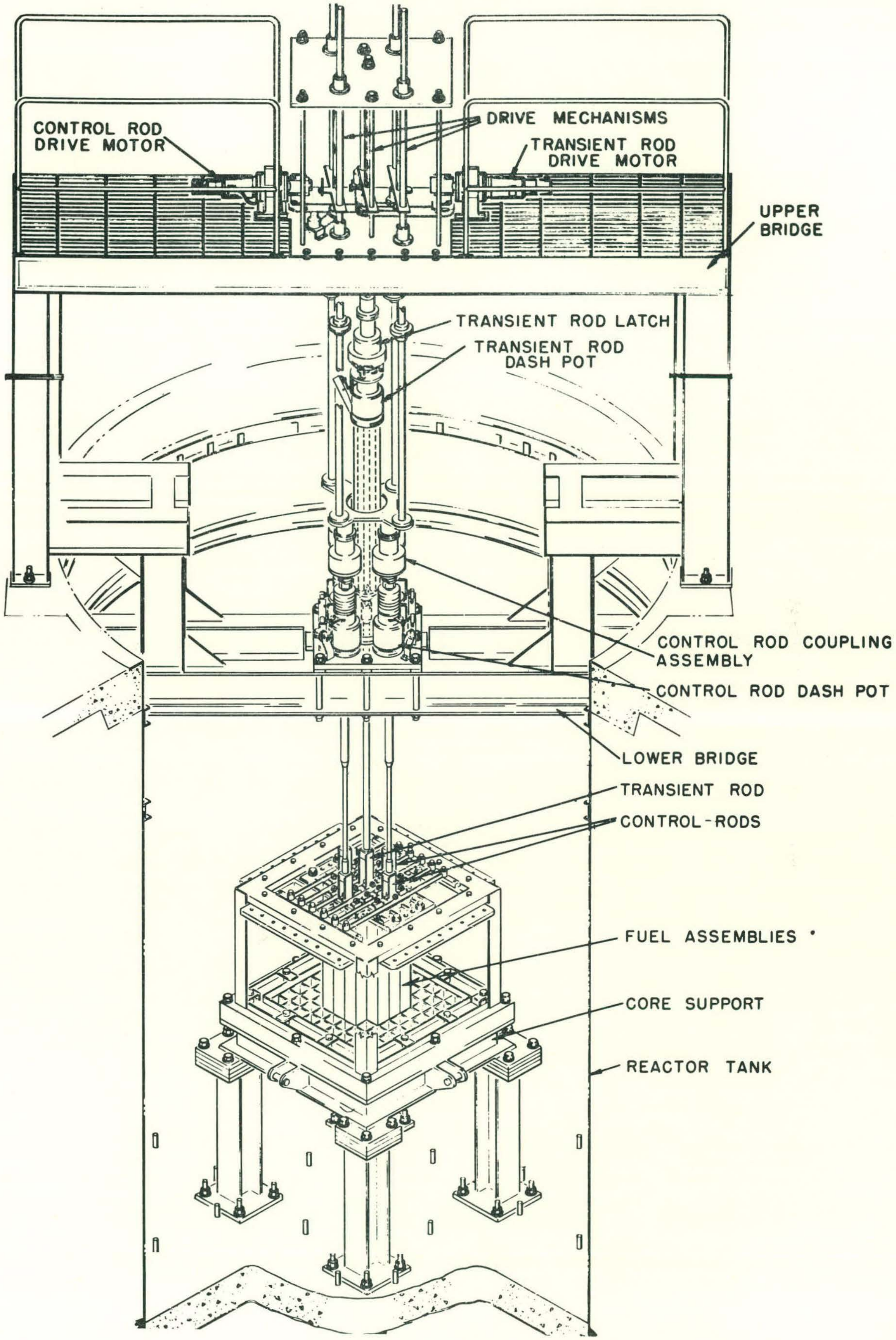


Fig. 2 Cutaway view of reactor

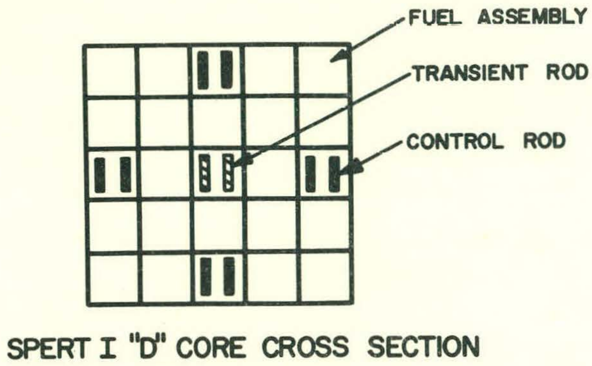


Fig. 3 Cross section through core

centrally-located "transient" rod assembly consisting of two aluminum blades with poison follower blades, is used to initiate experimental reactor transients.

The Spert I D-type fuel assembly is shown in Fig. 4. The fuel assembly consists of a square aluminum box, within which are two grooved aluminum side plates used for supporting removable fuel plates. The end-box section at the bottom of the fuel assembly fits into thc lowcr core support grid structure, and a lifting bail at the top is used for fuel assembly insertion and removal from the core. The standard fuel assembly contains 12 highly enriched U-Al alloy, aluminum clad fuel plates, which can be removed from the fuel assembly box to permit fuel plate inspection, installation of instrumentation, replacement, etc. The four control-rod and one transient-rod fuel assemblies contain only 6 fuel plates; the remaining 6 fuel plate positions are occupied by two control blades and blade-guide tubes. Fuel assembly design characteristics are summarized in Table I; more detailed descriptions of the reactor tank, core support structure, fuel assemblies, etc., are given in Appendix A-II.

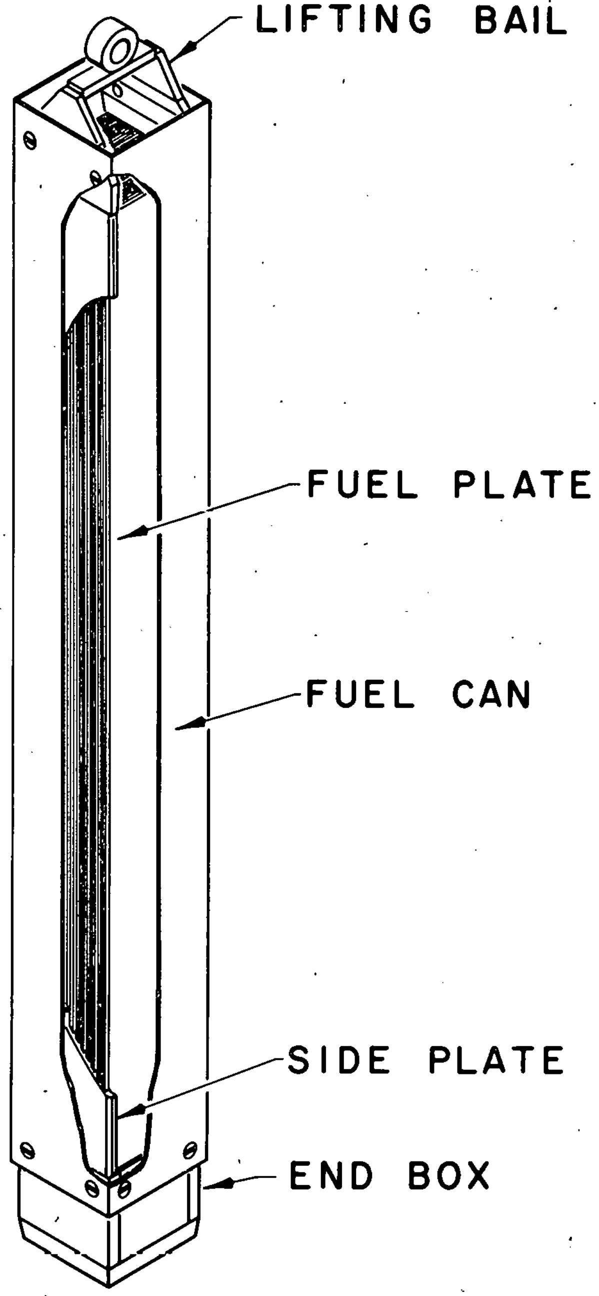


Fig. 4 Spert I D-type fuel assembly

TABLE I

Mechanical Properties of Spert I D-12/25 Reactor

Reactor Type	Open Pool
Moderator-Reflector	H_2O
Vessel Size	10-ft ID x 16-ft High
Vessel Material	Carbon Steel, covered with corrosion-resistant paint.
Number of Fuel Assemblies	25
Standard Fuel Assemblies	20
Modified Fuel Assemblies	5
Approximate Fuel Assembly Size	3 in. x 3 in. x 25 in.
No. of Fuel Plates per Assembly	Standard - 12 Modified - 6
Total Number of Fuel Plates in Core	270
Fuel Plate Thickness	0.060 in.
Meat Thickness	0.020 in.
Clad Thickness	0.020 in.
Active Length of Fuel Plate	24 in.
Coolant Channel Thickness	0.179 in.
Meat	93%-enriched U-Al alloy
U^{235} per Fuel Plate	14 g
Total U^{235} in Core	3.8 kg
Total Core Volume	5.3×10^3 in.^{3}
Moderator Volume	3.2×10^3 in.^{3}
Metal-to-Water Ratio	0.66
Heat Transfer Area	3.4×10^4 in.^{2}
Control Rods	4 double-bladed, gang-operated
Transient Rods	1 double-bladed
Control and Transient Rod Poison Material	Binal* (7 Wt. % Boron-Al alloy)

*Binal - Trademark for the Sintercast Corporation Aluminum-Boron powder-metallurgy processed material

C. Reactor Control

Nuclear operation of the Spert I reactor is carried out by remote control from the Spert I control room in the Control Center building, approximately 1/2 mile away. The basic reactor control system of the Spert I reactor consists of four blade-type control rods and one blade-type transient rod, with their associated drives and electrical controls.

The four control rod units are identical, each consisting of two blades connected at the top by a yoke assembly and shaft to the armature of the coupling magnet. The upper, poison section of the double-bladed control rod is constructed of a 7 wt% boron-aluminum alloy. The lower, follower sections of the control blades are constructed of aluminum. Poison is removed from the core by upward withdrawal of the control rods. The control rods are magnetically coupled to a single electro-mechanical rod-drive system, providing gang-operation of the four control rods. Each coupling magnet assembly is equipped with a short-stroke, air-operated piston to provide additional initial acceleration of the control rod when the control rods are scrammed.

The centrally-located transient rod is a double-bladed rod, consisting of an aluminum upper section and a poison lower section. The upper unpoisoned section is normally retained in the active core region. In preparation for a reactor transient, the transient rod is raised to make the reactor subcritical and the control rods are positioned in accordance with the desired reactivity insertion; the reactor transient is then initiated by release of the transient rod.

The transient rod is moved by its own electro-mechanical rod drive system. Since the transient rod is an inverted control rod, i.e., poison is added to the core by rod withdrawal, a positive-action, air-operated, mechanical-latch mechanism is utilized instead of a magnet to couple the transient rod to the drive unit. The latch mechanism can only be actuated by a key switch on the reactor console. A short-stroke, pressurized air piston, similar to that used on the control rod, provides additional initial acceleration of the transient rod.

Two pistol-grip switches on the reactor console control the movement of the transient rod and control rods. The rod positions are continuously monitored to within 0.01-in. by two digital indicators located above the respective control switches.

A multi-section timer unit with associated relays is used to initiate selected experimental functions in a given sequence during a reactor transient, i.e., the ejection of the transient rod, 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. The action of the sequence timer is, by means of limit switches and interlocks, compatible with the control system functions. There are no automatic power level or period scram circuits; reactor shutdown can be initiated by the sequence timer or by manual scram action.

A detailed description of the control system and a discussion of the design philosophy of the system is given in Appendix A-III.

D. Operational Instrumentation

The Spert I operational instrumentation includes the neutron instrumentation, reactor bulk water temperature instrument, reactor water level instrument, and radiation-detection equipment. Transient instrumentation is discussed below in Section III.

Operational neutron instrumentation includes four B^{10}-lined pulse chambers with amplifiers and counters, a B^{10}-lined gamma-compensated, ion chamber connected through a linear electrometer to a six-decade linear power recorder, and a B^{10}-lined, uncompensated, ion chamber connected through a log electrometer to a six-decade log power recorder. The chambers and electronic amplifiers are located in the reactor building and instrument bunker, respectively, and the counters and recorders are located in the Spert I Control Room.

The reactor bulk water temperature is measured by a thermopile consisting of four thermocouples connected in series and positioned near the tank wall at the approximate centerline of the reactor core. Leads from the thermopile extend toa constant temperature reference junction from which the signal is transmitted without amplification to a temperature recorder at the Control Center.

The water level in the reactor vessel is measured by a float-type apparatus coupled to a selsyn transmitting system driving a digital counter located at the control console. Water level is read to ± 0.01 ft.

The gamma radiation levels directly over the reactor vessel and at other points in the reactor area are measured by gamma-sensitive chambers. Signals from these detectors are transmitted to indicators in the Spert I Control Room and a recorder in the Health Physics Office at the Control Center. A warning bell at the reactor building is actuated whenever the gamma radiation level measured by any of the chambers exceeds a predetermined set point. Air in the reactor building is continually sampled and monintored for gaseous or particulate radioactive material by means of a constant-air-monitor instrument housed near the reactor building.

A detailed discussion of the operational instrumentation is found in Appendix A-IV.

E. Auxiliary Equipment

The auxiliary equipment for the Spert I reactor facility includes a water-treatment system, an air compressor, a one-ton overhead crane, an emergency soluble-poison injection system, and other miscellaneous items. Further discussion of the auxiliary equipment is given in Appendix A-V.

III. EXPERIMENTAL PROGRAM

A. Reactor Test Series

The destructive test program has for convenience been divided into four (roughly sequential) series of reactor tests, designated the Static, Fiducial-Transient, Exploratory-Transient, and Destructive-Transient Series. (At the time of writing of this report, which has been requested by the AEC prior to approval to conduct the Destructive-Transient Series, most of the first two series and the initial portion of the third series have been completed.)

The initial static tests are those required to determine the basic reactor characteristics of the test core. This series includes (a) the approach to a minimum critical loading and establishment of the 25-assembly operational core loading; (b) control rod and transient rod worth measurements and from these, determination of the excess and shutdown reactivities of the core; (c) flux distribution measurements; (d) void coefficient measurements for uniform and local distributions of voids; (e) isothermal temperature coefficient measurement; (f) static measurement of the reduced prompt neutron lifetime, ℓ/β_{eff}; and (g) power calibration of the nuclear instrumentation.

The Fiducial-Transient tests are routine (step-initiated) self-limiting power excursions, covering the reactor period range down to the point where the threshold for core damage occurs. For this core the threshold for damage occurs for reactor periods of about 10 msec. These tests establish base point data on the kinetic behavior of the core, for comparison with the behavior observed in other Spert cores and for providing a basis for extrapolation to shorter period tests.

The Exploratory-Transient tests are those in which the threshold of damage is crossed and the region of limited core damage is explored. (Previous Spert tests performed in the non-destructive region were too far below the damage threshold to provide a sound basis for realistic prediction of the results of highly destructive tests.) The exploratory tests are intended to provide data on reactor power burst shapes, transient pressures, fuel plate temperatures, and the nature of fuel plate damage for tests in which some fuel plate melting is observed at the core hot spots. The extent of core damage is ascertained from a detailed examination of the core following each test. The tests are limited to reactivity insertions for which there is reasonable evidence that only local core damage (limited to about 10% of the fuel plates) will result, and do not include tests in which structural damage exterior to the fuel assemblies occurs. The reactor period range for the Exploratory-Transient Series is $10 \gtrsim \tau_0 \gtrsim 4$ msec.

The Destructive-Transient Series tests will be performed at successively shorter periods, resulting in increasing damage to the core and other reactor components and necessitating replacement of fuel assemblies and instrumentation after each test. All tests in this series will be conducted with the meteorological controls, special operating procedures and personnel controls appropriate to a violent destructive test. Barring unexpected difficulties or the onset of weather conditions precluding the continuation of the destructive tests, the series will continue until a test occurs which results in sufficient damage to require about two months or more for area cleanup and reactivation of the facility.

B. Test Procedure - General

In carrying out the objectives of the destructive test series, and of the Spert experimental program in general, under conditions which would normally be considered unsafe for most reactor facilities, administrative control must be relied upon to minimize the possibility of unplanned nuclear incidents, ensure the safety of Spert personnel and the NRTS, and to eliminate hazard to the public. The nature of the experimental program also requires that, in general, the various reactor control systems not be provided with automatic safety circuits. The control system of the Spert I reactor does, however, contain a number of interlocks, both electrical and mechanical, in order to reduce the probability of unplanned reactor excursions and to prevent the carrying out of procedures which could lead to unanticpated situations or unsafe operating conditions.

To help ensure continuous administrative control of the reactor facility during all phases of nuclear and non-nuclear reactor operation, a formal testing procedure is followed. This procedure is based on the principle that no nuclear operation is permitted with any person within approximately one-third mile of the reactor*. Details of the testing procedure are summarized in Appendix B. The application of safe operating practices and the cognizance and prevention of potentially unsafe acts and situations is recognized to be the individual responsibility of all Spert personnel.

C. Transient Measurements and Instrumentation

1. General

The necessity for recording the data at the Control Center one-half mile from the reactor, arises from the possibility of destruction of instrumentation located in the vicinity of the reactor. It is also desirable to initiate the processing of data immediately after the conclusion of a test, without waiting for re-entry of the reactor area. The electrical signals from the transducers in the reactor are transmitted to the earth-shielded instrumentation bunker adjacent to the reactor facility, and thence, by means of special driver amplifiers, transmitted 3000 ft. to high-speed oscillograph recorders in the Control Center.

For the Fiducial-Transient and Exploratory-Transient Series, 76 data collecting channels are available for use during each transient test. For the Destructive-Transient Series, additional data channels will be made available for a total of up to 114 channels.

The instrumentation system requirements necessary to obtain good data on reactor power, fuel plate temperatures, pressures, flow, strain, and other quantities during a reactor transient are determined by the environmental capabilities, dynamic response and lower limit of sensitivity of the transducers. Physical size limitations are imposed on transducers placed within or in the vicinity of the core. In addition, all transducers must have an acceptably low radiation sensitivity and be capable of operating submerged in water, where the temperature may vary

*It is to be noted that while the nearest point on the perimeter fence is a minimum of 1/3 mile from the reactor, Spert personnel are withdrawn 1/2 mile during all nuclear operations.

from ambient to boiling. The DC accuracy of the various instrumentation systems can generally be established to within 1 or 2%. The requirements involved in following (to within 1 or 2%) changes associated with a 1-msec period transient, however, demand a frequency response of up to 20 kc. (This is in accordance with the rule of thumb that the bandwidth in cps required for following an exponentially rising signal to 1% for greater than 2 decades of rise is $100/2\pi$ times the reciprocal period of the exponential rise[8].)

The very limited number of Destructive-Transient Series tests implied by the destructive consequences of the tests require that there be a high probability of obtaining the desired data from each test. The precautions taken to avoid losing data during the destructive tests are as follows:

(a) Protection of Instruments and Instrumentation Leads
Cameras, periscope equipment, neutron detectors and pressure transducers outside the core have been protected against blast and missile damage. Thermocouples, strain gages, accelerometers, and pressure transducers located inside the core cannot be similarly protected because of space limitiation and replacement of these instruments will be made whenever necessary. Protection of transducer leads and ion chamber leads is provided.

(b) Multiple Instrumentation
There is multiple instrumentation for the different kinds of measurements, to permit measurements at various locations over a wide dynamic range. There is a redundancy of instrumentation for added protection. The several kinds of measurements made and the instruments used are described in detail in the following section. Amplified output signals from selected transducers are also recorded over two different ranges, differing by a factor of ten, to provide complete data monitoring during the early portion of the transient.

(c) Different Instrumentation
This includes, for example, use of ion chambers, and foil

activation techniques to measure power and energy, in addition to the energy information available from temperature, pressure and acousitc measurements.

(d) Parallel Instrumentation

Use is also made of multiple thermocouple junctions which are connected in parallel to the same thermocouple leads. This would provide added protection against loss of temperature information at a given point on a fuel plate in the event that a particular junction becomes disconnected from the fuel plate surface.

2. Reactor Power Measurements

An accurate measurement of the transient power level of the reactor is of basic importance to the study of reactor kinetic behavior because of the close relationship of the power to the instantaneous reactivity of the system and to the physical mechanisms of the reactivity feedback.

In a reactor transient, the determination of the initial asymptotic reactor period of the exponentially rising power, the peak power and the shape of the burst in the region of peak power are of primary interest. The breakaway from the exponential rise caused by the feedback effects normally occurs about a decade below peak power so that determination of the initial asymptotic reactor period is dependent on power level measurements made in the power range at least one decade below peak power. Measurement of the reactor power over a dynamic range of about 5 decades is obtained by several boron-lined, linear ionization chambers positioned within the reactor tank and in the ion chamber instrument tubes outside the reactor tank. Placement of these chambers is shown in Fig. 5. Each linear power recording circuit covers a power range of about 2.6 decades. A single logarithmic power recording circuit, covering more than 5 decades of power, and a linear circuit, covering the power range up to about 2 decades above the expected peak power level of a transient, serve primarily as backup instruments. The relative dynamic range that is nominally covered by the ion chambers during a transient test is shown schematically in Fig. 6. The dynamic range is adjusted upward or downward on an absolute power scale in accordance with the peak power level expected in a particular power excursion. Calibration of the ion chamber signals to

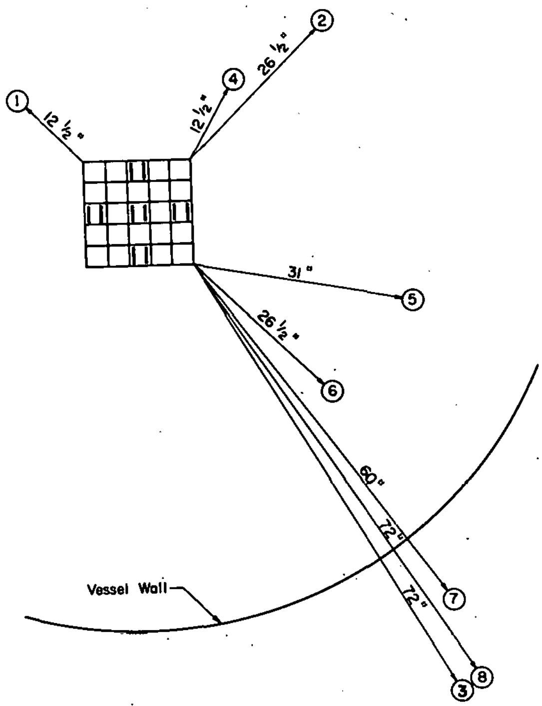


Fig. 5 Location of ion chambers in the reactor tank.

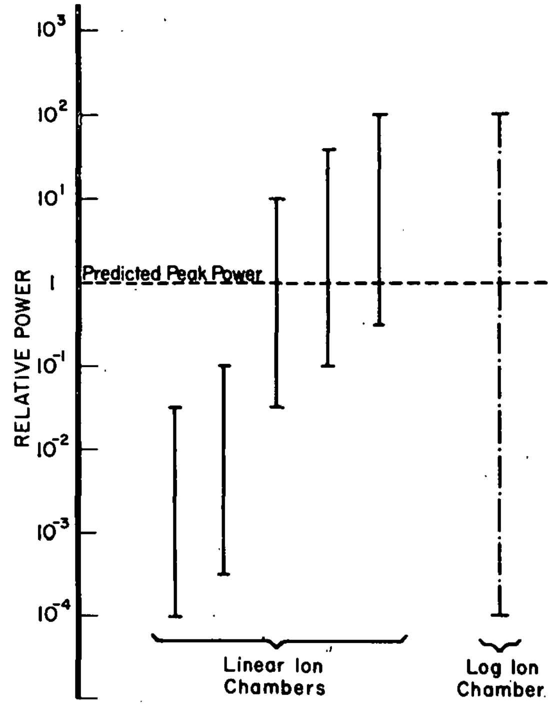


Fig. 6 Dynamic range of reactor power covered by five linear and one log ion chambers during a transient test.

the absolute power level is made on the basis of a calorimetric measurement similar to that described in Section 3 below.

The problem of determining the instantaneous power level for the situation where large, rapid fluctuations of power occur involves the requirement to discriminate against delayed gamma emission from fission products. Analysis indicates that the delayed gamma contribution to gamma sensitive chambers is of significance on the back side of a short period burst, approximately two or three decades below peak power. This contribution is reduced by the use of gamma compensated chambers, by the judicious placement of chambers, and by shielding of the chambers. Gamma compensation of the ion chambers is effective in obtaining about a 30 db reduction of the direct gamma signal; greater reduction can be obtained, but is usually not reliable in operation. Chamber locations in the vicinity of the core provide a relatively greater neutron-to-gamma signal ratio, since neutron attenuation in water is greater than gamma attenuation.

The leakage neutron flux measured by the ion chambers provides a measure of the total power of the reactor. The overall chamber sensitivity is therefore dependent on the degree of neutron leakage and can be expected to be affected by changes in power distribution or the onset of boiling during a reactor transient. The problem does not appear to be of consequence until about two "periods" after peak power, at which time boiling occurs in a significant fraction of the moderator volume in the core. To improve the determination of the power burst shape on the back side of the burst, use will be made of miniature ion chambers which can be placed inside the core.

Blast protection of ion chambers located within the reactor tank is provided by heavy-wall aluminum containers capable of withstanding sustained pressure loads of up to 8000 psi. Signals from the ion chambers are transmitted by coaxial cables protected by heavy-wall aluminum pipe. Amplification of the chamber signal is accomplished by a linear dc to 10 kc amplifier, with range settings to allow nearly full use of the six-decade dynamic range of the ion chamber. The amplifier output of 20 volts full scale is attenuated by factors of 1, 5 and 20, and the three proportional signals are then transmitted to the oscillograph recorders. In this way, three read-out oscillograph power traces are obtained from each chamber, permitting a more precise determination of power level over a range of about 2.6 decades. Power measurements above or below this range are obtained from other ion chambers, whose signals overlap the given range by about 1/2 to 1 full decade for accurate intercalibration of the ion chambers.

3. Determination of Energy

Time integration of the reactor power yields the nuclear energy released as a function of time during the power excursion. Computer integration of the digitized power data is performed, using approximately 20 time intervals per reactor period.

An integral measurement of the total nuclear energy release in a burst can be obtained by the activation of foils or flux wires. Comparison of the normalized foil activation data with the integrated power data obtained from successive tests in the destructive series can be used to indicate any changes which might occur in the power calibration of the ion chambers.

An additional integral measurement of the total burst energy is obtained by measuring the bulk water temperature rise in the reactor tank following a reactor transient. For this measurement, the reactor vessel and its contents are considered to constitute a calorimeter, whose heat capacity (due mostly to water) can be easily calculated. Immediately after a power excursion, a stirrer is used to bring the entire system to a uniform temperature; this is indicated by a distributed set of twelve thermocouples, which can be used to determine temperature rise to approximately $\pm$ 0.01°C. For a 20 Mw-sec energy burst, the calculated uniform bulk water temperature rise is approximately 0.2°C, a temperature change which can be measured to about 5%. The calorimetric technique provides a measure of the total heat generated during the burst from fission or other energy sources. Comparison of the calorimetric measurement of total energy with the measurement of total nuclear energy given by the integrated power or foil activation measurements provides a measure of non-nuclear energy release during a transient. An appreciable metal-water reaction occurring during a destructive power excursion could, in principle, be detected in this way, provided that little or no water is lost from the reactor tank.

4. Water Pressure Measurements

The large transient pressure that can be developed during a short period power excursion from rapid steam formation, plate expansion or other sources constitutes a principal destructive mechanism in reactor accidents, and its measurement during the destructive tests is of prime importance. The measurement and analysis of transient pressure, in general, is made complex as a result of the space-time dependence of the pressure source and of the propagation of the pressure pulse outward from the core. Tensile wave reflection, pressure multiplication at boundary interfaces, and possible shock wave buildup resulting from the change from good to poor acoustical properties of water all tend to make the analysis of pressure effects complicated.

The technique of transient pressure measurements in radiation fields has developed markedly in the past few years, permitting an improved effort to obtain more comprehensive pressure measurements for the Destructive Test Series. Approximately twenty pressure transducers will be used during the destructive tests to obtain data pertinent to (a) the time and directional dependence of the propagated pressure pulse, (b) nature of the pressure generating mechanisms, (c) energy stored in the pressure pulse, and (d) the nature of the pressure pulse and boundary effects as related to the containment problem. Detailed experimental information on these questions require the use of very sensitive pressure transducers. Additional, less sensitive devices will be used primarily for "back-up" (protective) purposes.

Transient pressure measurements are made using commercial, strain-gage, diaphragm transducers mounted in protective steel covers, with only the diaphragm exposed to the water. These transducers have pressure resolutions of approximately 5% of full scale. Care is taken in regard to details of water-loaded frequency response, resonant frequency, radiation sensitivity, and proximity to boundaries. Prior to use, all transducers are checked with respect to sensitivity and linearity. Based on radiation tests conducted in the TRIGA reactor, the sensitivity of individual transducers and associated signal cables to neutron and gamma radiation is determined. The results of these tests permit selection of the least radiation-sensitive transducers for placement nearest to the core. Near-field transducers for the destructive tests have a response to radiation intensities of the magnitude expected of no more than a few percent of full scale reading. Transducers with larger indicated radiation sensitivities will be placed further away from the core. Special mounting brackets are used to position the transducers in the three-dimensional array shown in Fig. 7. Transducer positions are given in Table II.

For short period transients, "back-up" transducers, capable of measuring pressures in excess of 10,000 psi, are employed even when only moderate pressures are predicted. This is to help ensure that data on unexpected, large, threshold-effect pressure surges are not missed.

5. Fuel Plate Surface Temperature Measurements

Fuel plate surface temperature measurements are of importance in providing data which can be correlated with power and pressure data in analyzing reactor shutdown behavior. Based on the temperature data, calculations can be made of the transient temperature distribution in meat, cladding and moderator to determine the energy partition, heat

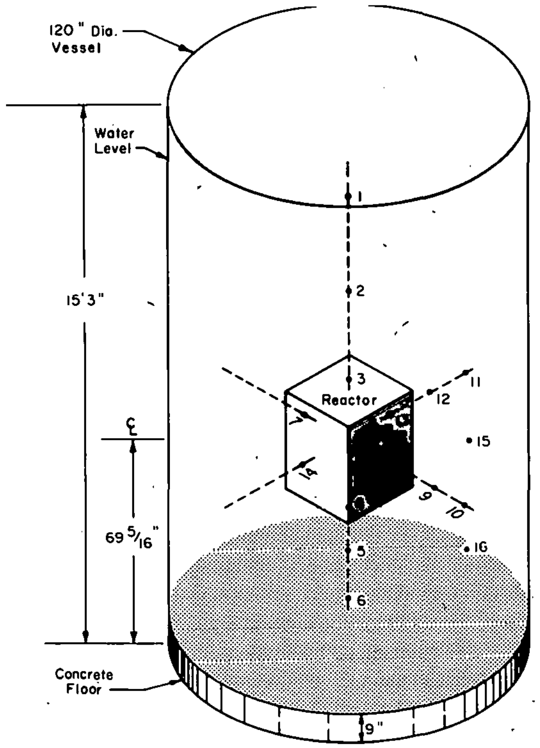


Fig. 7 Location of pressure transducers in the reactor tank

TABLE II

Pressure Transducer Instrumentation

Transducer Number	Position X (inches)	Y	Z	Pressure Ranges (psi)
1	3	2	$58\frac{1}{2}$	100
2	3	2	40	1000
3	3	3	$19\frac{1}{2}$	100; 300; 3000
4	3	3	$-19\frac{1}{2}$	100; 3000
5	3	3	-40	20,000
6 (floor)	3	3	-52	300; 1000
7	$-10\frac{1}{2}$	0	0	10,000
8	$10\frac{1}{2}$	0	0	100; 300; 3000
9	$25\frac{1}{2}$	0	0	1000
10 (wall)	55	0	0	300; 20,000
11 (wall)	0	52	0	300
12	0	$25\frac{1}{2}$	0	10,000
13	0	$10\frac{1}{2}$	0	100; 300; 3000
14	0	$-10\frac{1}{2}$	0	1000
15	20	20	0	1000
16	20	20	-40	1000

transfer rate to the water, plate expansion, etc., as functions of time during the transient. The onset of boiling at a particular point in the core is also indicated by the shape of the temperature curve in the region of peak power [9, 10]. Temperature data are also useful in helping to predict the pressure generation and extent of melting to be expected in a destructive test.

The transducers used in this measurement are thermocouples comprised of 0.005-in. diameter, chromel and alumel wires attached to the aluminum fuel plate surface. A photograph of a typical junction is shown in Fig. 8. Each chromel or alumel leg of the thermocouple consists of three flattened contact points, which are spot welded to the aluminum fuel plate surface to form three parallel, chromel-aluminum-alumel junctions. The junction wires are flattened to approximately 0.0005-in. thick wafers, to increase the thermocouple frequency response. The multiple junction measures to a first approximation the average temperature of the three junctions. This temperature is representative of the temperature of a somewhat larger area than that of a single junction and so reduces the temperature perturbations resulting from fuel inhomogeneities and localized boiling. The multiple junction feature also reduces the probability of loss of information in the event of thermocouple junction rupture. The estimated frequency response of a typical thermocouple is such that a 2-msec period, exponential temperature rise can be followed with less than 5% lag. The magnitude of the temperature rise can be measured with an error of less than 1°C.

The distribution of thermocouples in the core is intended to provide an experimental three-dimensional map of the temperature distribution in one quadrant of the core, which should be representative of the temperature distribution throughout the entire core, as indicated by the symmetrical neutron flux distribution measurements (see Section IV). As shown in Fig. 9, several thermocouples are attached at various vertical fuel plate positions to measure the axial temperature distribution along a given fuel plate.

6. Flow Measurements

Flow measurements to be made during the destructive tests are those associated with the expulsion of water from the fuel assemblies in

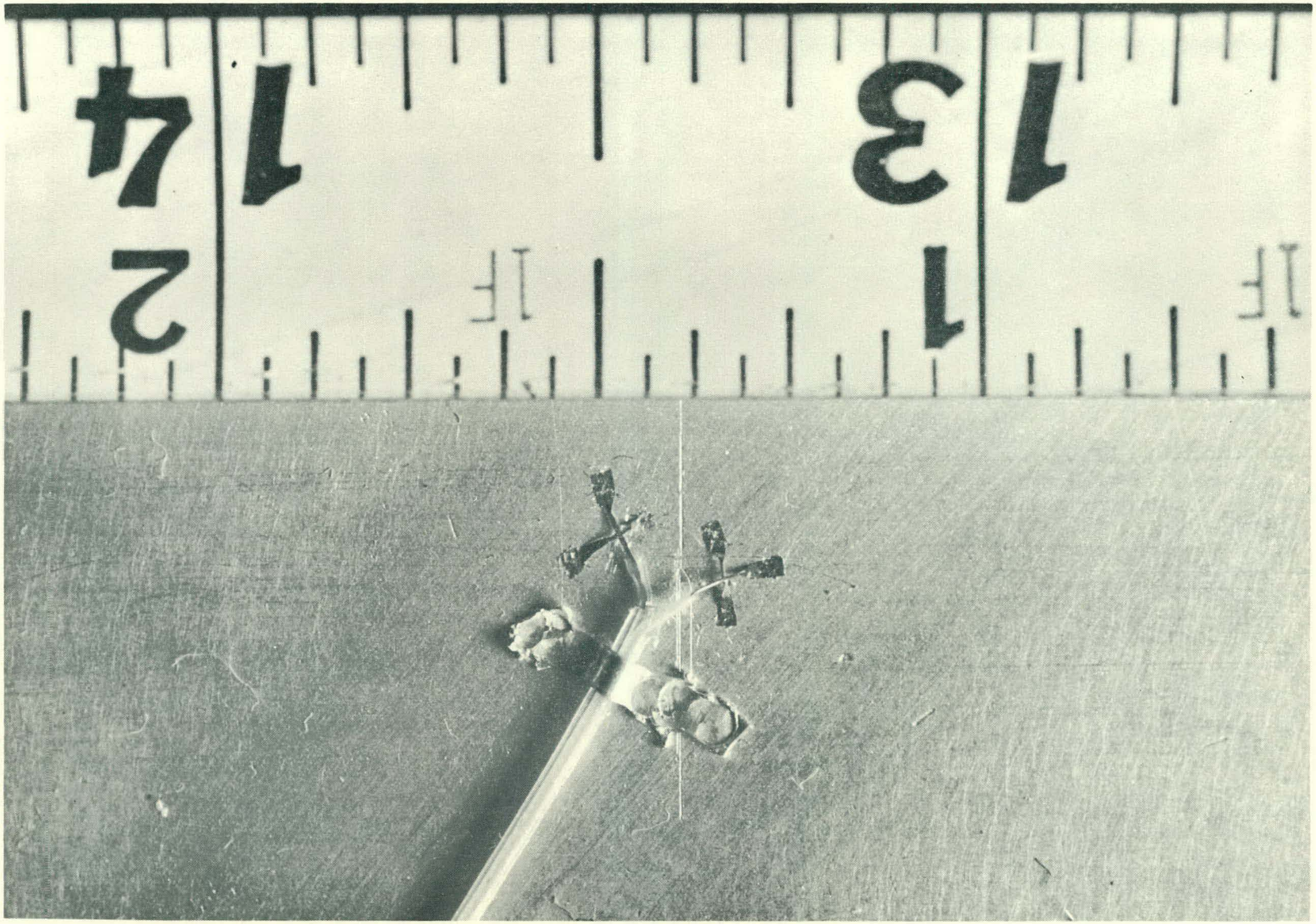


Fig. 8 Detail of typical fuel plate thermocouple junctions connected in parallel.

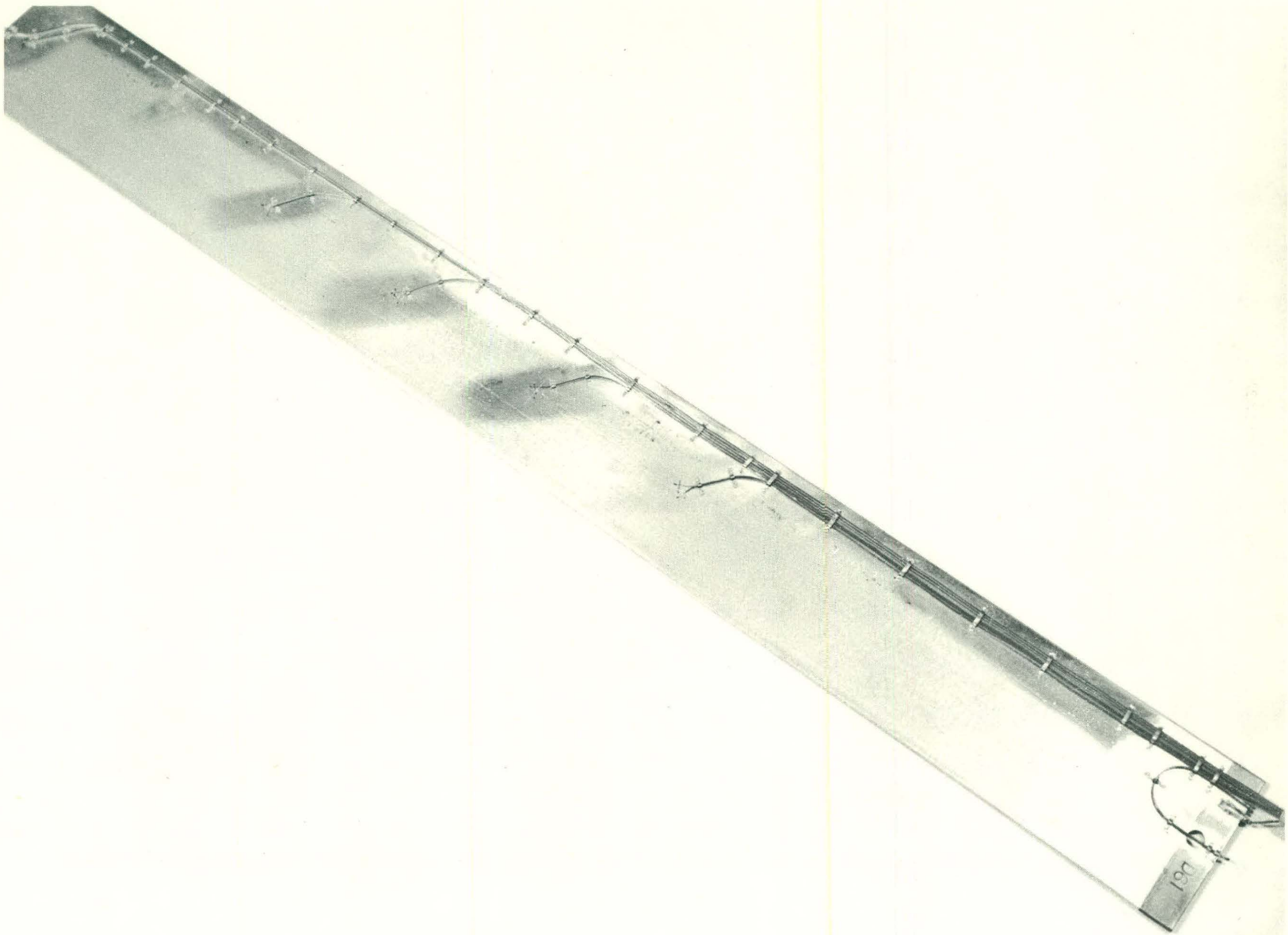


Fig. 9 Thermocouple junctions along length of fuel plate.

the core during self-limiting power excursions. The expulsion of water is believed to be intimately related in under-moderated cores to the shutdown mechanisms of plate expansion, water expansion, and water boiling. During the initial part of a power excursion, fuel plate heating and consequent expansion results in expulsion of the water from the channel between fuel plates. With subsequent transfer of heat to the water, thermal expansion of the water provides an additional contribution to the transient flow of water from the channels. Finally, with the commencement of boiling, large steam voids are formed, which result in a rapid ejection of the water. Measurement of the total water flow from the channels as a function of time during an excursion provides data relating to the time dependence of the negative reactivity effected by loss of moderator from the core, which can be compared to the compensated reactivity determined from the power data.

The expulsion of water from a fuel assembly will be measured by flow meters placed in the fuel assembly end boxes. The flow meters to be used should be able to cover a wide range of flow rates, offer negligible flow resistance, have fast response characteristics, and comparatively insensitive to thermal or radiation effects. No commercial instruments are presently available to fulfill these rigorous requirements and a developmental effort is currently underway to provide reasonably satisfactory devices. The flow meter instrument considered for the destructive tests employs a small vane mounted in the fuel assembly end box. Drag force, which is indicative of flow, is then measured by means of strain gages attached to the beam supporting the vane.

7. Strain Measurements

Pressures and large thermal gradients developed during a destructive test are expected to cause strain, plastic deformation and failure of core materials. It is expected that reactor strain instrumentation can be used to provide strain data which can be interpreted for indications of yield and fracturing before these processes actually take place. For this reason, a considerable effort is being made to develop the technique of transient strain measurements in reactors.

Strain measurements during the destructive tests will be obtained by means of conventional strain gages, which involve the measurement of change of electrical resistance of strained wires attached to the given component under study. The developmental effort is concerned with such problems as those of radiation sensitivity, thermal gradients and attachment techniques, all of which tend to affect the interpretation of data. For the destructive series, it is planned to make strain measurements on selected fuel assembly boxes, on various components of the core support structure, and on the reactor vessel wall.

8. Acceleration Measurements

Motion of reactor components during power excursions can be measured by the use of accelerometers. Interest in making this measurement stems from the expectation that permanent dislocation of various structural parts will occur during a violent destructive test, such as was observed in the Borax test. For a series of increasingly severe, limited damage tests, acceleration data can provide useful correlative and predictive information. From acceleration data, calculations can be made of the speed, displacement and of the forces acting on the instrumented component, and can be used to indicate the time of initiation of dislocation.

The basic accelerometer consists of a spring-held mass, mounted in a container filled with a damping fluid. Displacement of the mass due to acceleration is sensed by a strain gage and the data are interpreted directly in acceleration units. An accelerometer is required for each direction of motion anticipated.

Accelerometers are mounted on fuel cans, on core support structure members, and on the fuel assembly "hold-down" bars at the top of the core. In addition, it is planned to mount an accelerometer on the centrally-located transient rod to determine if there is any tendency of the transient rod to be ejected from the core as a possible result of large pressures developed during a destructive burst.

9. Air Pressure Measurements

A certain fraction of the total energy in a destructive burst is available in the form of mechanical energy in the transient pressure pulse which is developed in the reactor, a portion of which is transmitted across the water-air interface into the air. Based on the success obtained in underwater explosives research in interpreting water and air pressure measurements, it is expected that air pressure measurements in conjunction with water pressure measurements made during a destructive test, can be used to provide some measure of the total burst energy. Peak pressure, rise time of the pressure pulse, the pressure impulse (given by the time integral of the ratio of pressure to acoustic impedance), and the mechanical energy contained in the pressure pulse (given by the time integral of the ratio of the square of the pressure to the acoustic impedance) are important quantities pertinent to the nature and extent of damage, which the air and water pressure data can provide.

Microphones to be used for air pressure measurements have been selected on the basis of good frequency response and high-intensity measuring capabilities. It is planned to use at least two microphones for each of the destructive transients. One of these will be located about 30 feet directly above the reactor and the other about 30 feet horizontally from the lip of the reactor tank.

10. Radiological Measurements

The Destructive Test Series will offer an opportunity to evaluate the radiological hazards associated with a destructive excursion on this type of nuclear reactor. Measurements on a limited scale were commenced during the Exploratory-Transient Series and will be continued and expanded throughout the Destructive-Transient Series. The measurements are intended to show not only the immediate radiological consequences of a destructive nuclear excursion, but also the delayed consequences associated with fallout and contamination. Equipment and techniques to be employed in making the measurements will be standard and no development problems are anticipated.

At present, the radiological measurements program is still in a formulative stage. However, the following review will reflect current planning of this important aspect of destructive testing.

Integrated neutron and gamma doses for a test will be measured at various distances from the reactor building and correlated with the total nuclear energy release of the burst. Since the integrated dose is to include only the dose from the power burst and not from subsequent delayed radiation, it is important to effect retrieval of films and activation foils. The dosage measurements will be made from points close to the core to positions 1/2 mile away, at the Control Center and at the Spert II facility. Those measurements near the Control Center will be augmented by use of direct-reading instruments to provide direct monitoring of dosage to personnel at the Control Center.

The dose rate as a function of time during the excursion will be measured at a few selected locations near the core, using a high speed data-collecting system. The time variation of the dose rates at other locations will be inferred from the measured integrated dosages.

The gamma dose in the vicinity of the reactor facility will be monitored by the re-entry teams, as described in Section VII. Up to the time that portions of the core are thrown upward out of the reactor tank, the gamma dose will be confined by the earth shielding to the immediate vicinity of the reactor building.

Following a violent destructive burst, a contamination map of the reactor will be obtained by the re-entry teams, who will measure the dose rate as a function of time at various grid points. The gamma activity at a location following a violent excursion will consist of the direct dose from fragments in and near the reactor building, and from any fallout of particulate matter.

During the Destructive-Transient Series, samples of the reactor water will be taken and analyzed for fisside content. The measurements taken will include gross activity and iodine assays. These assays, in conjunction with a visual inspection of the fuel plates, will be used to ascertain the magnitude of the fisside release from damaged fuel plates.

The path of any contamination cloud from a given excursion will be followed with the aid of a smoke generator to indicate wind direction. Mobile (including air-borne) crews will be available to follow the cloud and obtain air samples. The samples collected will include filter samples for particulate activity and carbon collectors for gaseous activity, including the halogens.

11. Metallurgical Examination of Reactor Components

The types of damage anticipated to occur in the reactor during the Destructive-Transient Series will be those associated with high temperatures, thermal shocks to the fuel plates and pressure damage to the core structural components. Most of the present planning for metallurgical examination is based upon a study of the response of 6061 aluminum to various thermal shocks up to and including actual melting of the fuel plates. Examination of other reactor components will be based upon analysis of the methods of failure of the individual components.

Visual inspection of the core is performed after each transient to assure that damaged fuel plates and cans are replaced for subsequent excursions. A complete photographic history of core damage will be made by photographing damaged fuel plates and cans as they are removed from the core.

Plate examination presently consists of hardness determinations, bend tests, chemical analysis, and photomicrographs. Additional testing procedures will be employed as required to maintain a complete history of the core materials throughout the Destructive-Transient Series.

Metallurgical examination of damaged fuel plates is expected to yield information on the following quantities:

(a) Maximum temperatures of fuel and cladding
(b) Temperature profile of plate
(c) Diffusion of fuel into the cladding
(d) Cladding separation from the fuel
(e) Indication of grain growth
(f) Hot shorting of cladding
(g) Chemical composition of globules of melted fuel plate
(h) Indication of alloying fuel and cladding
(i) Brittleness or softness of plates
(j) Extent of melting of fuel and cladding

12. Photography

Photographic documentation of the transient tests was initiated during the Exploratory-Transient Series and will be continued with increased scope during the destructive series. Photographic coverage of each destructive transient is divided into two phases, the first phase comprising motion pictures taken during the transient test while the second phase will cover the re-entry and cleanup operations.

Photographic coverage during a destructive burst will consist of high speed motion pictures of the core, reactor, and the general area. The cameras will be controlled by action of the sequence timer with the power cables leading to the cameras protected from physical damage. Blast and missile protection of cameras is provided by shield boxes enclosing the cameras. Radiation exposure of the film in the cameras will be minimized by the shielding of the camera boxes and by retrieving the cameras located in the reactor building as soon after the end of a destructive test as possible. Use will be made of a shielded, self-propelled hydraulic crane if the radiation levels in the vicinity of the reactor building are excessive. The generation of steam in the reactor building during destructive tests may fog the camera lenses. However, fogging should not occur until the important portion of the transient has been concluded.

Camera types and locations for the Phase-I photography are shown in Fig. 10. The cameras located near the core and the reactor building, with the exception of cameras 9 and 10, will be equipped with time-marking devices. The timing marks, which are controlled by the sequence timer, will permit correlation of the photographs with the other data from the core, such as power, temperature, and pressure. The timing marks will also permit possible determination of the velocities, momenta and energies of missiles.

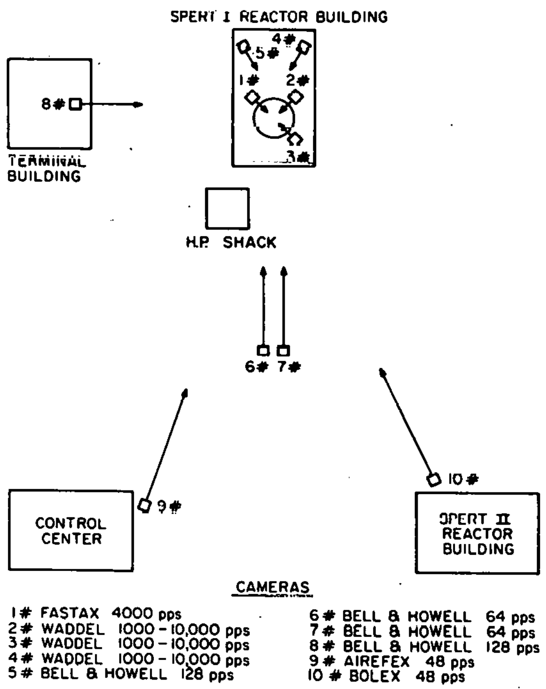


Fig. 10 Location of cameras for Destructive Test Series

Cameras 1, 2, and 3 (Fig. 10) are high speed cameras used in conjunction with periscopes to obtain close-up pictures of the core during destructive tests. Use of the periscopes allows placement of the cameras away from the core, resulting in reduced radiation fogging of the film, and avoids loss of detail resulting from disturbance of the water surface.

Cameras 1 and 2 will view the overall core region to give data on gross effects and on boiling conditions. One of these cameras will be aimed horizontally to view the top edge and one side of the core to photograph the motion of steam bubbles above the core and to record any horizontal and vertical motion of the core assemblies. The other camera will be aimed to view the top and the edges of two sides of the core to record horizontal motion of the top of the core. At the present time, it is planned that camera 3 will obtain a close-up, detailed view of a section of fuel plate placed in the reflector flux peak region, with the object of obtaining information on the onset and nature of boiling occurring on a fuel plate surface during a transient. This information is of importance in attempting to ascertain the nature of the steam pressures developed during a violent excursion.

Cameras 4 and 5 in Fig. 10 will be located in the reactor building at approximately floor level to observe the expulsion of water and other debris from the reactor tank during the violent transients. The use of two cameras here is to provide back-up protection in the event of failure of one of the cameras. Cameras 6 through 10 will be used to obtain long range photographs of destructive events. Cameras 6 and 7 will be located roughly 100 feet in front of the reactor building, while camera 8 will be located at the Spert I terminal building, 400 feet away, to obtain a view of the reactor area at right angles to the view seen by cameras 6 and 7. Cameras 9 and 10, equipped with telephoto lenses, will be located approximately 1/2 mile away from the reactor area. These two cameras will not be controlled by the sequence timer.

Phase II photography of reactor damage will be performed by photographers accompanying the re-entry teams. In the event of excessive radiation levels, use will be made of the self-propelled crane to obtain close-up photographs of damaged core components.

IV. EXPERIMENTAL RESULTS TO DATE

A. Introduction

To date, a number of tests of the Static, Fiducial-Transient and Exploratory-Transient Series have been performed. The static measurements were made to ascertain that the core was operationally adequate for a destructive test, and to obtain measurements of those parameters which are important to the analysis of self-limiting reactor power excursions. Analysis of the results of transient tests serve as the basis for extrapolation to test results of the Destructive Series to be conducted later this year.

B. Static Test Results

1. Initial Core Loading

Initial loading of DU-core began on March 3, 1962, and criticality was achieved on March 5, 1962, with a loading of 20 fuel assemblies and the control rods withdrawn 17.95 in. The indicated critical mass was approximately 2.8 kg of U^{235}. Loading then proceeded until an operational core of 25 assemblies or 3.8 kg of U^{235} was achieved. The operational core loading, designated the DU-12/25 core, was completed on March 6, 1962, and the reactor was critical with the control rods withdrawn 9.20 in. The available excess reactivity of the core was determined to be ~\$8.2 and the shut-down reactivity margin to be ~\$4.4; these were considered adequate for the proposed test program.

2. Rod Calibrations

a. Control Rod

The differential reactivity worth of the control rods over the range of rod travel from the critical position to the fully-withdrawn position was measured by the conventional period method. Both boric acid solution and transient rod poison insertion were used as reactivity shims to permit measurement over the entire range of control rod travel. At given control rod positions, exponential power rises were initiated, with periods in the range between 80 and 15 sec. The excess reactivity above delayed critical was determined by use of the inhour relationship, which is essentially independent of the prompt

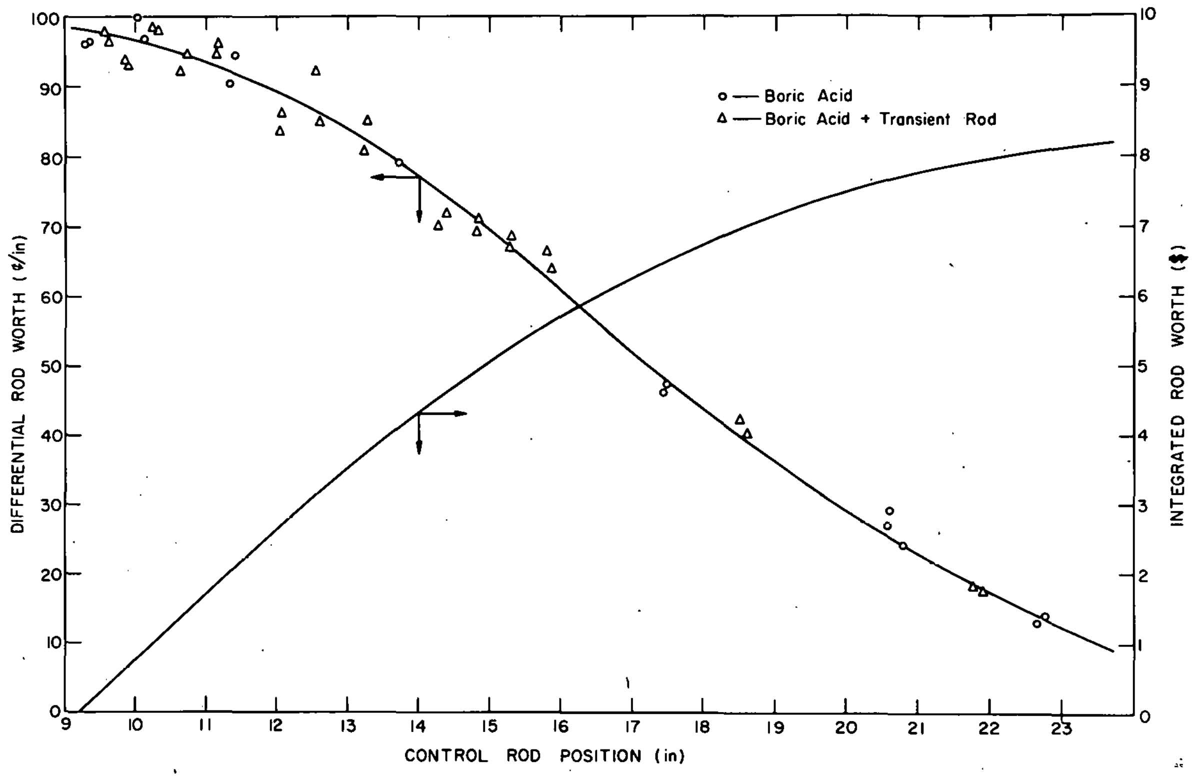


Fig. 11 Differential and integral control rod calibration curves

neutron lifetime in this long-period region.

Differential and integral reactivity worths of the control rods are shown in Fig. 11. Scatter in the data of differential reactivity worth has been attributed to uncertainties of ± 0.02 in. in the rod position increment and to possible inhomogeneity in the concentration of the boric acid solution used to shim the reactivity. The integral curve indicates an available excess reactivity of $8.2, which, if added as a step, would result in about a 1-msec period power excursion.

A shutdown reactivity of $4.4 was inferred from integration of a linear extrapolation of the differential rod worth curve from 9.20 in. to zero inches withdrawn.

b. Transient Rod

Reactivity worth of the transient rod was determined by intercalibration with the calibrated control rods, and the integral transient rod reactivity worth is shown in Fig. 12. The total worth of the transient rod is $7.1, which indicates that the minimum period possible for test purposes is about 1.3 msec, based upon a measured reduced prompt neutron lifetime of 8.2 msec.

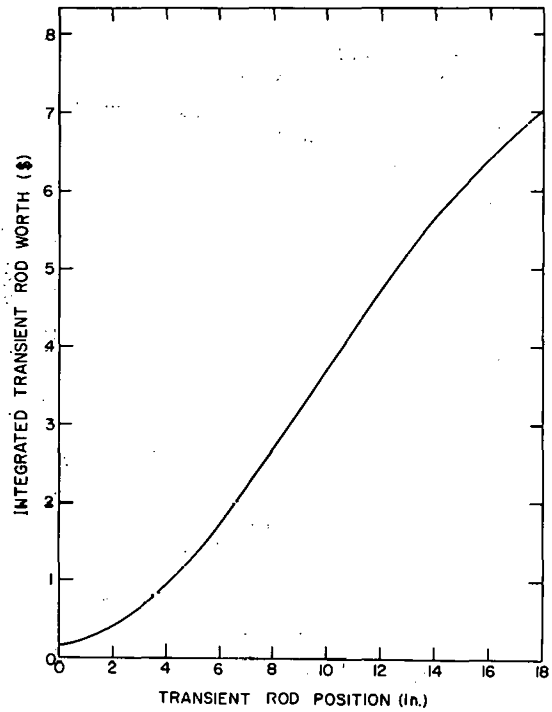


Fig. 12 Integral transient rod calibration curve

3. Void Coefficient

Two types of void coefficient measurements were made; one for uniformly distributed voids and the other for voids distributed at various axial positions in the central region of the core. In both cases, magnesium strips were used to displace water and simulate void.

a. Uniform Void Coefficient

The measurement of the uniform void coefficient was performed by inserting 30-in. long by 0.610-in. wide by 0.159-in. thick magnesium strips in alternate channels of each non-rodded fuel assembly to simulate a uniform distribution of voids. The lateral position of the voids was staggered to reduce interaction. Following determination of the critical position of the control rods, selected voids were removed, a new critical position established, etc., until all the void strips were removed from the core. Reactivity loss for each step was obtained from the change in the calibrated control rod positions. The void coefficient was calculated by correcting the reactivity effect for neutron absorption by the magnesium and by considering only the magnesium within the active length of the fuel region as void. Results of these measurements, shown in Fig. 13, indicate a value for the coefficient of -36¢/% decrease in moderator density.

b. Central Void Coefficient

Central void coefficient measurements were performed by use of 4-in. long by 0.610-in. wide by 0.159-in. thick magnesium strips located in the central region of the core. Each strip was

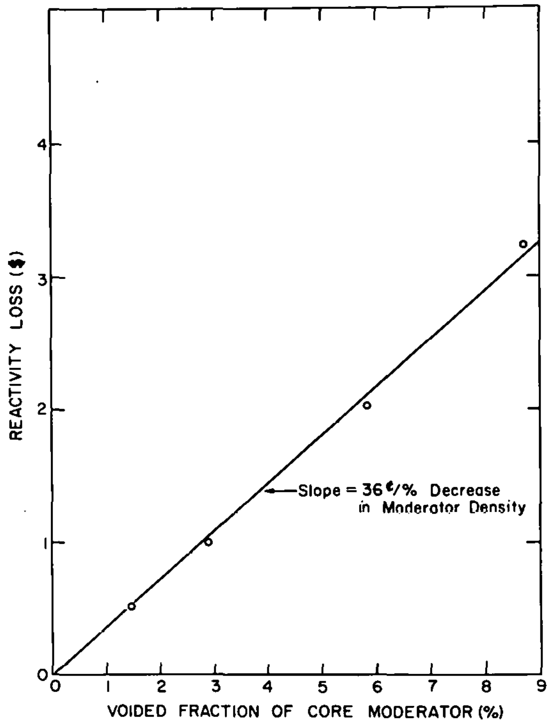


Fig. 13 Reactivity loss as a function of void fraction for a uniform distribution of voids

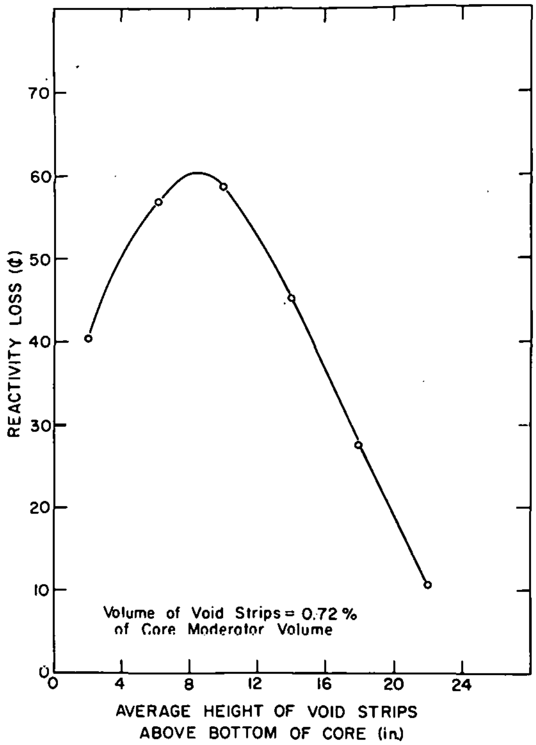


Fig. 14 Reactivity worth of centrally-located, 4-in. long void strips as a function of height above the bottom of the core

attached to the transient rod drive mechanism to enable remote axial positioning of the voids. The axial profile of the void coefficient was obtained from the control rod worth by determining the critical rod position as a function of void position. The results of this measurement are shown in Fig. 14. The peak of the void worth curve occurs at a void position of about 8.5-in. above the bottom of the fuel. The maximum central void coefficient is -84¢/% decrease in moderator density. This coefficient has been determined to be independent of void volume for void volumes as large as 370 cm^3.

4. Neutron Flux Distribution

The steady-state neutron flux distribution was determined from activation of twenty-nine cobalt wires located in the core as shown in Fig. 15. The wires extended the full length of the fuel plates and were irradiated for 135 minutes at a power level of 95 kw.

Figures 16 - 20 illustrate representative vertical and horizontal flux profiles at selected core positions. The normalized flux distributions are plotted as functions of height above the bottom of the

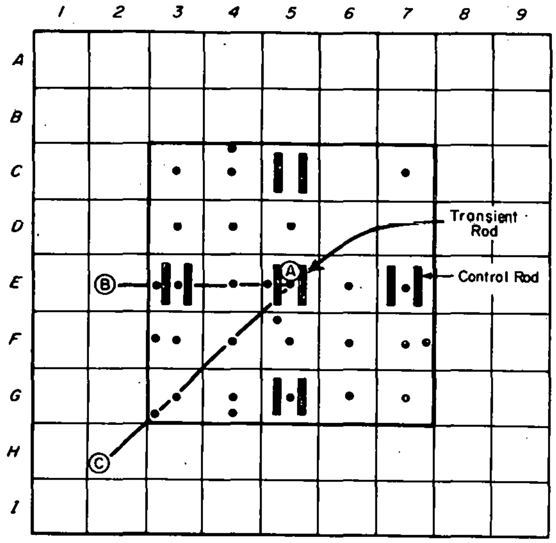


Fig. 15 Flux wire activation positions

core or distance from the core centerline. The maximum flux was determined to be in position E5-5 at about 8-in. from the bottom of the core (E5-5 means the 5th water channel of the fuel assembly in the E5 grid position). The peak-to-average flux ration was determined to be 2.4.

5. Isothermal Temperature Coefficient

During the preceding neutron flux distribution measurement, water in the reactor vessel was heated 9.4°C and an isothermal temperature coefficient of -2.1 ¢/°C was determined from the change in the control rod critical position.

6. Reduced Prompt Neutron Lifetime

A value of the reduced prompt neutron lifetime, ℓ/β_{eff}, for the DU-12/25 core was determined by analysis of subcritical statistical behavior of the neutron population (9, 10, 11). The value of the reduced prompt neutron lifetime obtained in this experiment was $\ell/\beta_{eff} = 8.2 \pm 0.4$ msec.

C. Transient Test Results

1. Summary

At the time of this writing (June 1, 1962), the DU-12/25 core in the Spert I facility has been subjected to 28 self-limiting power excursions with initial asymptotic periods in the range from about 1 sec to about 5 msec. These tests, comprising portions of the Fiducial-Transient and Exploratory-Transient Series, demonstrated that the DU-12/25 core responds in a similar manner to other aluminum-clad, light-water-moderated cores which have been tested at Spert. However, for tests with periods of about 8 msec and below, some significant departures have been observed from the expectations based upon data from previous cores.

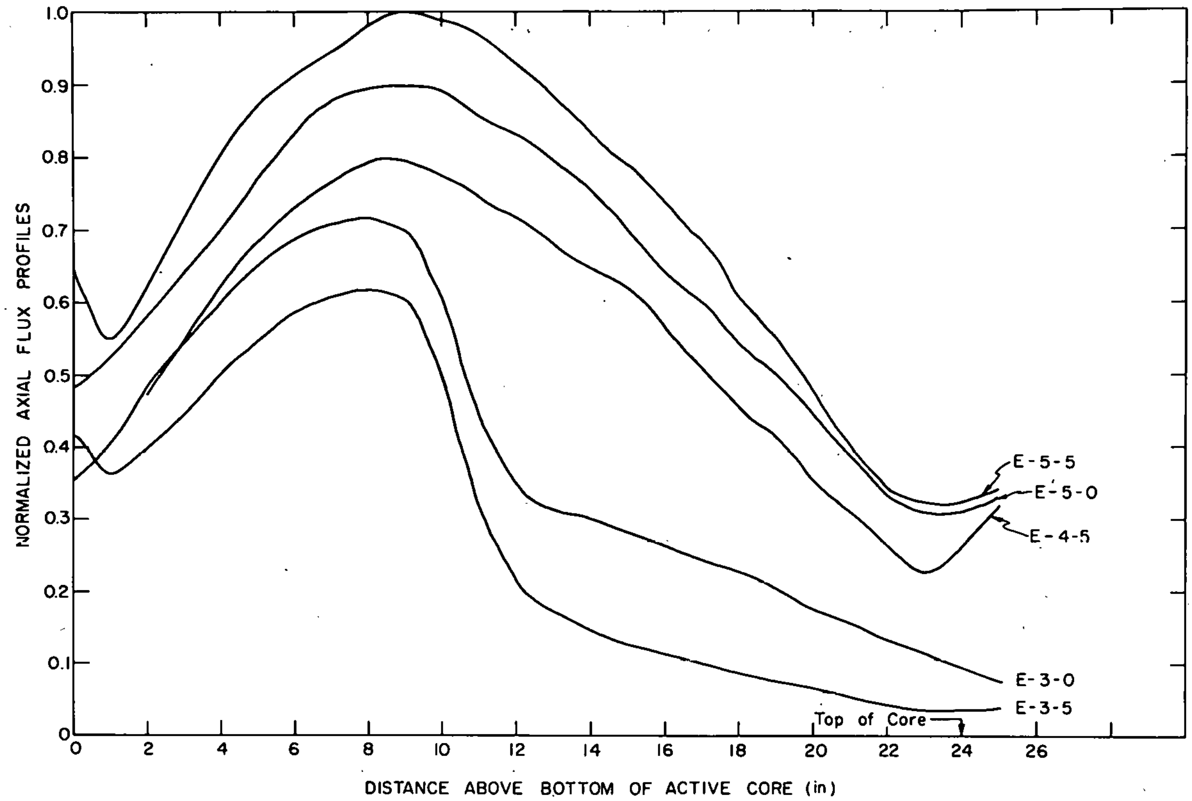


Fig. 16 Vertical flux profiles in-fuel assemblies E-3, E-4, and E-5

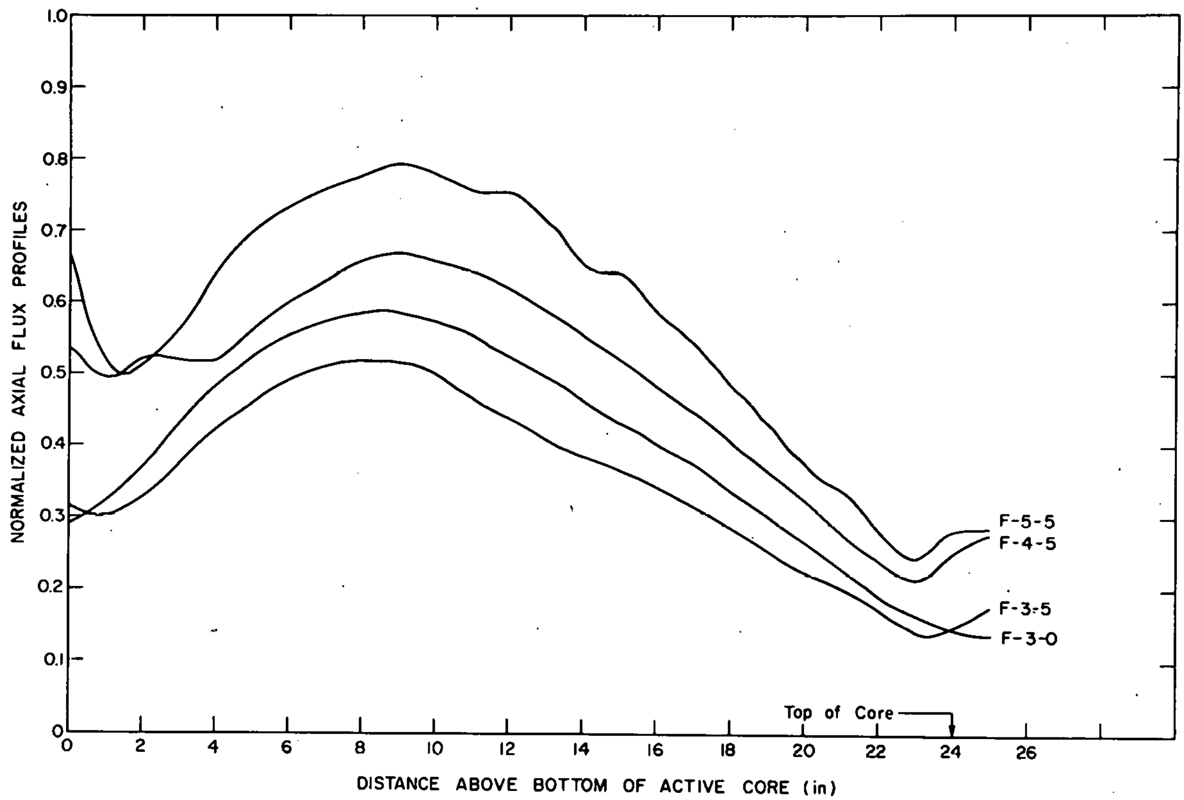


Fig. 17 Vertical flux profiles in-fuel assemblies F-3, F-4, and F-5

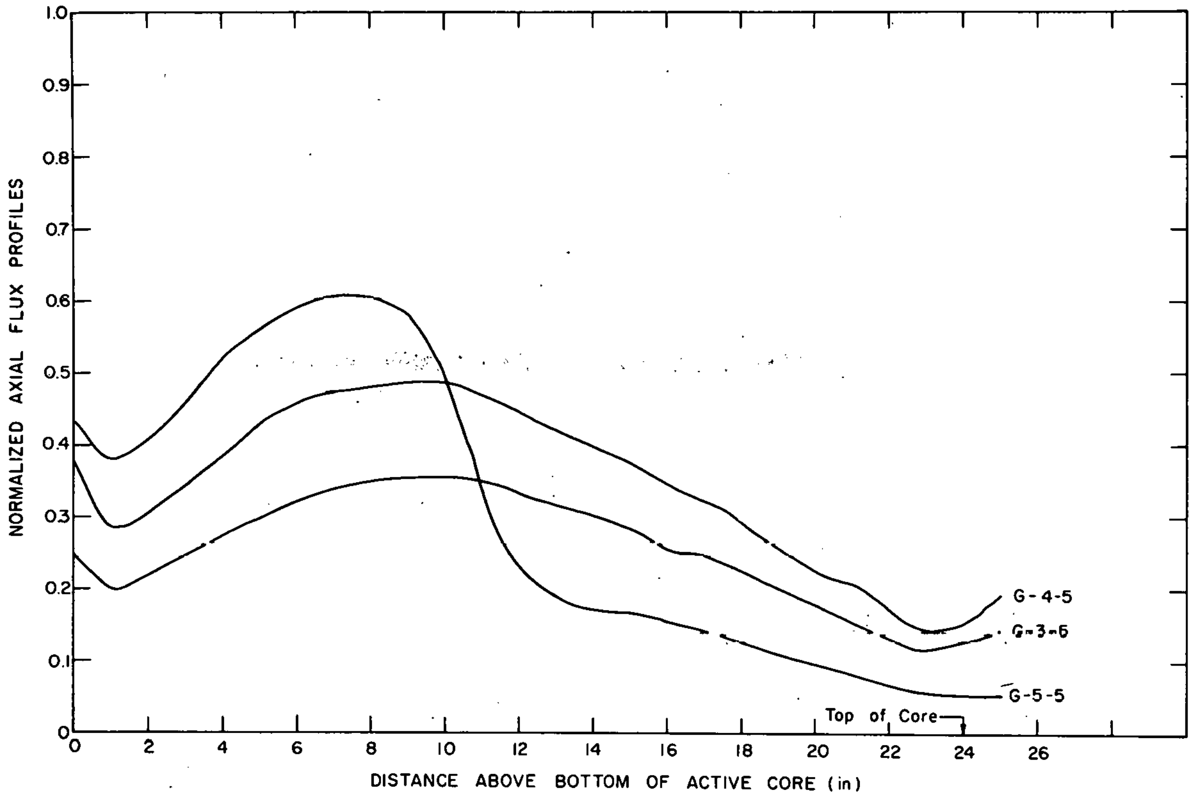


Fig. 18 Vertical flux profiles in-fuel assemblies G-3, G-4, and G-5

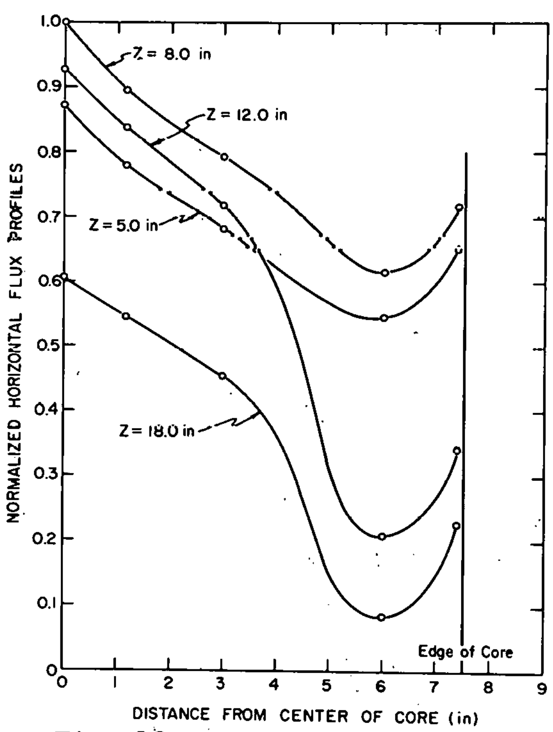


Fig. 19 Horizontal flux profiles along direction A-B (Fig. 15)

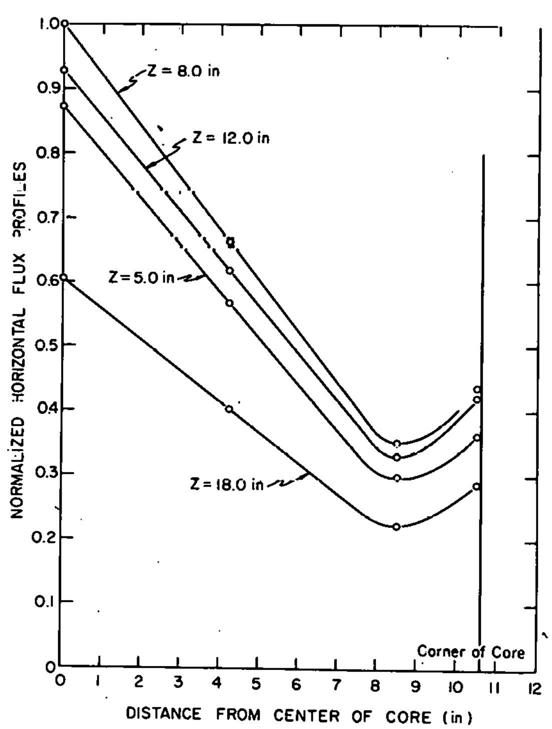


Fig. 20 Horizontal flux profiles along direction A-C (Fig. 15)

The Fiducial series of transients, included excursions with periods down to about 7 msec. This series provided important preliminary data for comparisons with other cores and for extrapolation to shorter periods. Damage was observed in this series only to the extent of slight fuel-plate "bowing". In the last of the series, a peak power of 660 Mw was achieved with a total energy release of about 10 Mw-sec.

With the onset of significant fuel plate damage, the Exploratory series was initiated for the purpose of continuing the investigations into a shorter period domain. This series is now in progress and is aimed toward providing more information and better extrapolations into the region of full-scale destructive tests. Destructive, or maximum-consequence tests are planned for the latter part of this year. The procedure of testing involves successive tests with only small reductions of the initial period and correspondingly short extrapolations of the data.

The most recent of the exploratory tests achieved an initial period of 5 msec, a peak power of about 1130 Mw, and a total energy release in the initial burst of about 17 Mw-sec. Melting of several fuel plates was observed near the center of the core and plate deformation was widespread and often severe. However, all measured variables were within the range of prediction, no significant transient pressures were observed, and contamination of the water moderator was negligible.

2. Description of Tests and Test Data

Prior to a discussion of experimental results obtained from the DU-12/25 core, it is appropriate to describe the nature of these tests and the data obtained. The previous section on measurements has already described purposes and methods of instrumentation and the following will serve to illustrate the nature of typical data and correlations.

All tests in the present program are of the "step-transient" type, i.e., reactivity is inserted into the system suddenly and the reactor power is allowed to self-limit under the influence only of inherent, reactivity-compensating mechanisms. As shown in Fig. 21, the initial response is an exponential rise of both the power and the energy release. During this time, and as a direct result of the energy release, fuel plate temperature also begins to rise exponentially.

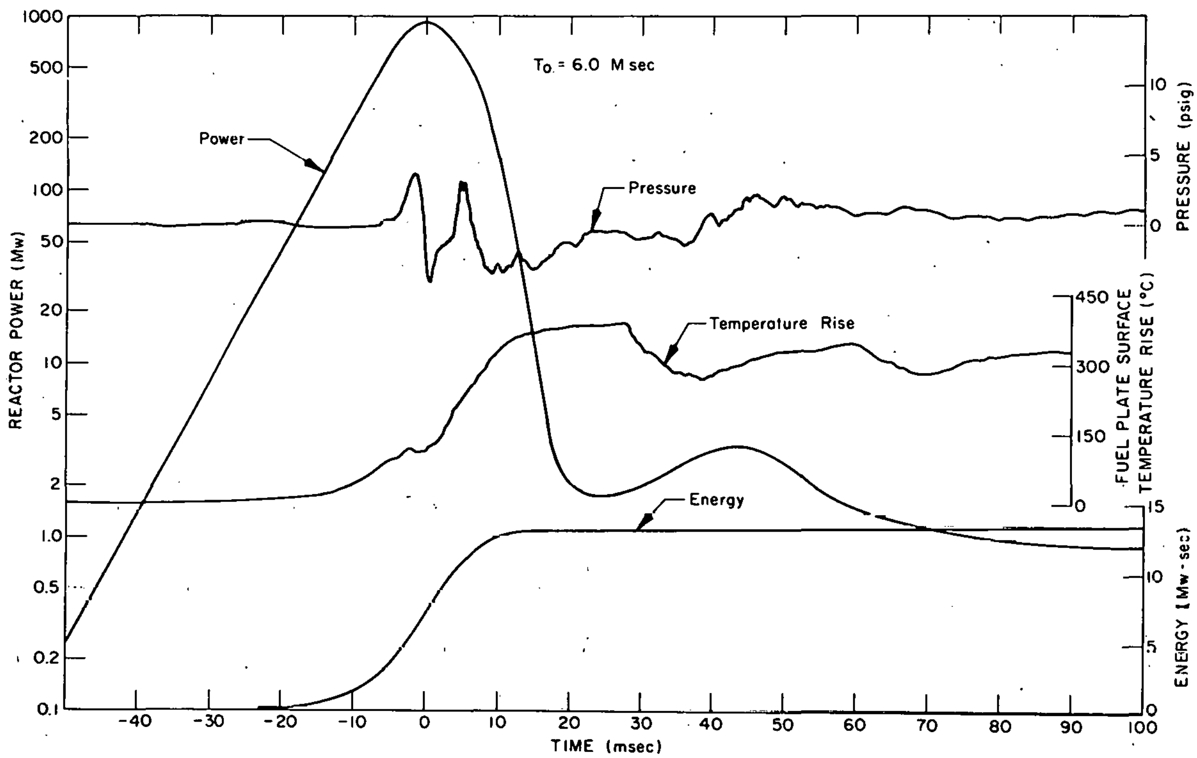


Fig. 21 Power, energy, temperature, and pressure behavior during a 6.0-msec period reactor test.

At some point during the exponential power rise, energy-dependent shutdown mechanisms, such as boiling and thermal expansion of metal and water, begin to remove reactivity from the system and cause the power to deviate from the exponential. The time and power level at which these mechanisms first become apparent depends upon several factors, principally, the energy and reactor period. At the time of peak power ($t=t_m$) during a short-period transient (i.e., initial period less than about 20 msec), all of the prompt reactivity has been removed from the system by these mechanisms and power begins to decline.

At the time of peak power, energy, fuel plate surface temperatures and the excess reactivity in the system, $\delta k(t)$, are changing rapidly. As indicated above, only the prompt reactivity has been removed at the time of peak power during a short-period test, the remaining excess reactivity is still positive at about one dollar, $\delta k(t_m) \simeq \$1$. As seen in Fig. 21, power declines rapidly after the peak and seeks a quasi-equilibrium value several decades below the peak. In the region of peak power and for at least a decade of power decline, the time rate of change of reactivity is proportional to the time rate of change of the logarithmic slope of the power, and it is apparent in transients such as these that shutdown mechanisms remove far more reactivity than that which has been removed at peak power. Shortly before the post-burst minimum, the reactivity reaches a minimum, which for the case shown is about - \$4.

In the example shown, the reactivity compensation occurred early enough and with sufficient magnitude to limit the energy release to about 13 Mw-sec. As a direct consequence of this, fuel plate temperature was also prevented from going higher than about 560°C. Nevertheless severe thermal deformations of the fuel plates resulted from the excursion. A small delay in the onset of shutdown or a reduced effectiveness of the shutdown mechanisms would have resulted in higher energy release and melted fuel plates, whereas, an improved shutdown effectiveness would have resulted in lower energy and temperatures, and in reduced damage to the core.

As further indicators of shutdown efficiency, both energy and peak temperature data are studied following each transient test. Data presentation usually includes plots of the peak power, $\phi(t_m)$, energy at peak power, $E(t_m)$, temperature at peak power, $\theta(t_m)$, total burst energy, and maximum temperature, $\theta(max)$, as functions of the initial asymptotic reciprocal period, α_o.

The DU-12/25 core is under-moderated, with the measured void coefficients of reactivity negative throughout the core for the range of void volumes investigated. The shutdown mechanisms of interest are primarily fuel plate and moderator thermal expansion and moderator boiling. Whereas the fuel expansion is "prompt", water expansion, which is dependent upon heat transfer from the fuel plate, does involve a delay time which is characteristic of the fuel plate material and geometry. Both metal and water expansion processes have an approximate linear dependence upon available energy, but, the response may be non-linear in time due to non-linear heat transfer. Moderator boiling is inherently a more effective shutdown mechanism than thermal-expansion processes as a result of the very large voids produced when water is converted to steam. The experimental data obtained indicates that long period (i.e.,

$\alpha_0 \lesssim 1 \text{ sec}^{-1}$) shutdown is probably effected by non-boiling mechanisms alone, but that shorter period transients involving larger reactivity compensations rely to an increasing degree on steam void formation for shutdown.

Since void formation involves ejection of moderator from the core, the process of shutdown will, in general, give rise to pressures within the core, which are constrained by core materials. Although only very small pressures (i.e., less than 1 psi) are generated due to the thermal-expansion processes mentioned above, the vapor pressure of superheated water can be very high and as such is a potential source of destruction during short-period power excursions wherein superheating exists. Even though boiling can be a very effective shutdown mechanism over a certain range of reactivity insertions, it may also be the source of pressures causing disassembly and blast effects in large reactivity-insertion, destructive power excursions. Measurements have indicated superheats of up to 50°C occurring prior to the onset of boiling during some transients(12,13). At later times during these transients, it can be assumed that some of the steam and water acquire temperatures about equal to the plate surface temperature, which in recent tests has exceeded 600°C, or a superheat of about 500°C. Thermal gradients constitute another source of damage, since these can cause stresses in fuel plates which exceed the yield point. Temperature measurements obtained during short-period tests have indicated large temperature fluctuations and gradients. Such temperature behavior results in plastic deformation of fuel plates even at temperatures far below melting and to such a degree that plate replacement becomes necessary. Deformation of this nature has manifested itself during several tests in the Exploratory Series as "rippling", "bowing", or failure by fracturing of fuel plates. Although techniques for the measurement of transient strain in the reactor environment are not well developed, some attempts are being made to obtain useful strain data. At the present time, strain measurements have been successfully accomplished on non-fuel-bearing structures and results indicate that this measurement may become a primary indicator of imminent core structure damage.

In summary, it appears that there exist three generically different types of core damage: meltdown, thermal stresses, and pressure distortion. Data on these three types of damage are obtained during each test by temperature, pressure, and strain measurements, and by visual inspection of the core after the test.

3. Experimental Data and Comparison with Other Core Data

a. Power, Energy, and Temperature

The maximum reactor power acheived during the initial burst of the self-limiting power excursion as a function of the reciprocal of the initial asymptotic period is shown in Fig. 22. For purposes of comparison, the data obtained in Spert I with the A-17/18 core, the B-24/32 core and the B-12/64 core are also shown[14], as are the data that were obtained with the Borax I tests[7]. All of these cores, with which the data from the DU-12/25 core are compared, are aluminum plate-type cores with highly enriched fuel which were tested in an open pool of light water. The nuclear parameters of these cores are compared in Table III.

As seen in Fig. 22, the maximum power for a given period is lower for the DU-12/25 core than for the other cores. This is to be expected since the DU-12/25 core has a lower heat capacity than the other cores and a comparable, or larger, void coefficient of reactivity. Shutdown mechanisms, as discussed above, are primarily temperature (or energy) dependent, and a core with a reduced heat capacity can effect shutdown earlier and therefore at a lower initial

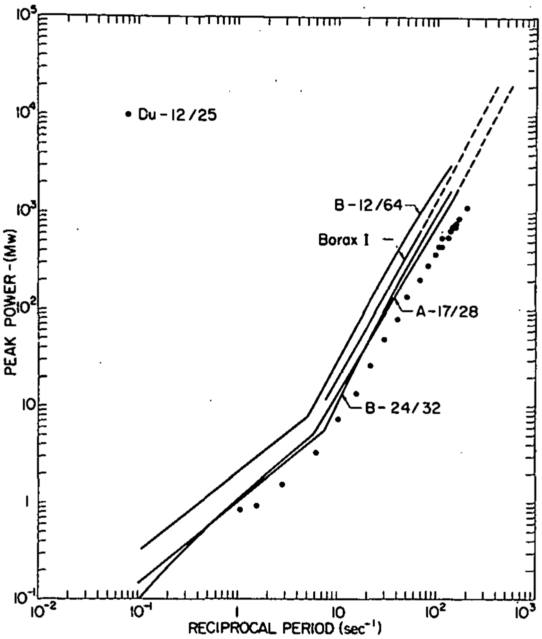


Fig. 22 Peak power as a function of reciprocal period for several aluminum plate-type cores

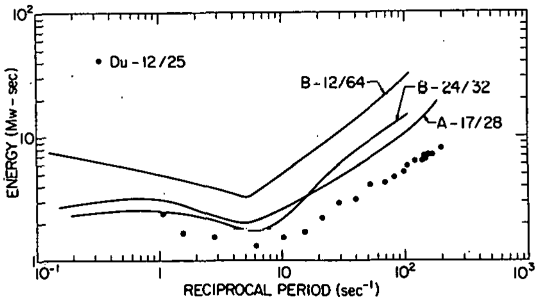


Fig. 23 Energy at the time of peak power as a function of reciprocal period for several aluminum plate-type cores

TABLE III

Comparison of Static Core Characteristics

Parameters	Core				
	DU-12/25	A-17/28	B-12/64	B-24/32	Borax - I*
Core Volume (cm^3)	8.8×10^4	9.7×10^4	2.2×10^5	1.1×10^5	1.2×10^5
M/W Ratio	0.66	0.79	0.46	1.1	0.63
Total Plate Area (cm^2)	2.2×10^5	3.5×10^5	6.2×10^5	6.2×10^5	4.5×10^5
Total U^{235} Loading (kg)	3.8	4.7	5.4	5.4	4.2
Excess Reactivity ($)	8.2	5.2	4.3	6.6	3.1
Temperature Coefficient 20°C (¢/°C)	-2.1	-0.67	-1.8	-1.1	-0.50
Moderator Void Coefficient					
Core Average (¢/%)	-36	-25	-15	-40	-36
Core Average (¢/cm^3)	-0.067	-0.046	-0.0093	-0.073	-0.051
Central (¢/cm^3)	-0.16	-0.093	+0.008	-0.17	-0.088
Peak/Average Flux Ratio	2.4	2.0	2.2	2.5	2.0
ℓ/β_{eff} (msec)	8.2	7	11	7	8.7
H/U Ratio	360	320	760	270	440

* Effective delayed neutron fraction used: 0.0075

power rise and energy release. A comparison plot of energy release at the time of peak power, $E(t_m)$, is seen in Fig. 23.

It is noted that the shape of the curve defined by the power data points for the DU-12/25 core is, in general, similar to that for other cores. In particular, it appears that the DU-12/25 peak power is quite similar to that of the A-17/28 core, except that it is a factor of 1.5 times lower. Also, in the energy plots of Fig. 23 it can be seen that the relationships between $E(t_m)$ for the various cores as a function of α are quite similar to the corresponding relationships between maximum powers, indicating that the power burst shape to time of peak power is nearly the same for all of these cores.

Even though $E(t_m)$ for the DU-12/25 core is considerably less than that observed for the B-24/32, B-16/32, and A-17/28 cores, the maximum measured fuel plate surface temperatures at time of peak power, $\theta(t_m)$, for these four cores are seen in Fig. 24 to be nearly the same in the longer period region. The similarity of these data for several cores is, in part, to be expected as a result of the approximately equivalent void coefficients of reactivity, with the consequence that the remperature rise and void change would have to be approximately equal to provide a given reactivity compensation. The B-12/64 core, however, has a significantly lower density or temperature coefficient and consequently requires a larger temperature rise to provide the same reactivity effect.

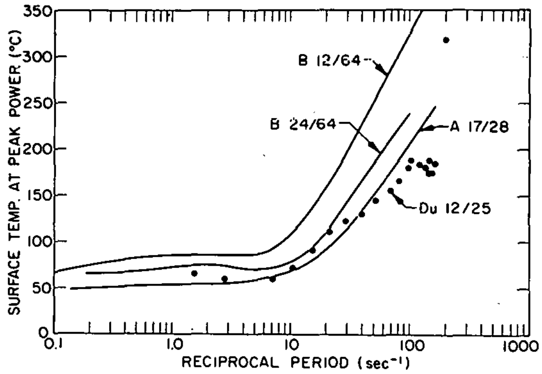


Fig. 24 Surface temperature at time of peak power as a function of reciprocal period for several aluminum plate-type cores

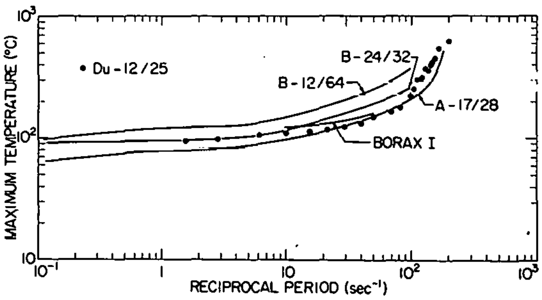


Fig. 25 Maximum surface temperature as a function of reciprocal period for several aluminum plate-type cores

Maximum fuel plate surface temperatures are shown in Fig. 25 for the same aluminum cores. As can be seen, the data are nearly equal for all except the B-12/64 core, which, for the reasons stated above, attains higher values. The rapid rise in maximum temperature as period is decreased is thought to be a result of "vapor blanketing." With the onset of temperatures much above 100°C, vapor blanketing can be established early in a power excursion, allowing the temperature to increase rapidly.

This behavior is demonstrated in Fig. 26, in which the response of a typical thermocouple is shown in idealized form. A break from the initial exponential at t_2 usually occurs in the original data after several degrees of superheating. The decreased rate of temperature rise after t_2 implies a marked increase in heat transfer rate. At t_3, several milliseconds later, a temperature "setback" usually occurs followed by a rapid temperature rise. Experiments have shown that the setback is caused by an extremely high heat transfer rate as a massive and coherent volume of steam is suddenly generated on the fuel plate. Upon creation of this steam, heat transfer is greatly inhibited, as is reflected both in the rapid temperature rise during the remaining power burst, and in the relatively slow cooling rate after the burst. It is an important experimental result that as shorter periods are attained, boiling heat transfer becomes less and less effective as a temperature suppressor. As a consequence, fuel plate temperature must rise more nearly in proportion to the total nuclear energy release. In the extreme limiting case, energy release and temperature rise should become strictly proportional throughout a power excursion. Indication of this proportionality can already be seen by a comparison of the temperature and energy in Fig. 21. Referring again to Fig. 25, the following regions may be defined.

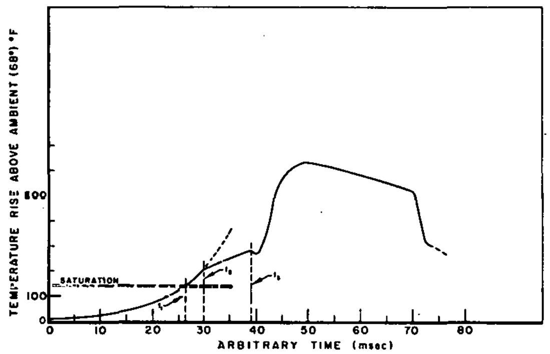


Fig. 26 Idealized temperature behavior during a self-limiting power excursion

I $\alpha_0 \lesssim 20 \text{ sec}^{-1}$,	Boiling heat transfer adequate to prevent appreciable superheat of plates. Maximum temperatures about 100°C.
II $20 \lesssim \alpha_0 \lesssim 100 \text{ sec}^{-1}$,	Transitional region. Occasional film blanketing reduces effectiveness of boiling heat transfer. Maximum surface temperatures begin to increase with α.
III $\alpha_0 > 100 \text{ sec}^{-1}$	θ(max) primarily limited by E(total) with very little effect from boiling heat transfer.

In Region I there exists a very weak dependence of θ(max) upon α since boiling is almost completely effective in preventing superheat. Region II, however, is characterized by a gradual increase in superheat and a complex dependence upon the conflicting roles of energy deposition and heat transfer. Finally, in Region III, the maximum heat content in a fuel plate is nearly proportional to E (total). If other factors remain unchanged (i.e., the importance of boiling heat transfer, flux distribution, etc.) then E (total) may provide a good indication of the overall seriousness of a short period transient in terms of temperatures reached, possible meltdown, and perhaps, strain, pressures, and contamination.

As seen in Fig. 23, $E(t_m)$ is a very regular and predictable quantity. Experience with previous cores has shown that the total energy, E (total), in the region of $\alpha_0 > 50 \text{ sec}^{-1}$ is also very regular with nearly the same dependence upon α_0 so that the ratio, R, of the total energy to the energy released to time of peak power has been nearly constant on several cores with a value of approximately 1.5. Tests performed on the DU-12/25 core also demonstrated similar behavior in this period region. However, as tests with periods below 10 msec were studied, the ratio, R, has become much larger as indicated in Fig. 27. It is clear that in this case the total energy as a function of α_0 is increasing at a faster rate than the energy release at time of peak power. Since a constant value of R implies a constant power burst "shape", the increasing value of R implies a change in burst shape as shorter periods are attained. In particular, it means that there is a tendency in the DU-12/25 core to sustain high power levels following peak power. This has appeared as a "hump" on

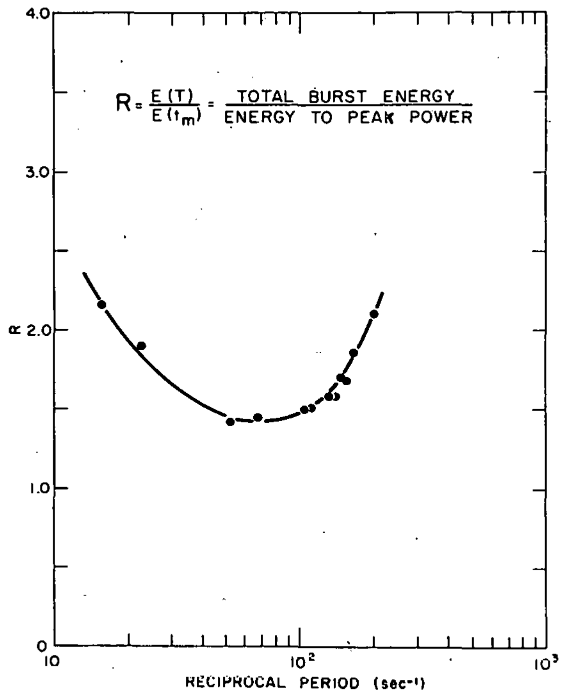


Fig. 27 Ratio of total burst energy to energy released at time of peak power as a function of reciprocal period

the backside of the power burst and can be seen in Fig. 28, where several burst shapes have been normalized to a common maximum value and plotted in terms of time-in-periods to demonstrate this result.

The change in burst shape which is taking place with increasing α_o has several implications besides those of an increased energy release. More important, is the apparent loss of shutdown effectiveness. As discussed previously, a change in reactivity at about the time of peak power is directly related to a change in slope of the log power, so it can be seen from Fig. 28 that since post-peak curvature of the

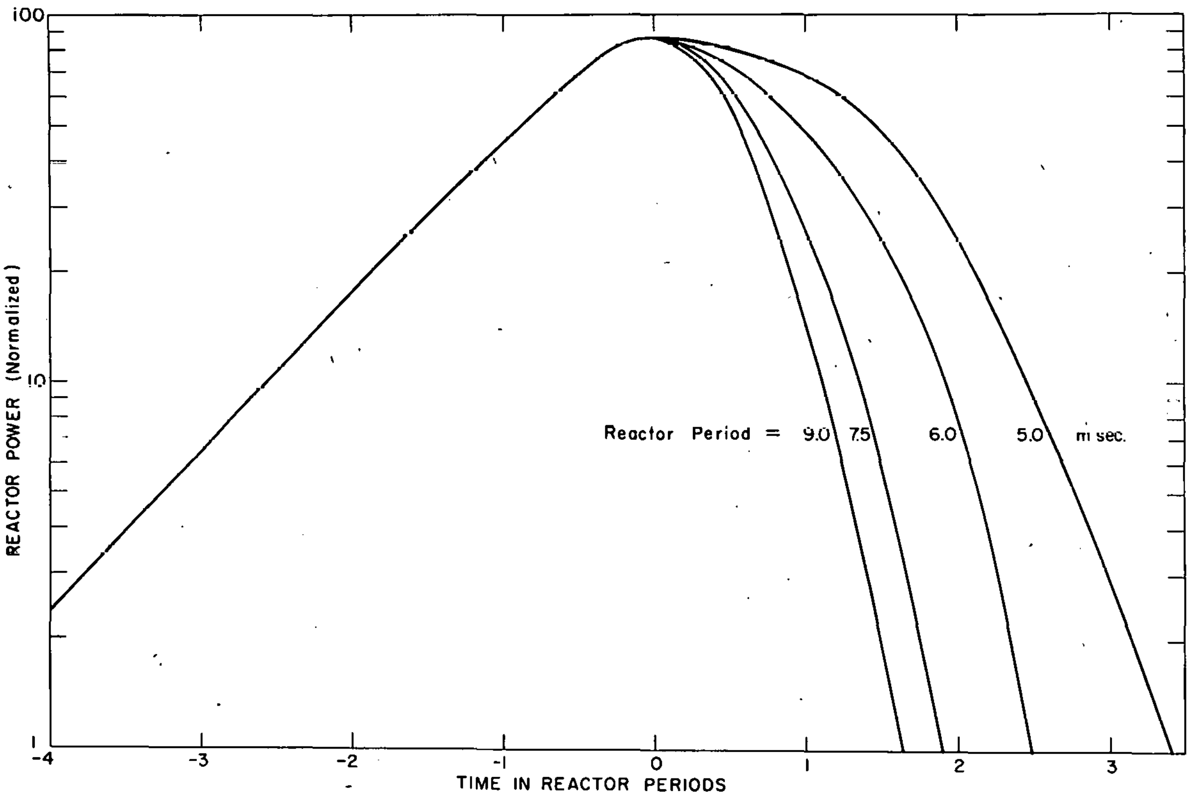


Fig. 28 Normalized power burst behavior as a function of time in reactor periods

power trace decreases with decreasing period, the rate of production of shutdown reactivity is correspondingly reduced during this part of the excursion.

Although burst-shape change is easily discerned when power is plotted as in Fig. 28, reactivity effects are more easily interpreted when the power is plotted as a function of real time as in Fig. 29. Here, only the time and magnitude of peak power are normalized and time is in milliseconds. The post-peak curvature relationships are still in the same sequence as discussed above, although less prominent. In the region of time just before peak power, curvature increases as the period becomes smaller. Thus a situation exists before peak power which is just the converse of the post-peak behavior; that is, compensating reactivity is initially produced more rapidly as period is decreased. This is expected, since, with short-period excursions, both the amount of superheat and the fraction of the core experiencing superheat are increased.

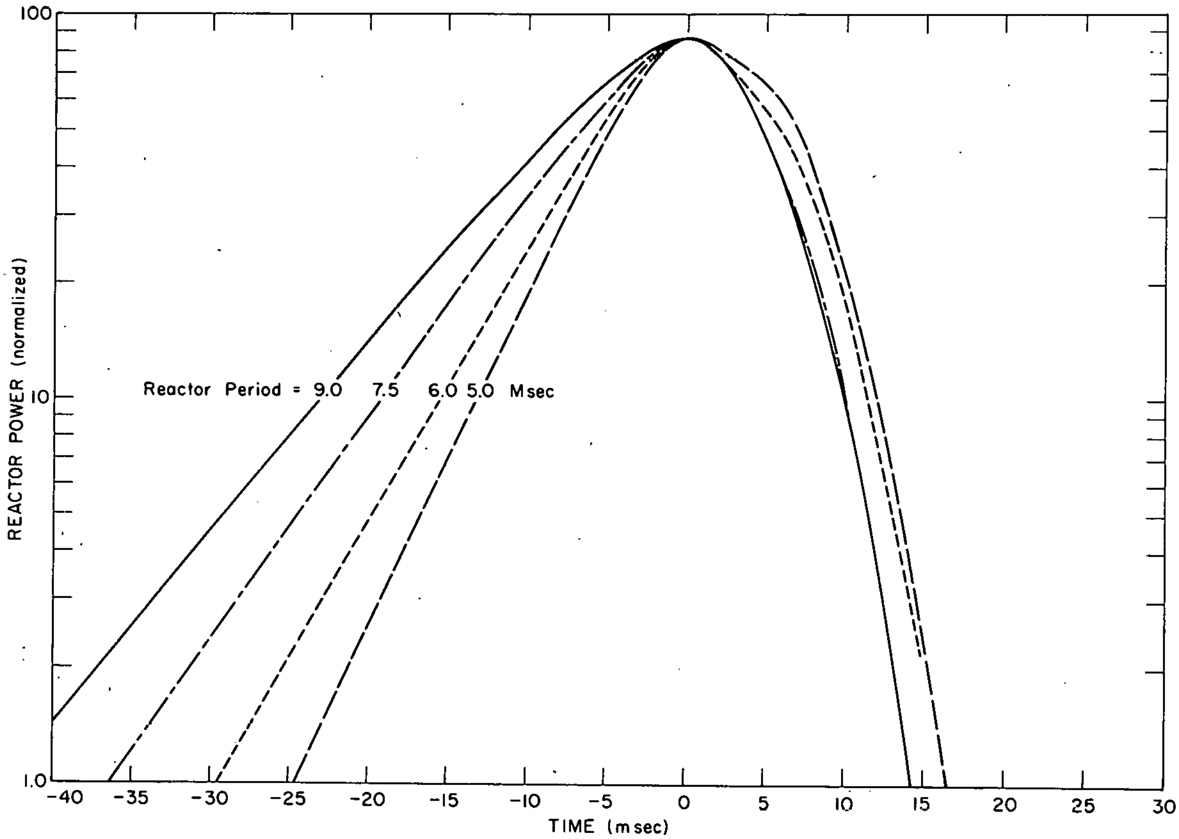


Fig. 29 Normalized power burst behavior as a function of time

Experiments with other Spert reactors have shown[12] that when a fuel plate is heated rapidly while submerged in water, some degree of superheating is experienced before boiling occurs. When boiling finally does occur under these conditions, its appearance is nearly instantaneous and its displaced volume is apparently constant. That is, within a short period of time (~ 1/2 msec) a type of boiling will arise on a superheated plate and maintain a nearly constant average bubble size and bubble density. After a short time (between t_2 and t_3 on Fig. 26), this "initial phase" of boiling is interrupted and replaced by extensive film blanketing. Thus, referring again to the rate of reactivity compensation, it is reasonable to expect that since short-period excursions involve higher initial superheats, larger fractions of the core become effective in producing initial-phase boiling and the initial rate and total magnitude of compensation are correspondingly increased. In the shortest period transient shown, $\tau_0 = 5$ msec, results indicate that initial-phase boiling had occurred on about two-thirds of the entire fuel plate surface area by the time of peak power.

b. Reactivity Compensation

Fig. 30 shows the results of preliminary calculations of compensated reactivity for the same four transients as in the preceding figures. In this graph, time is taken from an arbitrary zero and the four curves have been shifted in time such that the initially rising portion of all the curves coincide in order to demonstrate more clearly the time dependencies of the compensation processes. Occurrence of peak power is indicated for each case.

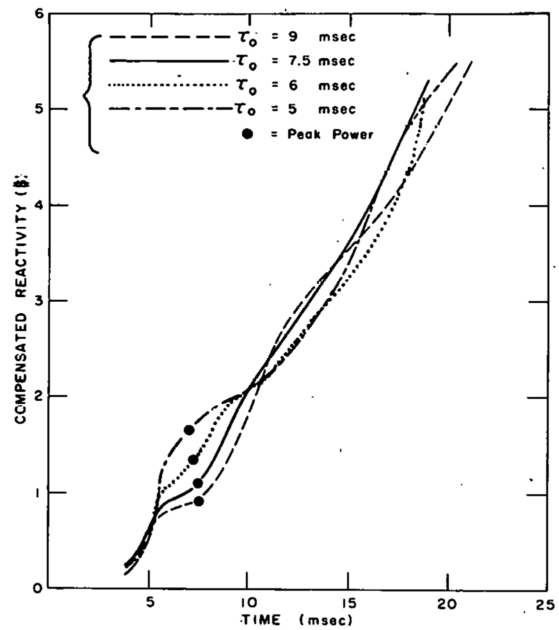


Fig. 30 Compensated reactivity during four different power excursions

The trend mentioned above toward more rapid initial reactivity compensation is apparent. In particular, it is noted that the time required for compensation of the prompt reactivity, δk_p, decreases even though there has been nearly a two-fold increase in δk_p, between the 9-msec case and the 5-msec case. Prompt reactivity, δk_p, is defined as: $\delta k_p = (\ell/\beta_{eff})\ \alpha(t)$ where $\alpha(t)$ = d/dt (log power). Thus, at peak power where $\alpha(t_m) = 0$, all prompt reactivity has been compensated and about one dollar "delayed" reactivity remains in the system uncompensated, i.e., $\delta k(t_m) \approx \$1$.

As indicated in Fig. 30, the rate of reactivity compensation for several milliseconds immediately after peak power, decreases as the period decreases, and this behavior has given rise to the power behavior discussed above. Since all curves shown in Fig. 30 tend to group together several milliseconds after peak power, it appears that the increased demand for prompt reactivity compensation with short-period excursions is met only by delaying (for a short time) the buildup in post-peak reactivity compensation. Ultimately, the amount of compensated reactivity obtained appears to be independent of reactor period for the periods shown.

c. Pressure

Pressure measurements in transient tests are, in general, sensitive to many variables such as position, orientation, and nearness to boundaries, making the interpretation of pressure data difficult. The importance of dynamic pressures as potential sources of damage in the destructive test has made the measurement of pressure one of the most important ones, to which a great deal of attention has been given including the provision of improved instrumentation. Details of the instrumentation have been described in Section III. While "point" measurements of pressure would be valuable in the study of transient boiling and other pressure-generating mechanisms, pressure transducers which are small enough to be inserted inside the core are not available at present. The pressure measurements which are made externally to the core in effect "see" the entire core as a source, or, in other words, record the cumulative external response to a distributed source. These external pressure measurements will greatly aid in the study of reactor containment since, currently, this phase of study must rely heavily upon theoretical considerations.

The present effort toward the prediction of transient pressures must also rely upon many theoretical considerations since there exists virtually no information upon which to base predictions or study. A single measurement made during the Borax-I destructive test indicated about 6000 psi as a maximum transient pressure. Other considerations of vessel failure have substantiated the order-of-magnitude of this measurement but no information is available concerning the time history of the pulse or its relationship to other variables. Likewise, the only other destructive excursion on record, the SL-I accident, has indicated the presence of high pressure but absolute magnitudes are yet conjecture.

The Spert I destructive testing program, includes effort toward the study of several pressure effects and dependencies. Among these are the following:

(1) Maximum pressure - as an extrapolable quantity and possible indicator of total damage.

(2) Vessel loading in terms of both peak pressure and impulse, an extrapolable quantity and possible indicator of vessel failure.

(3) Pressure profile (time dependence) and relationships to various fuel plate temperature measurements - to aid in the study and understanding of shutdown mechanism growth, propagation of pressure through steam-water mixtures, and "chugging" effects.

(4) Pressure directionality - to study core geometry and directional aspects of pressure sources.

These and other pressure effects will be studied.

Approximately twenty transducers are planned for installation within the reactor vessel to begin this study. As indicated in Section III a large fraction of these transducers are installed for "backup" measurements. That is, since pressure transducers are capable of reliable pressure indication for a range of only about two decades (i.e., 1 psi to 100 psi), additional transducers must be located at all positions of interest to cover the range of possible and expected pressures. Complete range coverage for all positions would require an excessive number of transducers and readout channels, and is not feasible. Present instrumentation, however, includes at least two transducers to provide pressure data for all ranges of pressure up to about 10,000 psi.

The measured values of peak transient pressure at several positions in the vessel are shown in Fig. 31 as a function of reciprocal period.

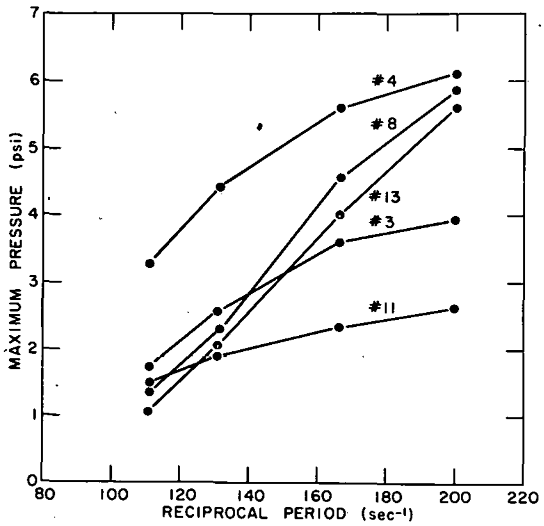


Fig. 31 Maximum transient pressure as a function of reciprocal period for five pressure transducer positions (refer to Fig. 7)

Numbers on the curves refer to transducer locations as presented in Table II. The peak pressures obtained from the DU-12/25 core are, in general, lower than the pressures expected from consideration of data obtained in other cores. The largest pressure observed in previous Spert I cores was of the order of 80 psig. Further, as seen from the data, no unified trend is yet apparent as periods are decreased whereas previous core pressure data generally demonstrated a marked period dependence; that is, peak pressures were observed to increase apparently as some power, n, of the reciprocal period, with n between unity and about five, depending upon the core and the nature of the measurement.

d. Discussion of Results from a 5-msec Period Test.

The most recent test performed in the Exploratory Series (Run No. 28) attained a period of 5.0 ± 0.1 msec, a peak power of 1130 Mw and a total energy release of 17 Mw-sec with the surface temperature reaching the melting point of aluminum, 650°C. Selected plots of the preliminary data are shown in Figs. 32, 33, and 34.

Fig. 32 demonstrates the general nature of several measurements made during Run No. 28 and the temporal relationships to be discussed. The power burst shape to a time shortly after peak power has been discussed above. Following the "hump" of this burst, the data indicates a very rapid power decline to an initial minimum of about 2 Mw, approximately 2.7 decades below peak power. A single oscillation is seen to occur at about t = 43 msec and this is followed by a quasi-equilibrium

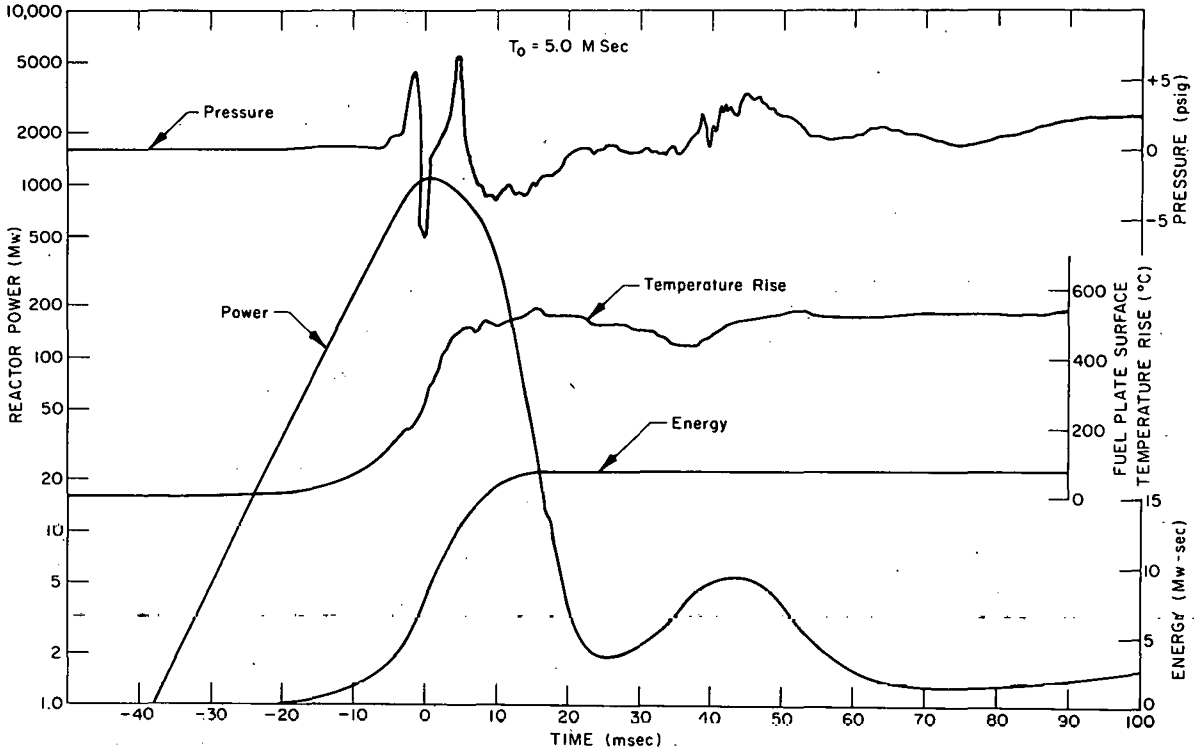


Fig. 32 Power, energy, temperature, and pressure behavior during the 5.0 msec. period transient

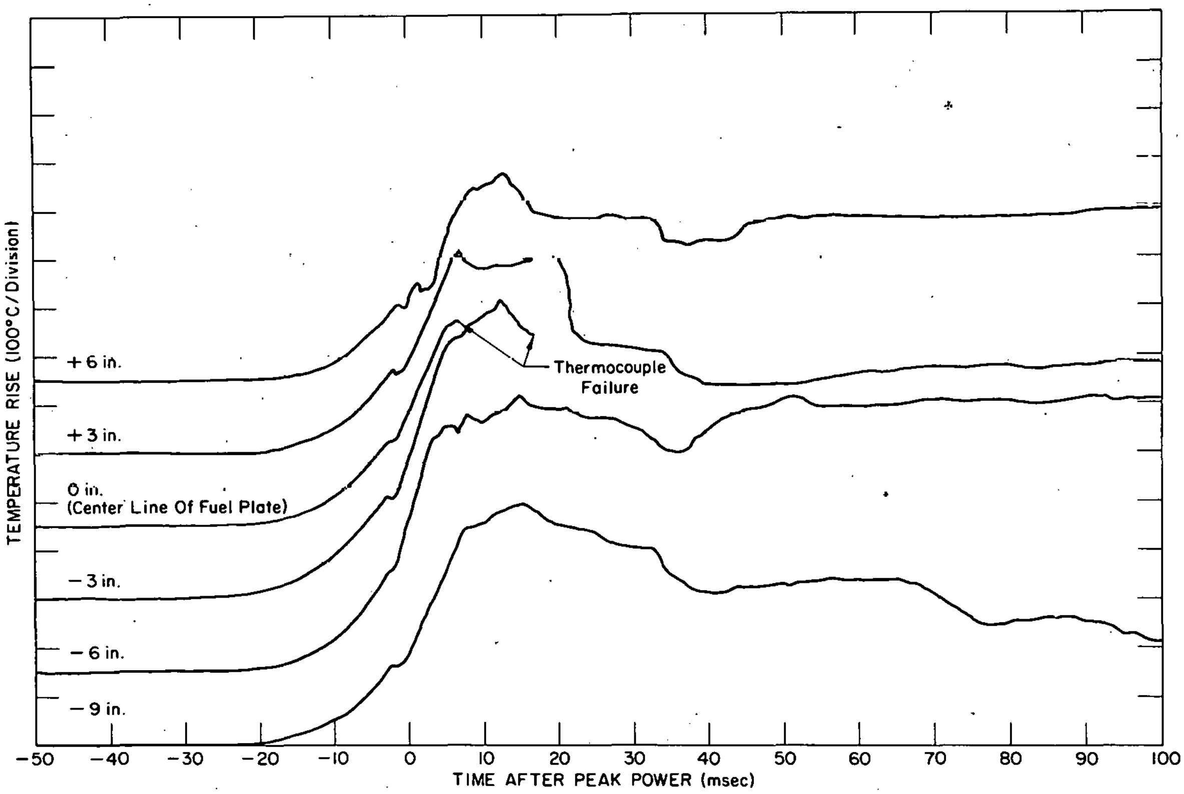


Fig. 33 Temperature behavior at 6 axial positions of fuel plate E-5-7 during the 5.0 msec period transient

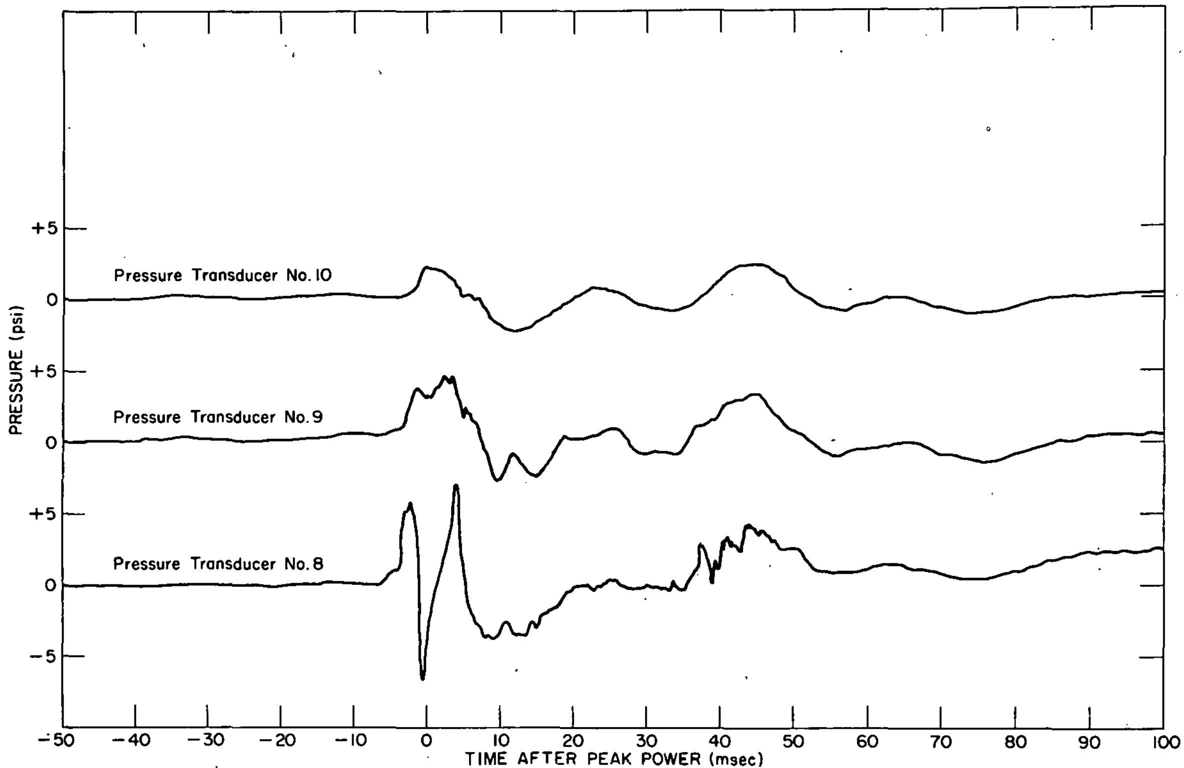


Fig. 34 Pressure behavior at three transducer positions during the 5.0-msec period transient

power level of about 1.5 Mw for the duration of the test. The post-peak power data below about the 20 Mw level requires some correction for the gamma sensitivity of the ion chambers. The energy and temperature during this power burst show a reasonably similar behavior to that of other tests. Examination of the temperature data indicates that the "break" from the initial exponential rise occurs about 100°C temperature rise. (This initial temperature break does not show up clearly in Fig. 32, but is quite obvious in the original oscillograph records; a typical break is shown in Fig. 26.) The break marks the time at which the initial phase of boiling first occurs, and, since the data shown in Fig. 32 are from approximately the hottest point in the core, this time ($t \approx -7$ msec) represents the first occurrence of boiling anywhere in the core.

A temperature drop (setback) occurs at about 200°C rise ($t \approx -3$ msec). By the time of peak power the boiling which causes this setback is expected to "blanket" the plate and possibly to void the entire channel around this location. Immediately following the temperature setback a rapid temperature rise takes place at a rate approaching $60^{\circ}C$/msec. During this time, the plate is insulated from significant heat loss and the maximum temperature rise reflects total energy release. Fluctuations in the temperature during the upper plateau of the temperature trace are the result of instabilities in film blanketing, i.e., momentary contacts of water with the fuel plate. Normally, blanketing is not as complete and long-lasting as is evidenced here by the very long, sustained high temperature. This appears to be the cause of the very high surface temperature reached in this transient. Both partial and complete film-breakdown are often indicated in other temperature data shortly after peak power.

The pressure data shown in Fig. 32 were provided by a transducer located about 3 in. away from the center of one side of the core (location No. 8 in Fig. 7). Initially, at $t \approx -6$ msec, a slight rise in pressure to about 2 psi occurs. This is tentatively regarded to be due to the initial-phase boiling which started at about this time. At $t \approx -3$ msec, a sudden rise to about 6 psi occurs which is probably due to the extensive production of steam occurring during the temperature setback.

A second pressure pulse is seen in Fig. 32 to occur about 5 msec after peak power. Although the boiling dynamics are not known at this time of the burst, the data reveal a consistent picture of events. The second pressure pulse indicates an increased ejection of moderator from the core, corresponding to the increased rate of power decline where the power "hump" terminates.

Both the temperature and the pressure curves shown in Fig. 32 are the result of "point" measurements. That is, they represent the physical conditions only at the instrumented position and, as such, do not fully represent the over-all picture. Fuel plates are commonly instrumented at several positions to obtain vertical temperature profiles such as are shown for the hottest plate (E-5-7) in Fig. 33. This plate was instrumented at the vertical centerline (0 in.) and at 3-in. increments to + 6 in. and -9 in.

A sudden drop of plate surface temperature at $t = 20$ msec can be seen on the + 3 in. thermocouple trace in Fig. 33. This drop in temperature is typical, and is the result of film collapse, which allows cold water to momentarily quench the fuel plate.

Thermocouple failures indicated in Fig. 33 for the - 3-in. and 0-in. thermocouples are probably the result of hydraulic forces acting on the thermocouple junction as water is thrust from the plate when steam is suddenly developed. Also, in this case, since the plate surface temperature actually reached the melting point, a possible weakening of the thermocouple bond may have contributed to failure. Other thermocouples, not shown here, are of an imbedded type and appear to be less subject to this type of failure.

Fig. 34 illustrates the positional dependence of pressure data. Transducer responses shown here are for positions 8, 9, and 10, respectively, about 3-in., 18-in., and 48-in. from the reactor as indicated in Fig. 7. The pressure traces indicate that the higher frequency components seen near the core are attenuated at greater distances.

Melting occurred on seven of the 270 fuel plates in the core as a result of the 5-msec period test. The melting was generally confined to a 6-in. high region about the centerline of the core, and occurred in about one percent of the total core fuel plate area. Fig. 35 shows a typical damaged fuel plate from fuel assembly E-5 after the test. The plate shown in Fig. 36 was also from the E-5 fuel assembly. This plate was instrumented with six surface-type thermocouples and the data obtained from these thermocouples have been shown in Fig. 33. The thermocouple designated in Fig. 33 as "-9 in." corresponds in Fig. 36 to the tape measure position of 6 in., the "-6-in." thermocouple corresponds to the 9 in. tape measure position, etc. Fig. 37 shows a close-up view of melting around the "-3- in." thermocouple. It is to be noted that the thermocouple positions do not coincide with the positions where the molten metal erupted; this is reflected by the measured peak temperature at this point which was several degrees below the melting point. Square-topped ripples were prevalent on most of the affected plates and, as demonstrated in Figs. 38 and 39, cladding failures occurred preferentially in the regions of greatest plate curvature. A large crack can be seen in Fig. 40 and many smaller fractures can be observed in Fig. 41. An indication of the nature and extent of rippling can be seen in

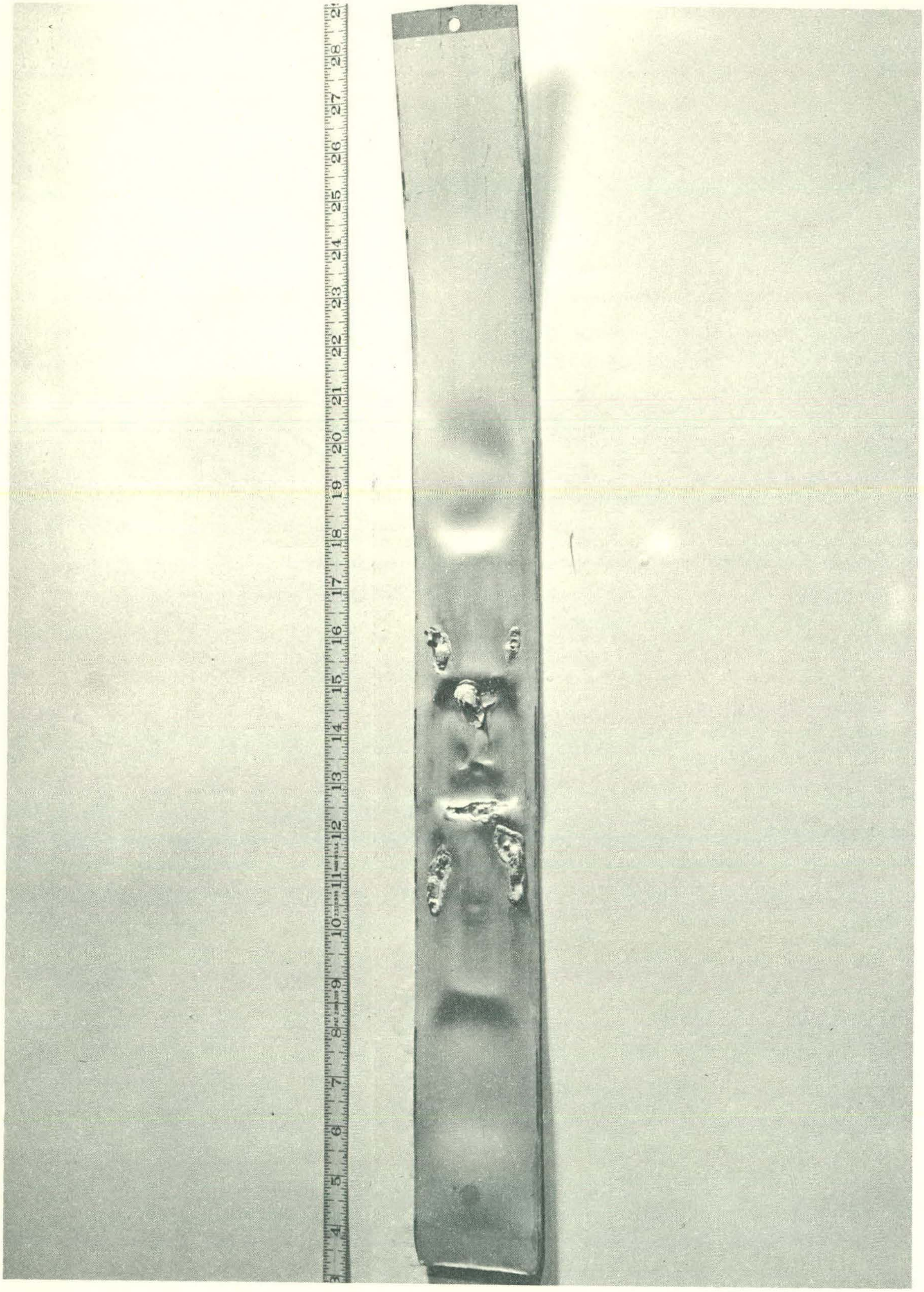


Fig. 35 Fuel plate damaged in 5 msec test

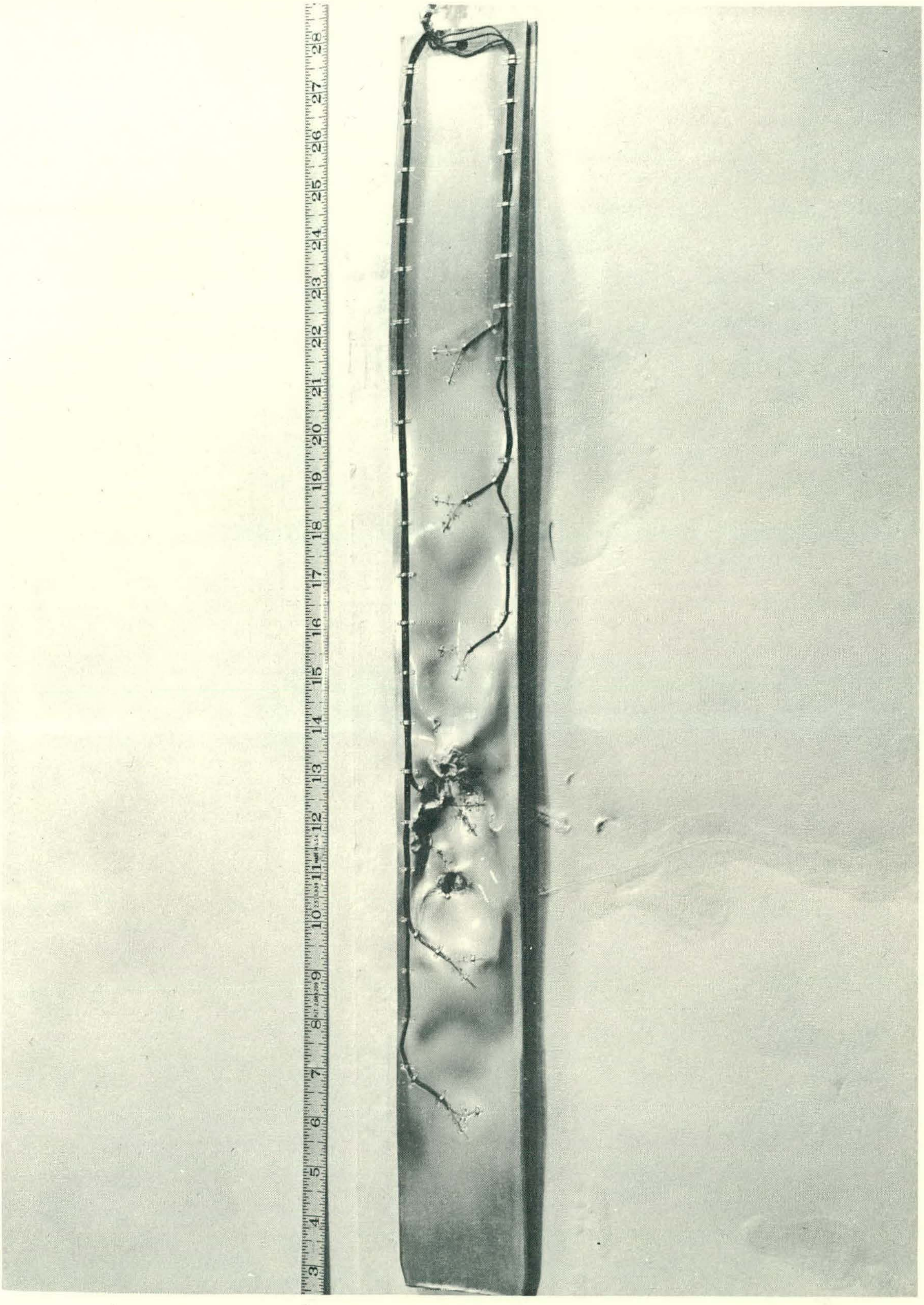


Fig. 36 Instrumented fuel plate damaged in 5 msec test
Two plates shown fused together

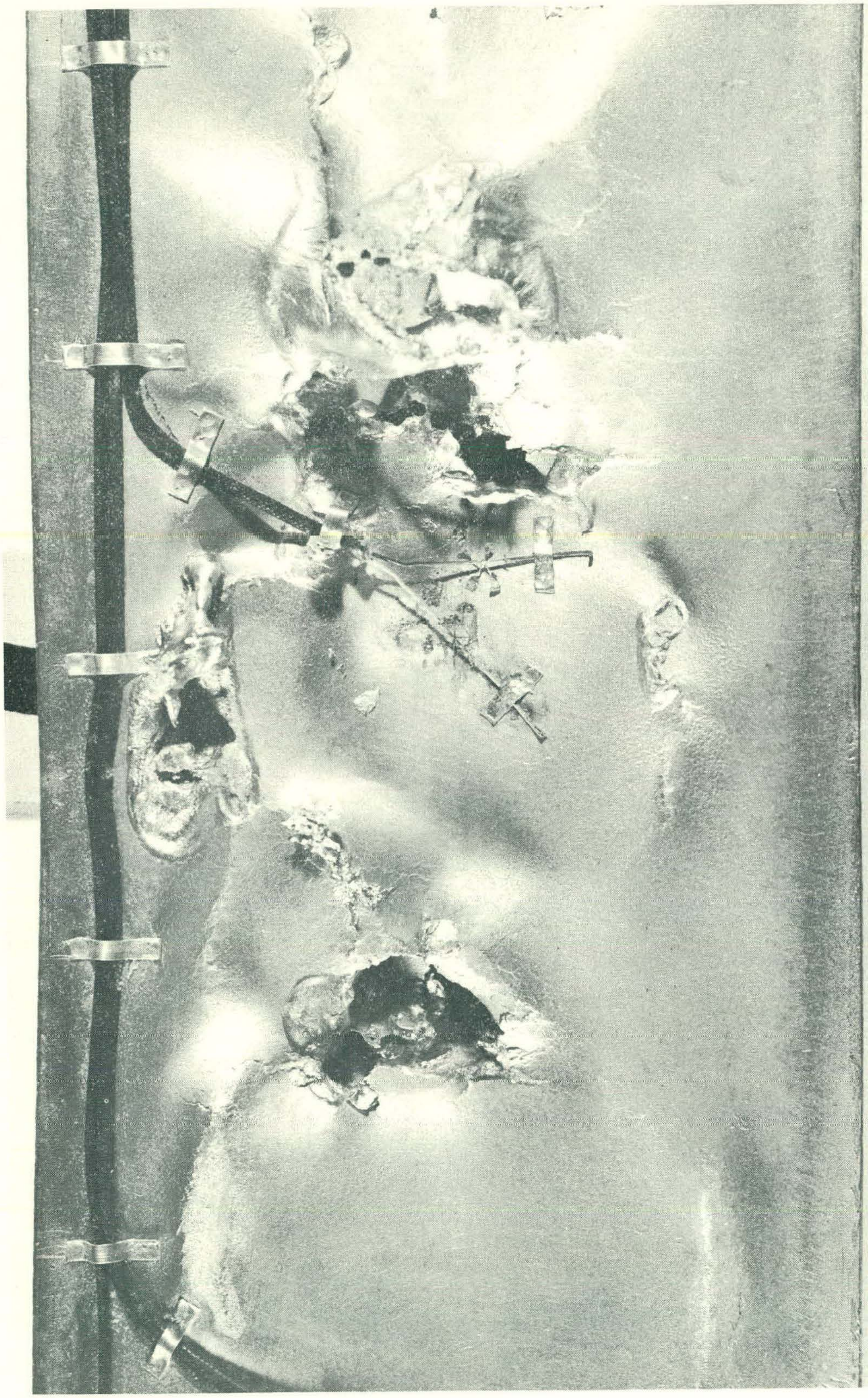


Fig. 37 Close-up view of thermocouple and melts shown in Fig. 36

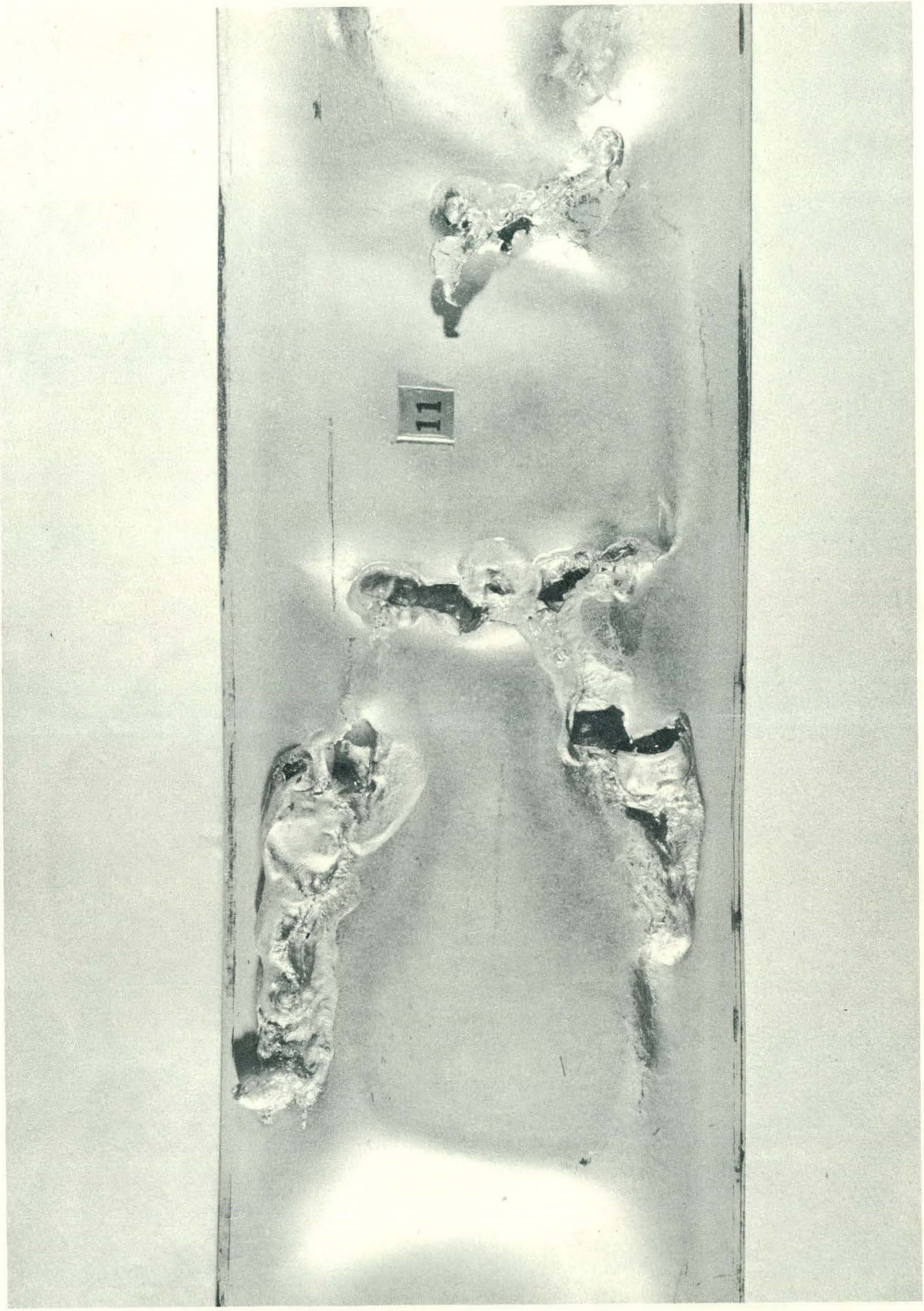


Fig. 38 Typical melt pattern obtained in 5 msec test

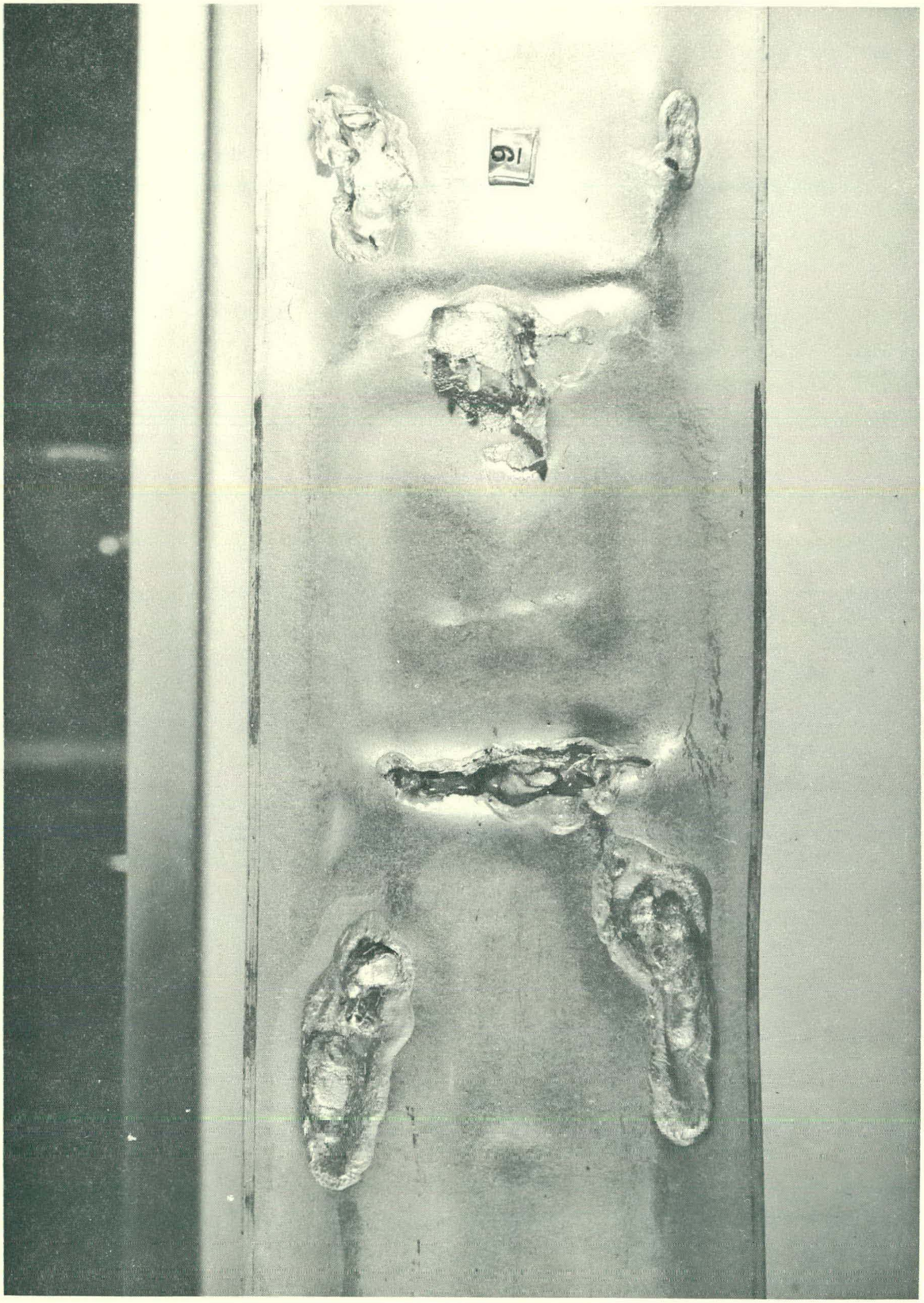


Fig. 39 Typical melt pattern obtained in 5 msec test

Fig. 40 Melts and fractures - 5 msec test

Fig. 41 Melts and fractures - 5 msec test

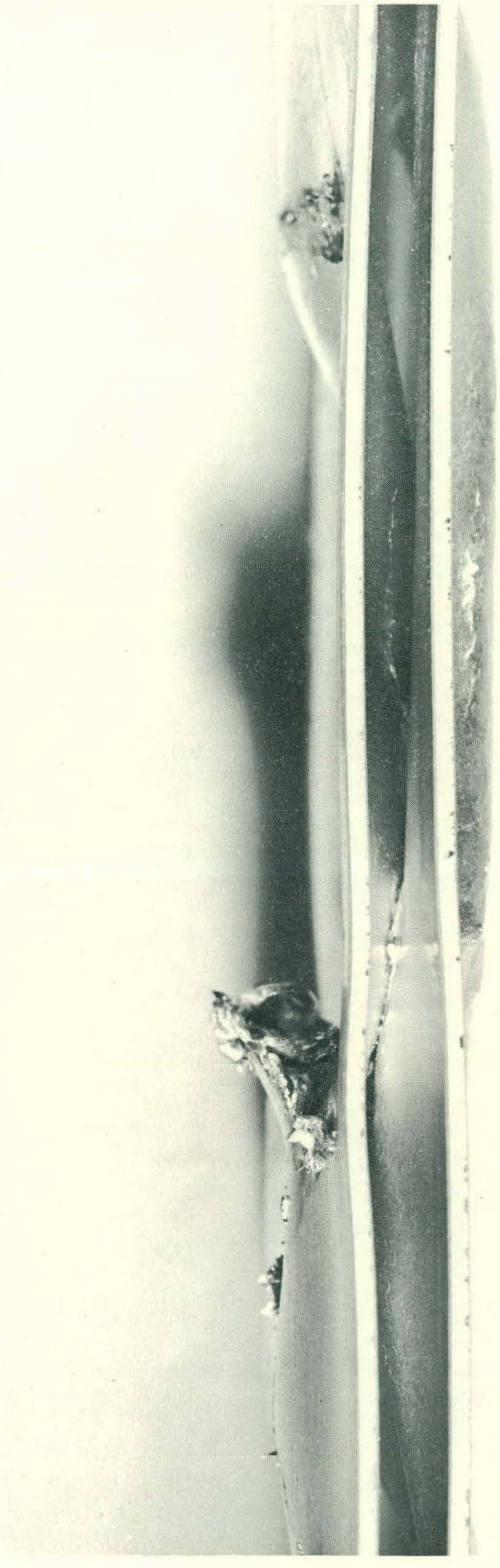


Fig. 42

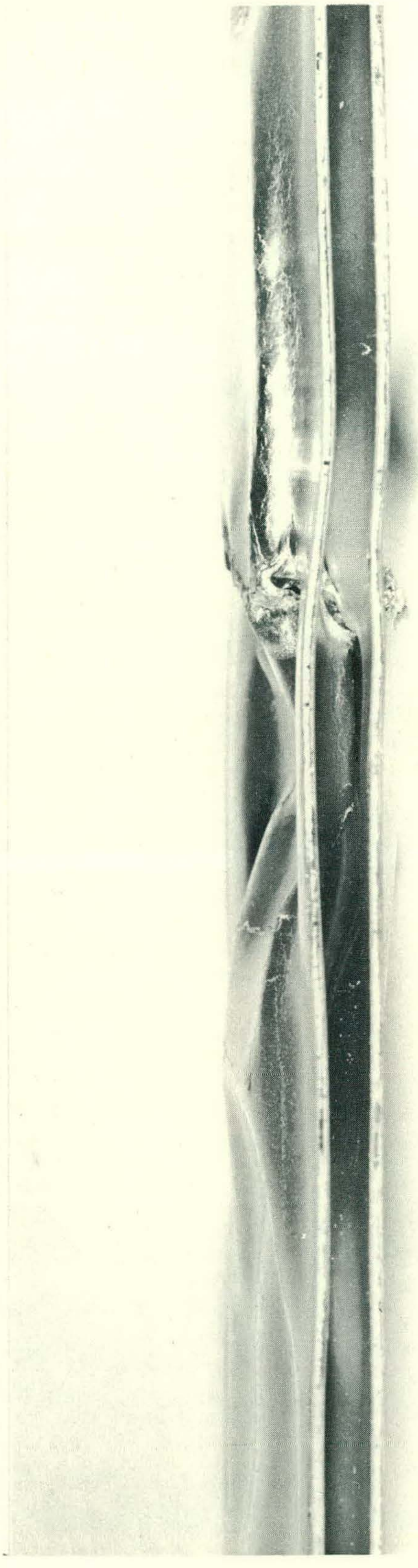
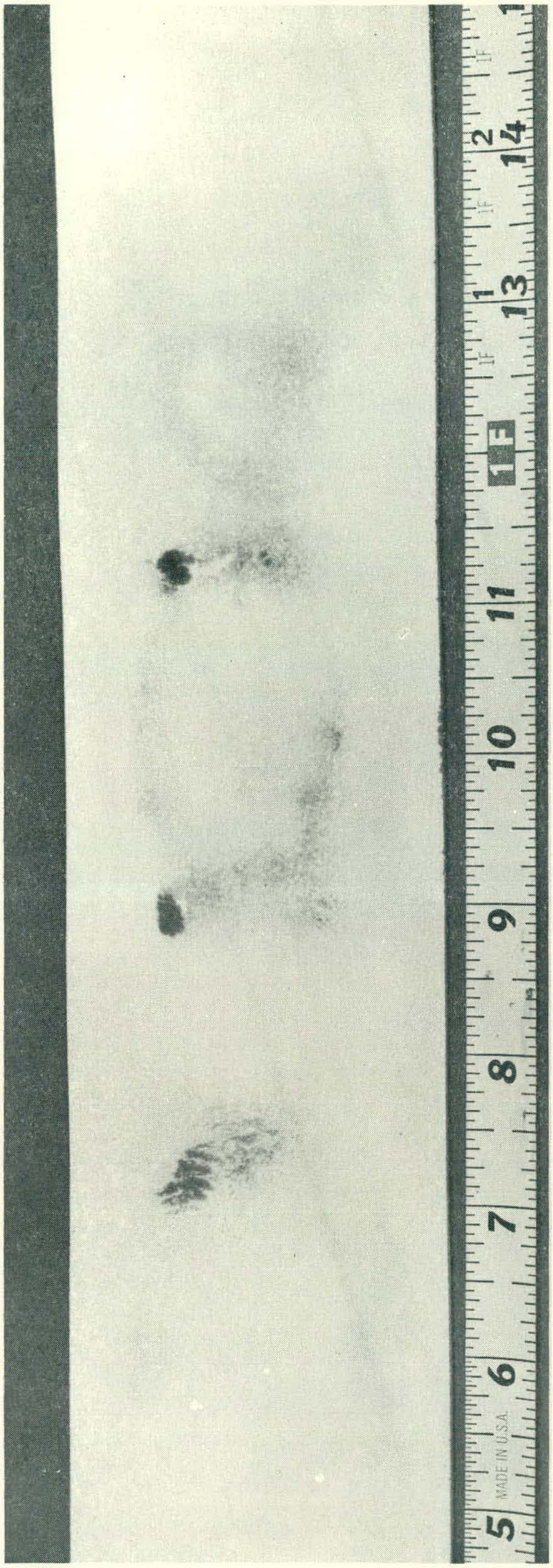


Fig. 43

Edge views of fused plates - 5 msec test

Fig. 44 Close-up view of melt
showing hole through fuel plate - 5 msec test



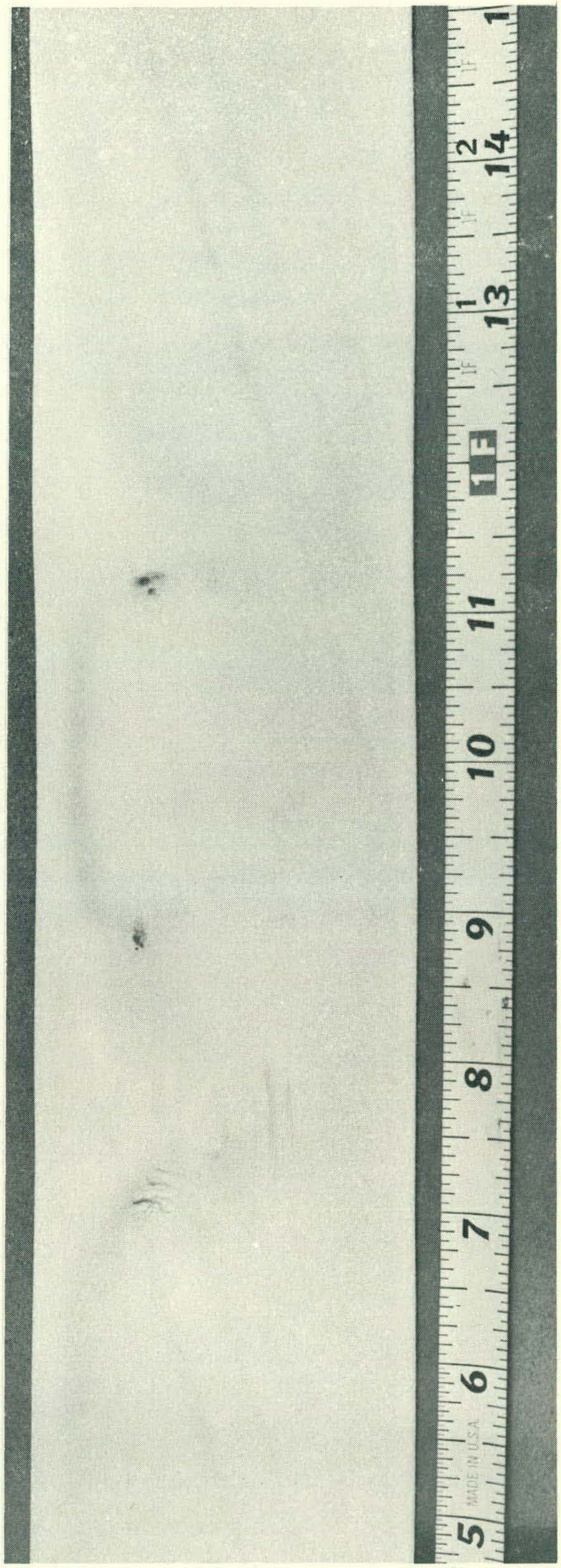


Fig. 45 Fig. 46

Dye-check patterns on unmelted fuel plates
Dark areas indicate fracture - 5 msec test

Figs. 42 and 43, where two plates have been fused together by the melting. A hole through the fuel plate is shown in Fig. 44.

In addition to visual inspection of the fuel plates, metallurgical examination is also being carried out on representative fuel plate sections to determine the extent of "hot cracking" of the cladding material and melting at grain boundaries. Several "dye checks" were made on plates which had no obvious cladding failure, and some of the results obtained are shown in Figs. 45 and 46. In these photographs the plates have been covered with a white "fix" coat and cladding failures show up as the dark regions. Closer examination reveals these dark areas to be composed of many small fractures, some of which penetrate to the fuel.

D. Extrapolation of Transient Test Data to a 2-msec Period Test

It is the primary purpose of this report to evaluate the safety of the proposed Spert I Destructive Test Program. The hazards analysis presented in the following section (V) of this report must rely for its basic assumptions on predictions based on extrapolation of the presently available data. During the course of the Fiducial-Transient and Exploratory Test Series for the DU-12/25 core and all previous transient tests at Spert, short extrapolations have been used very successfully in the prediction of peak power, maximum temperatures, $E(t_m)$, E(total), and other measurements. A modest accuracy of prediction by extrapolation has also been possible in the case of transient pressure data.

A degree of caution is required even in short extrapolations if the next test to be performed results in crossing the "threshold" for a new reactivity effect such as may arise when temperatures reach melting or vaporization points for the fuel plates, strain yield points of certain reactor components are exceeded, etc., if it is possible that the new effect cam materially change the dynamics of the system as a whole. Thus, in the case of the 5-msec period test, described above, wherein melting temperatures were predicted, it was felt that fuel melting might constitute a new mechanism for the generation of steam and therefore might result in a marked increase in the observed pressures. For this reason it was desirable to melt only a small fraction of the core as a first step and investigate any new trends, not only in pressure, but in all

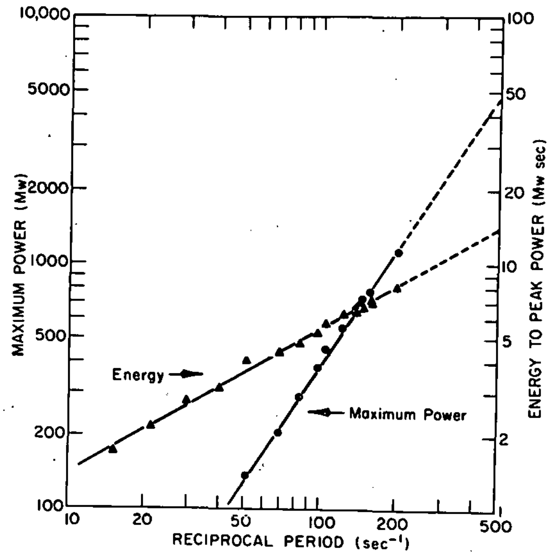


Fig. 47 Extrapolation plots of peak power and $E(t_m)$ for the DU-12/25 core

other parameters. However, it is now apparent that the "threshold" for severe pressure effects does not necessarily correspond to fuel plate meltdown. Continued testing of the DU-12/25 core to shorter periods and greater energy releases will be planned and executed with the same degree of caution and preparation. Insofar as it is possible to make inferences from longer extrapolations, this will be done now to provide an indication of the trends or "expected" results from testing in the region between 5-msec periods and 2-msec periods. As seen from the replots of DU-12/25 data, Figs. 47 and 48, both peak power and energy to the time of peak power, $E(t_m)$, appear to be well-behaved functions of the reciprocal period and

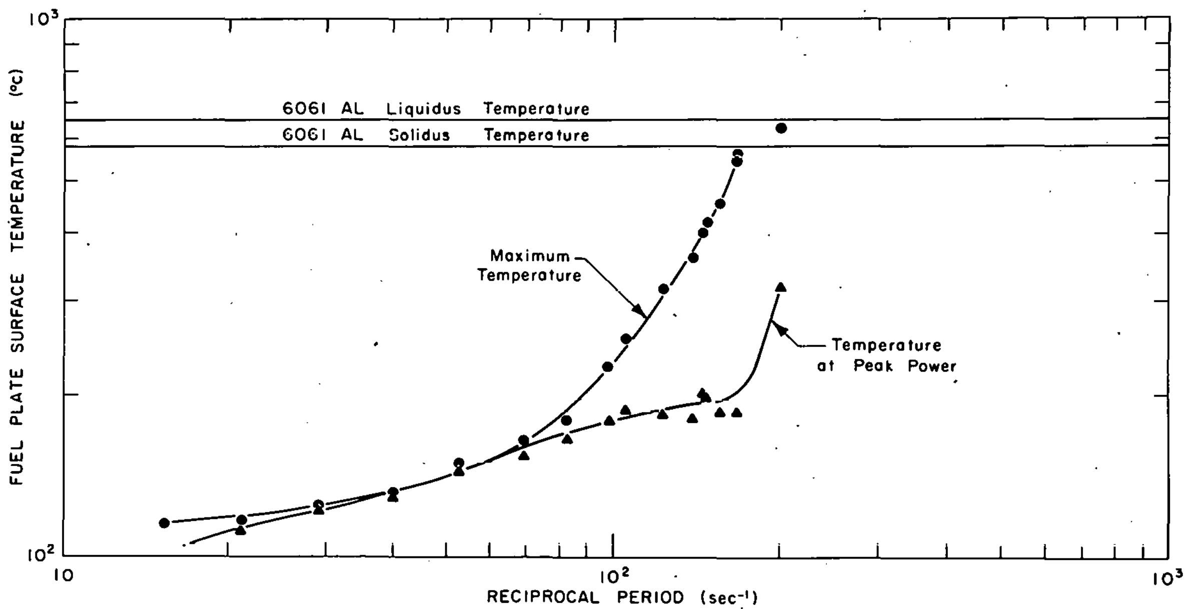


Fig. 48 Peak temperature and temperature at time of peak power - DU-12/25 core

straight-line extrapolations have been made on the logarithmic plots. At a period of 2 msec ($\alpha = 500$) these extrapolations yield an estimated peak power of about 5000 Mw and an energy of about 15 Mw-sec. Referring back to Fig. 27, an extrapolation of R, the ratio of E(total) to $E(t_m)$ yields a value of about 4.0 at $\alpha = 500$ sec^{-1}. Thus, on this basis, it appears that a 2-msec period transient would release about 60 Mw-sec. A 40 Mw-sec total burst energy release would be predicted at a period of 2.6 msec which can be compared with the estimated Borax-I energy release of 135 Mw-sec.

The maximum observed temperature of the fuel plate surface, Fig. 47, also appears to vary smoothly with reciprocal period except for the test ($\alpha = 200$) in which melting occurred and some heat begins to be absorbed in the heat of fusion of the materials. Since, as was observed earlier, maximum fuel plate temperature is nearly proportional to E(total), the same regularity would be expected for the internal energy contained in the fuel plate and on this basis, the enthalpy rise consistent with a 60 Mw-sec energy release becomes approximately 625 calories/gram, or, taking full account of a 94 cal/gm heat of fusion, this would imply a maximum temperature of about 2100°C which is just above the vaporization point of aluminum.

Present trends of the pressure data (see Fig. 31) indicate only modest peaks of less than 25 psi for a 2-msec period power excursion, although these particular extrapolations are probably the least valid, particularly if vaporization occurs.

Consideration of the reactivity behavior, Fig. 30, would indicate an increased tendency for "broadening" of the power burst shape as the bump which occurs after peak power becomes more prominent.

In evaluating the radiological hazard potential of a 2-msec period test (see Section V), the extrapolated total energy of 60 Mw-sec obtained here has been deliberately overestimated to be 200 Mw-sec in order to better account for uncertainties inherent in the process of extrapolation. In addition, since the extrapolated severity of pressures obtained in a 2-msec period test are not sufficient to produce appreciable disassembly or ejection of the core, pressure also has been deliberately overestimated for the hazards analysis by assuming a pressure sufficient to "lift" the entire core out of its earth shielding onto the surrounding terrain where shielding does not exist. It is, furthermore, assumed that a high degree of core dispersion occurs due to this pressure, and that vaporization of the fuel is extensive.

V. HAZARDS EVALUATION FOR A 2-MSEC PERIOD CORE DESTRUCTIVE TEST

A. Radiation Hazards

1. General

In this section an evaluation is presented of the radiation hazards to personnel as a result of a 2-msec period destructive test. The evaluation is based on a pessimistic set of assumptions regarding the power history and consequent fission product inventory in the test core immediately prior to the destructive test, the extent of fission product release and the prevailing weather conditions at the time of the test. The results of the analysis indicate that with the procedural controls (Appendix B) which are normally exercised in the conduct of any transient test at Spert, no significant hazard to personnel or the general public will be sustained in the execution of destructive tests. All nuclear operation is conducted remotely 1/2 mile from the reactor site.

2. Direct Gamma Radiation Exposure

As a basis for estimating the fission product inventory in the core just prior to a 2-msec period destructive test, it is assumed that during the previous six months the core had been subjected to 100 transients, each resulting in an energy release of 20 Mw-sec for a total fission energy release of 2000 Mw-sec. (This is roughly a factor of 4 greater than the nvt which would be expected were the same complement of fuel assemblies in the core to be used throughout the test series; actually, the replacement of damaged fuel assemblies following each transient test may be expected to result in a substantially lower fission product inventory than that postulated.) The series of 100 transients is then assumed to be followed by the 2-msec period destructive test burst, which, for the purpose of this analysis, is assumed to release 200 Mw-sec and result in the total destruction of the core. The 200 Mw-sec value is substantially higher than the 135 Mw-sec energy obtained in the Borax test and is at least a factor of two greater than the expected energy release in the Spert test (see Section IV). It is then assumed that all of the fuel plates in the core are melted and that there is a subsequent maximum release to the atmosphere of 1% of the contained solid fission products, 50% of the halide fission products, and 100% of the inert gaseous fission products. This amount of fission product release in the various categories is based upon the recommendations given in 10 CFR-100[15], the guide setforth by the AEC for evaluation of proposed reactor sites.

The gross fission product activity of the core following the final transient is shown, as a function of time, in Fig. 49. The transients prior to the final transient will have resulted in approximately 200 curies of relatively long-lived gross fission products in the core. This long-lived activity level will be relatively low compared to the high-level activity resulting from the final burst for a day following the test. After about a week, however, the reverse situation will be true, with the fission product activity from the earlier transients providing the major contribution to the radioactivity. In estimating the direct gamma radiation dosage to personnel at the Control Center building, the assumption is made that the entire core is lifted from the vessel and set down above ground level. The average gamma energy per disintegration has been taken to be equal to 0.7 Mev; the dosage conversion factor equal to 2×10^{-3} (mr/hr)/(Mev/cm^2); and the attenuation length in air equal

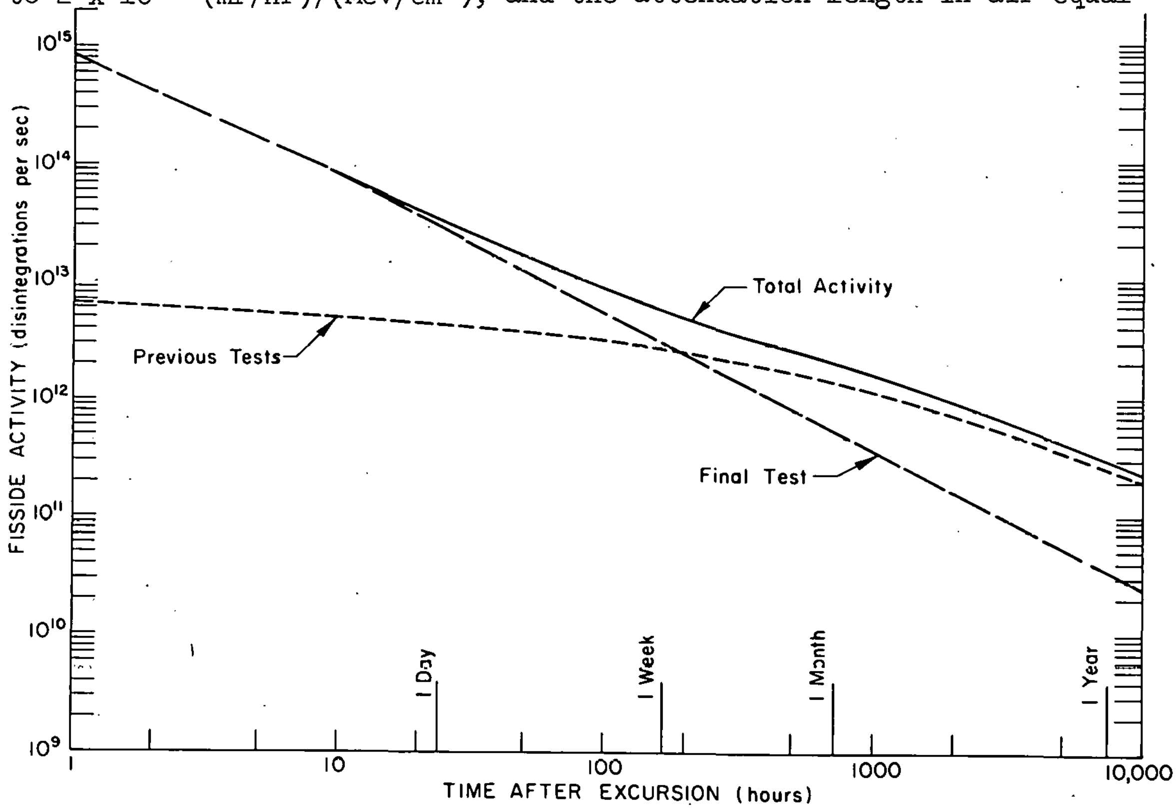


Fig. 49 Assumed gross fisside activity following a power excursion releasing 200 Mw-sec energy

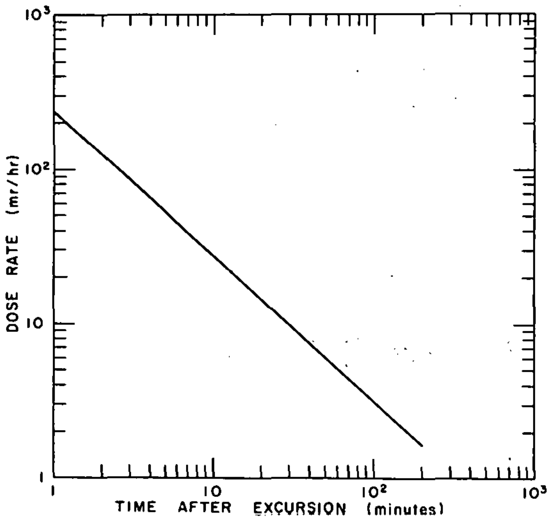


Fig. 50 Estimated direct gamma dose rate at the Spert Control Center building following a 200 Mw-sec destructive power excursion

to 270 meters. For these conditions, the direct gamma dose rate at the Control Center building (1/2 mile away) has been calculated as a function of time after the destructive burst, and the results are shown in Fig. 50 . The 3-hour integrated dose at 1/2 mile is 48 mr; at 1 mile it is 0.6 mr; and at Atomic City, the nearest population area (8 miles away), the dose is negligible.

3. Local Exposure, Radioactive Cloud and Fallout

Locations downwind from the reactor building will be exposed not only to the direct gamma radiation from the core but also to dosage from the radioactive cloud passing by, and to the dosage from the particulate matter settling out on the ground. All tests in the destructive series will be conducted under strict meteorological control: the winds at the site at the time of the tests coming from the southwest (between 190° and 250°), at a minimum speed of about 5 mph (for the final test, between 5 and 15 mph).

For the present purposes, it is assumed, however, that inversion weather conditions prevail at the time of the test and that the winds are blowing directly toward the Spert Control Center building at a speed of 3 mph. Under these conditions, and using the calculational methods outlined in Ref. 16 , the local dosage at the Control Center building due to exposure to the radioactive cloud is calculated to be 300 mr; at a distance of 1 mile, the exposure is 40 mr. The initial 3-hour integrated dosage from fallout is 170 mr at 1/2 mile and 50 mr at 1 mile. The total integrated external gamma dosage for the first three hours after the test is given in Table IV. An interval of three hours was selected for consideration because this is estimated to be the maximum time required for any one person to remain in the vicinity following the final transient. It is noted that the maximum downwind external gamma dosage at 1/2 mile is not substantially higher than the recommended maximum weekly exposure(17).

TABLE IV

External Gamma Dosage Integrated Over a Three Hour Interval Following the Final Transient

Distance (Miles)	Downwind Exposure mr	Exposure in Other Directions (mr)
0.5	350	48
1	88	0.6
8 (Atomic City)	< 0.9	0

The ingested dosage to personnel is computed on the basis of halide ingestion, in accordance with Ref. 15 . The total thyroid dosage from inhalation during the passage of the radioactive cloud is 13 rem at 1/2 mile and 3.9 rem at 1 mile in the downwind direction. There would be no thyroid exposure for personnel upwind of the facility. These doses are to be compared with a 300 rem maximum permissible exposure(17).

The most important long-lived activities in the fallout will be those of strontium, yttrium, and cesium. Table V indicates the fallout densities of these nuclides at distances of 1/2 mile and 1 mile as computed by the methods of Ref. 16 . In Ref. 18 it is recommended that

TABLE V

Heavy Metal Fission Product Fallout from 200 Mw-sec. Excursion ($\mu\mu c/cm^2$)

Distance (miles)	Isotope Sr^{89}	Sr^{90}	Y^{91}	Cs^{137}
1/2	160	1	160	1
1	40	0.3	40	0.3
8	0.4	.003	0.4	.003

the maximum permissible concentration (MPC) of the above cumulative fission products be 1 μc per kg of calcium contained in the top 2.5 inches of soil. Using this as the MPC, the fallout at 1/2 mile would be about 0.5% of the MPC at 1/2 mile. Since the closest land used for production of commercial agricultural products is about 8 miles from the Spert I facility, the calculated fallout as a result of the destructive burst is expected to be negligible at such areas.

4. Conclusion

The results of this pessimistic evaluation of the radiation hazards associated with the performance of core destructive tests indicate that the radiation hazards are not significant and that under the worst conditions ample time would be available for orderly evacuation of all personnel in the vicinity of the test area. During the preliminary series of partial core destructive tests leading to the final destructive test, radiation and fission product contamination measurements will provide data for a more realistic evaluation of the radiation hazards problem expected in the final test.

B. Missile Damage Potential

An evaluation of the possibility of damage to the Spert II and III facilities or to the Control Center by missiles from Spert I destructive tests must draw heavily on the experience of Borax and SL-1. In the case of Borax, no missiles or other debris were thrown more than 200 ft and the major components were found within a few feet of the reactor. This is to be compared with the approximate 1/2 mile isolation distance involved at Spert. While in the case of SL-1 some items did penetrate the ceiling of the reactor room, none escaped the building.

In the extremely unlikely event of missiles traveling 1/2 mile, the buildings at Spert II and III and Control Center should provide adequate protection. Should a missile penetrate the concrete block of the Spert II or III reactor buildings, no damage which would seriously disable the plants is envisioned.

During the tests no personnel will be permitted to be outside the Control Center building, unless necessary to the performance of the test or the acquisition of data, and in such case shelter protection will be provided and hard hats required.

C. Control of Personnel During Destructive Tests

During the final preparations for performance of a destructive test in which all or a major portion of the core is expected to be damaged, access to the Spert Control Center area will be restricted to personnel whose presence is necessary to the test operation. All other Spert personnel in the Spert area will be located at the Spert IV reactor area under the supervision of a designated evacuation warden and adequate bus transportation will be standing by. Non-Spert personnel will be admitted to the Spert IV area by special permission and will be under escort of a Spert liaison representative. A limited number of non-Spert personnel may be admitted to the Control Center area. During the actual performance of tests in the destructive series a road block will be set up on the Spert access road, several hundred feet from the Control Center, to prevent access to the Spert area.

The following personnel will be present in the Spert I control room during the performance of tests in the destructive series:

Supervisor-in-Charge of Test Operation
Senior Operator-in-Charge
Operator-on-Console
Instrumentation Engineer
Electronics Technicians (3)
Supervisor-in-Charge of Experimental Program
Group Leader for Experimental Program.

Approximately two cameramen will probably be required to operate cameras in the Control Center area outside of the building. One photographic liaison man may be required in the control room during the performance of the tests.

The presence of certain personnel will be required in the adjacent Spert II control room during the performance of these tests. The required personnel are as follows:

Reactor Projects Senior Staff Personnel
Engineer-in-Charge of Spert I Plant
Spert Health Physics Supervisor
IDO Site Survey Liaison Representative.

The following personnel may be present in the Spert II control room as required by the nature of the test and their duties:

Other members of Destructive Test Group

Other personnel having responsibility for portions of the test

Selected non-Spert personnel.

The presence of the following personnel will be required in the Control Center building during the performance of tests in the destructive series:

Reactor Technicians (two)

Health Physicists (two)

Dark-Room Technician

Instrumentation Engineer for Magnetic Tape System

Electronics Technician for Magnetic Tape System

Evacuation Bus Driver

Electrical Engineer for Communications Equipment

Electronics Technician (standby).

It is estimated that approximately 30 personnel will be required in the Control Center area for each test. A maximum of about 40 persons will be permitted in the Control Center area during the tests.

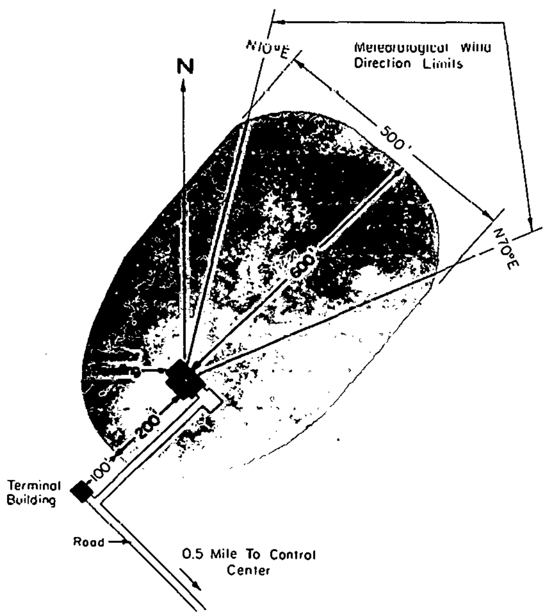


Fig. 51 Approximate area to be cleared around reactor building

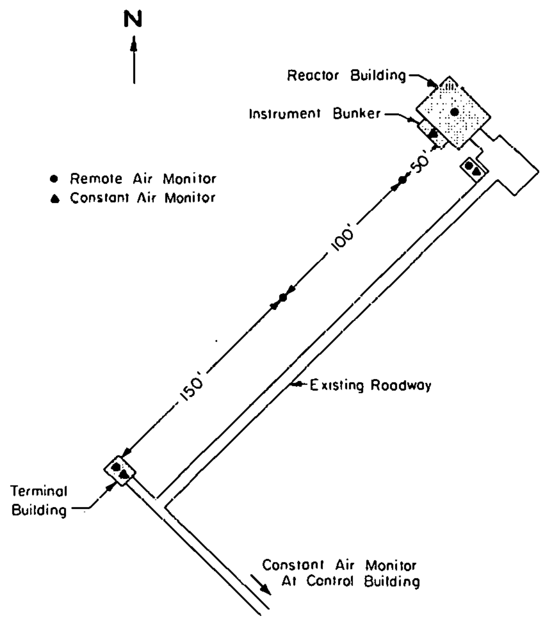


Fig. 52 Location of radiation monitors during destructive tests

VI. RE-ENTRY AND CLEANUP OPERATIONS

A. General

All re-entry and cleanup operations at the reactor site will be carried out in accordance with the Health Physics rules pertaining to the control of radiological hazards that are set forth in Section 6.100 and 6.200 of the Spert Standard Practices Manual. Operation in contaminated or radiation-field areas will be subject to Health Physics approval at all times.

Visibility and access to the area will be good since the sheet metal sides and roof of the reactor building will be removed for each test in the destructive series. An area surrounding the Spert I reactor building will have been bulldozed clean of sagebrush and leveled prior to the test, to permit easier identification and recovery of debris and easier access with mobile equipment (see Fig. 51). If radiation levels require its use, a shielded, self-propelled hydraulic crane ("cherry-picker") will be available for re-entry, to permit survey photography of the area and debris, and determination of high radiation sources by means of gamma detectors.

B. Re-Entry Procedures

As soon after the test as possible, consistent with the monitored progress of any fission product cloud, a special mobile Health Physis trailer will be set up on the Spert I access road at the closest, convenient point to the reactor area, commensurable with radiation levels. This trailer will contain re-entry supplies and personnel decontamination facilities. A second contamination check point will be located near the guard house, adjacent to the Control Center building, to ensure thorough decontamination of personnel leaving the reactor site.

It is expected that initial re-entry operations after the destructive test will be carried out by two-man, emergency monitoring teams comprised of Spert Health Physics personnel.

Prior to entry of the reactor area by Health Physicists and technical personnel, briefing discussions will be held covering the specific operations to be performed, radiation levels to be expected, supervisory control, transportation, alarm systems, etc.

Following a destructive transient, the remote area monitoring meters will be checked to determine the radiation levels at the various locations in the reactor area. When the radiation field drops to a safe point to permit re-entry without excessive exposure (accumulative dose of 3 rem per quarter) to personnel, the first monitoring team will enter the area. This team will be clothed in plastic suits, which completely cover the body and accommodate air-tank breathing apparatus. They will be equipped with portable radiation monitoring instruments and portable radio units. Upon entering the area, they will survey for radioactive debris and for direct radiation coming from the reactor vessel. This information will be transferred back to the Control Center by radio and plotted on a special map containing pre-located identification points. If required, based on information obtained by the first team, a second team will continue the survey of the contaminated area, while the first team proceeds to the mobile "hot exchange" trailer. This will have been located, if possible, adjacent to the terminal building, where an adequate water supply is available. Following clothing change and any necessary decontamination, the first team will report to the Control Center. Additional Health Physics personnel will be available to accompany Spert technical personnel involved in the initial operations of removing film from cameras located in the reactor area and removing selected foils and activation monitoring packs.

In addition to the environmental monitoring coverage (air sampling, film badge radiation exposure, fall out, wind pattern, soil fisside content, etc.) provided by the IDO Health and Safety Branch, Phillips Health Physics personnel will also obtain dosage measurements, using radiation accident dosimeters, constant air monitors, and remote area monitors located in the immediate area surrounding the reactor building (see Fig. 52). The area monitors will be connected to remote meters and a strip chart for read-out purposes and to obtain permanent records of various radiation levels as functions of time. All monitors will be supplied by emergency power in the event of power failure at the Control Center building.

C. Cleanup Operations

Although the specific actions involved in the cleanup operation cannot be given, since they will depend on the nature and extent of destruction and the problems encountered upon re-entry, the general procedure to be followed may be indicated.

Prior to actual cleanup and removal operations, reasonably complete written descriptions of the conditions of the reactors, building, nature and location of debris, etc., will be made, supplemented wherever possible by photographic data. It is recognized that a major problem in the cleanup and removal of components is expected to be the high radiation levels existing in and around the reactor building, necessitating, at all times, the application of Health Physics procedures. Following the initial general survey of radiation sources by Health Physics, a complete photographic and radiation survey of the reactor building and area will be made. This will not only provide information relative to the destruction but will provide a guide for further operation. This survey will be accomplished by movie cameras mounted on the self-propelled crane and by still pictures made upon physical re-entry of personnel. A detailed radiation survey will then be made, using collimated ion chambers mounted on the extension boom of the crane in conjunction with movie cameras, to provide simultaneous photographic and radiation data of various debris and other high-radiation sources. Pinhole-camera photography may be used to define localized gamma radiation sources. Where radiation levels permit, Health Physics instruments will be used to complete the survey.

Upon completion of the photographic and radiation surveys, cleanup operations will be initiated with removal of the highly radioactive objects by means of the self-propelled crane. Depending upon the objects to be removed, the crane will lift the objects from the reactor building using a hook device or an electromagnet suspended from the boom of the crane. High level contamination will be removed by a vacuum system operated by a manipulator. A closed-circuit television system with the camera fixed to the boom and pre-focused will permit remote visual inspection for these operations. These objects will be placed on tarpaulins or in buckets, depending on size, and removed to an area of relatively low radiation level for decontamination and closer inspection. After inspection these objects will be placed in a temporary burial ground, pending future decisions concerning their disposal. It is proposed to transport pieces of debris and major components of the reactor in shielded containers to available hot shop facilities at the NRTS for detailed visual inspection, disassembly, and metallurgical examination. Removal of low-level radioactive objects can be done manually, following vacuuming to remove the low-level contamination.

Further decontamination of materials and area will be accomplished by scrubbing with detergents and/cr various decontamination solutions, steaming, or sandblasting, followed by drying and vacuum cleaning. The water and/or solvents used in the decontamination processes will be carried off to a properly marked leach pond.

Objects having no further usc will be sealed off in suitable containers or sprayed with a fix-coat and removed to the temporary burial ground pending removal to the NRTS Burial Ground. Objects which can be made further use of will be decontaminated. If their level of contamination is above the limits of the low level decontamination facilities, they will be placed in suitable shielded containers and removed to the NRTS Chemical Processing Plant for decontamination.

Estimates of the time required for complete cleanup and reactivation of the area following a full scale destructive test are difficult to make until the extent of damage is known, but it is expected by reference to the Borax test that the facility can be placed in operation in at least six months after such a test. The major delay which is foreseen would be replacement of the reactor tank, if the present one is ruptured.

VII. CONCLUSION

The Destructive Test Program represents an important phase of the Spert Experimental Reactor Safety Program. As described in this report, the destructive tests will be conducted in accordance with the standard Spert procedural controls for reactor testing and the special controls established for the Destructive Program. All nuclear operation is conducted remotely, with Spert personnel withdrawn about 1/2 mile from the test site. An analysis of the radiation hazards involved in a 2-msec period violent destructive test indicates that the total 3 hour integrated dose at the Spert I control center would not exceed 350 mrem, which is approximately the maximum permissible dose for one week. This analysis is based on assumptions of very unfavorable weather conditions and of a significant core fission product inventory prior to the destructive test. The Spert destructive tests, however, will be conducted under strict meteorological control, using essentially fresh cores, as a result of the continued replacement of previously damaged fuel assemblies. This analysis, therefore, constitutes an over-estimate of the dosage to be expected as a result of the test.

As a result of these considerations and the results of the Borax-I destructive test, which demonstrated the feasibility of core destructive testing under similar conditions, it is concluded that the Spert I Destructive Test Program can be conducted in a fashion consistent with the general policy of the AEC to protect government and contractor personnel and the general public against undue exposure to radiation and against all other potential health and safety hazards which may arise in the execution of nuclear activities.

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APPENDIX A - DETAILED DESCRIPTION OF SPERT I REACTOR FACILITY

by

P. L. Mahoney and D. R. Barton

I. Plant Site and Buildings

1. Spert Site

The Spert site is located within the boundaries of the National Reactor Testing Station (NRTS), approximately 50 miles west of Idaho Falls, Idaho. The location of the site with respect to other NRTS installations is shown in Fig. A-1.

A general plan of the Spert site is shown in Fig. A-2. The reactor areas have been arranged in a semicircle of approximately 1/2-mile radius from the Control Center and about 1/2 mile from each other. Spert I is approximately northwest of the Control Center, with the other three reactor areas spaced at approximately 60° increments clockwise. The entire site is enclosed by a three-strand barbed wire perimeter fence approximately 1/2 mile, but no closer than 1/3 mile, from the nearest reactor facility.

2. Control Center Area

The Control Center area forms the center of the Spert operations. Within the 250- x 250-ft fenced area are the Control Center building and the raw water storage and distribution for the Spert site. The Control Center building houses offices and laboratories, a darkroom, instrument and mechanical work areas, and the reactor controls and instrumentation for all of the Spert reactors.

Water for the Spert site is supplied from two wells located near the Control Center area. Well 1 is 653-ft deep and Well 2 is 1217-ft deep. A 400-gpm deep-well pump on Well 1 and a 550-gpm deep-well pump on Well 2 supply water to the two ground-level storage tanks. A total capacity of 75,000 gal of ground-level storage is available. An automatic level control maintains the tank levels by intermittent operation of the pumps.

NATIONAL REACTOR TESTING STATION

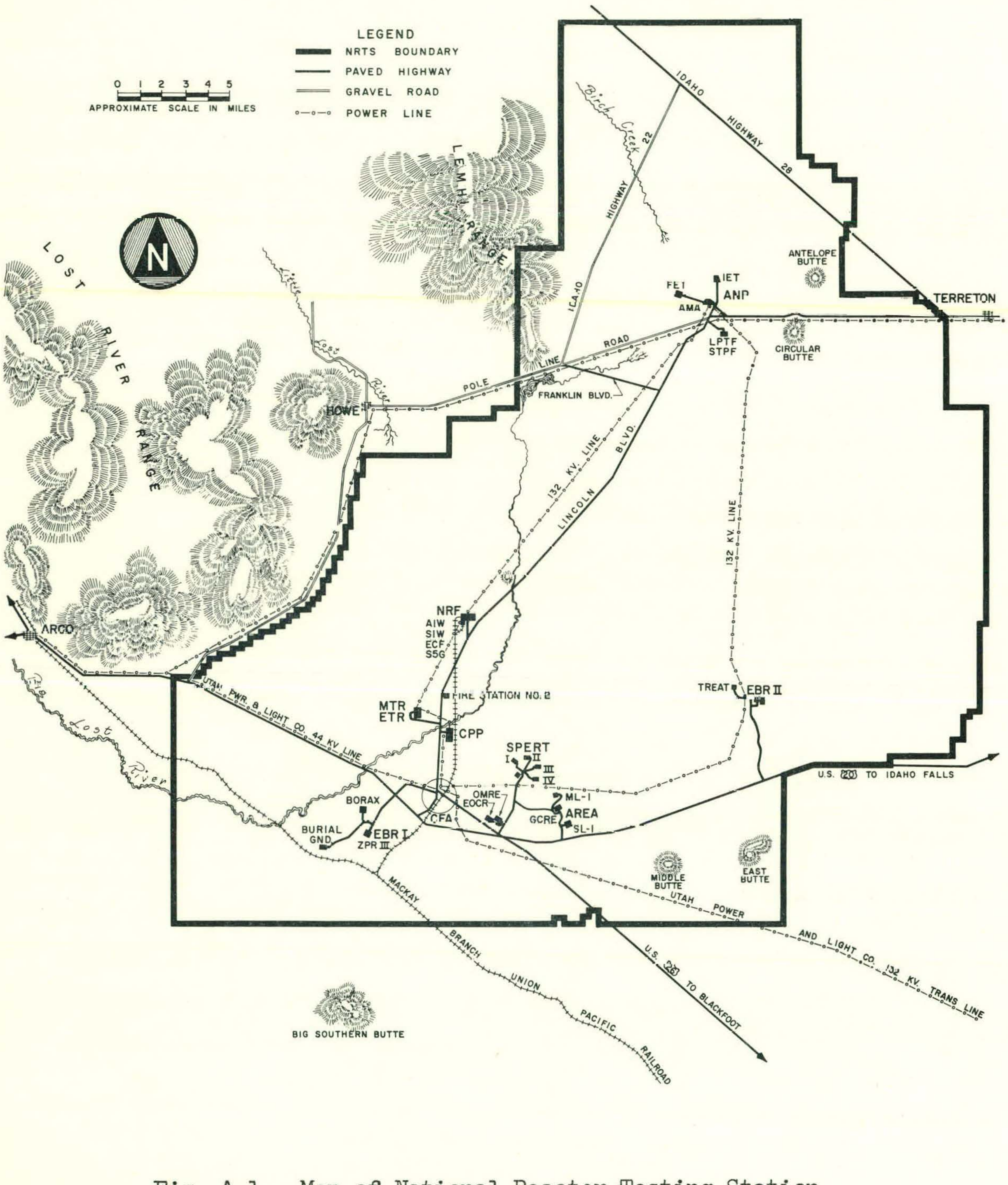

Fig. A-1 - Map of National Reactor Testing Station

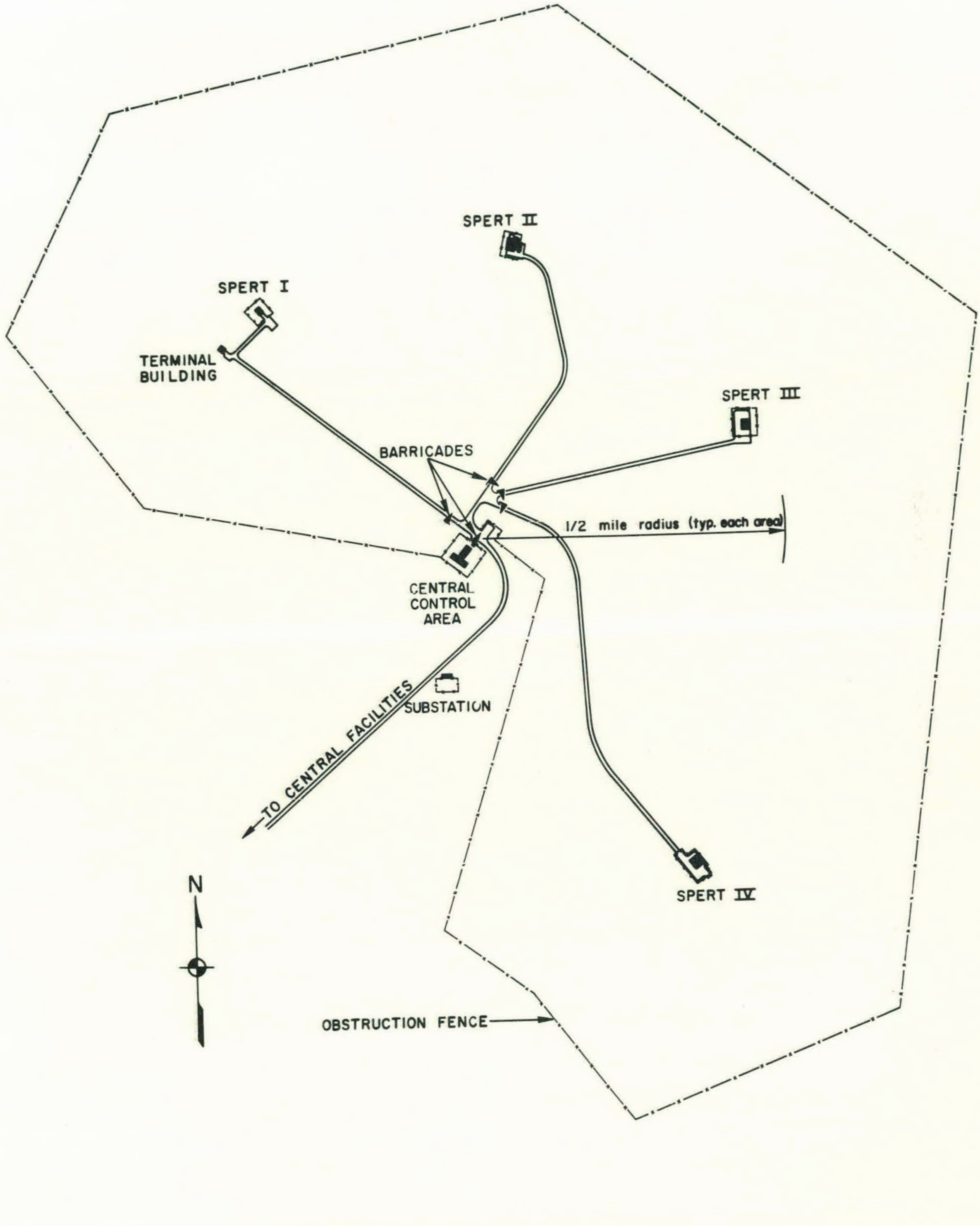

Fig. A-2 - Spert Site Plan

Water is distributed to all areas by two 400-gpm booster pumps which, in conjunction with a pressure control valve, maintain a line pressure of about 72 psi. A 750-gpm water pump supplies extra capacity if the water demand exceeds the capacity of the booster pumps.

Electrical power is supplied to the Control Center area and reactor areas from 13.8-kv feeders located at the Spert substation. Power to the substation is obtained from the 132-kv NRTS distribution loop.

3. Terminal Building

In the original construction of the Spert I facility, provision was made for a series of Spert I reactors placed in pits and spaced generally in the Spert I area. (To date, only the Spert-I facility has been constructed in the Spert-I area.) A single utility building, the Terminal Building, serves as a support facility by providing demineralized water, compressed air, and change-room facilities for the Spert-I area.

The Terminal Building is a prefabricated galvanized iron structure located approximately 2800 ft from the Control Center and 400 ft from the Spert I reactor building. An external photograph of the building is shown in Fig. A-3 and the floor plan is shown in Fig. A-4. The Terminal Building is rectangular shaped, 20-ft wide, 30-ft long and 14-ft high. The building houses the service facilities for the Spert I reactor, including a 10-gpm water softener and mixed-bed deionizer, a 1000-gal deionized water storage tank, a 120-gpm reactor fill pump, a 110-psig compressed-air system, and a personnel decontamination and change room.

A substation located near the terminal building provides electrical power for the Spert-I area.

Fig. A-3 - Spert I Terminal Building

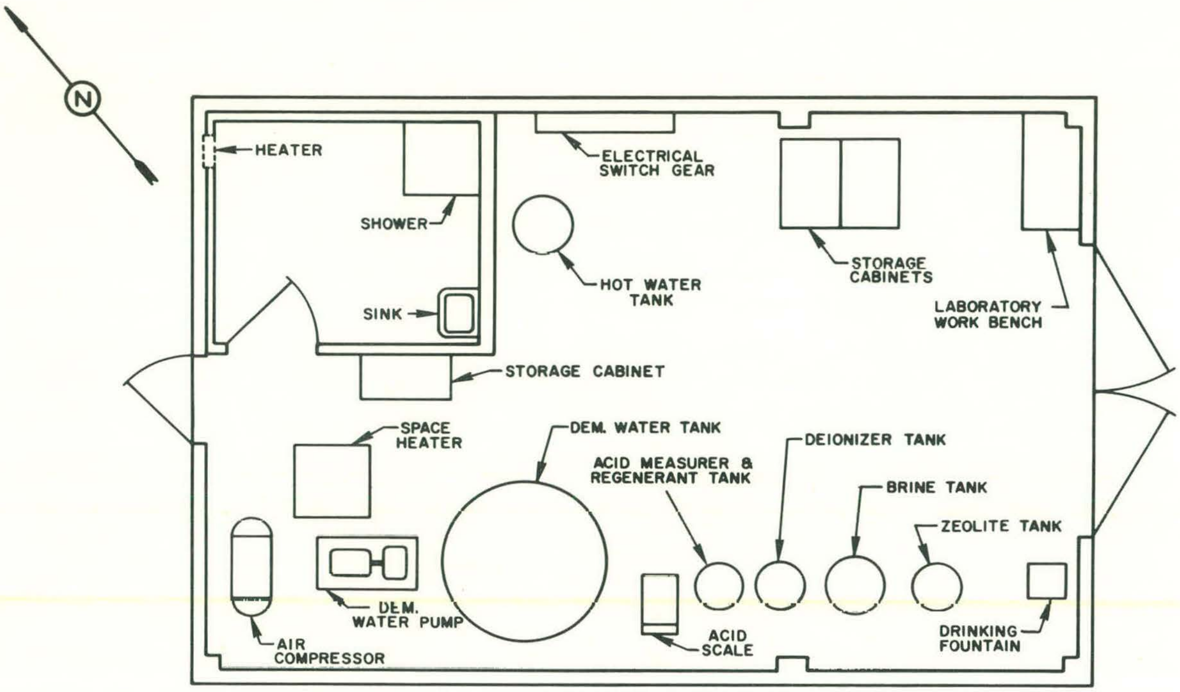

Fig. A-4 - Spert I Terminal Building - Floor Plan

4. Spert I Reactor Building

The Spert I reactor building is a single story, prefabricated, uninsulated galvanized iron structure, 20-ft wide, 40-ft long and 15-ft 10-in. high. An external view, plan and elevation are shown in Figs. A-5, A-6, and A-7. The reactor building houses the 16-ft deep by 10-ft ID reactor tank and associated equipment. A set of 12-ft by 11-ft metal sliding doors arc provided at each end of the building. The building is heated with an oil-fired, forced-air circulation furnace.

Adjacent to the northwest side of the reactor tank and embedded in the concrete building floor, are eighteen 6-in. ID by 14-ft-long tubes, with lead plugs, for the temporary storage of fuel or other radioactive material. On the south side of the reactor, four similar tubes are provided to accommodate neutron-sensing devices. These instrument tubes extend diagonally through the building floor to the reactor tank wall to a position approximately 2 ft above the concrete floor of the reactor tank.

Fig. A-5 - Spert I Reactor Building

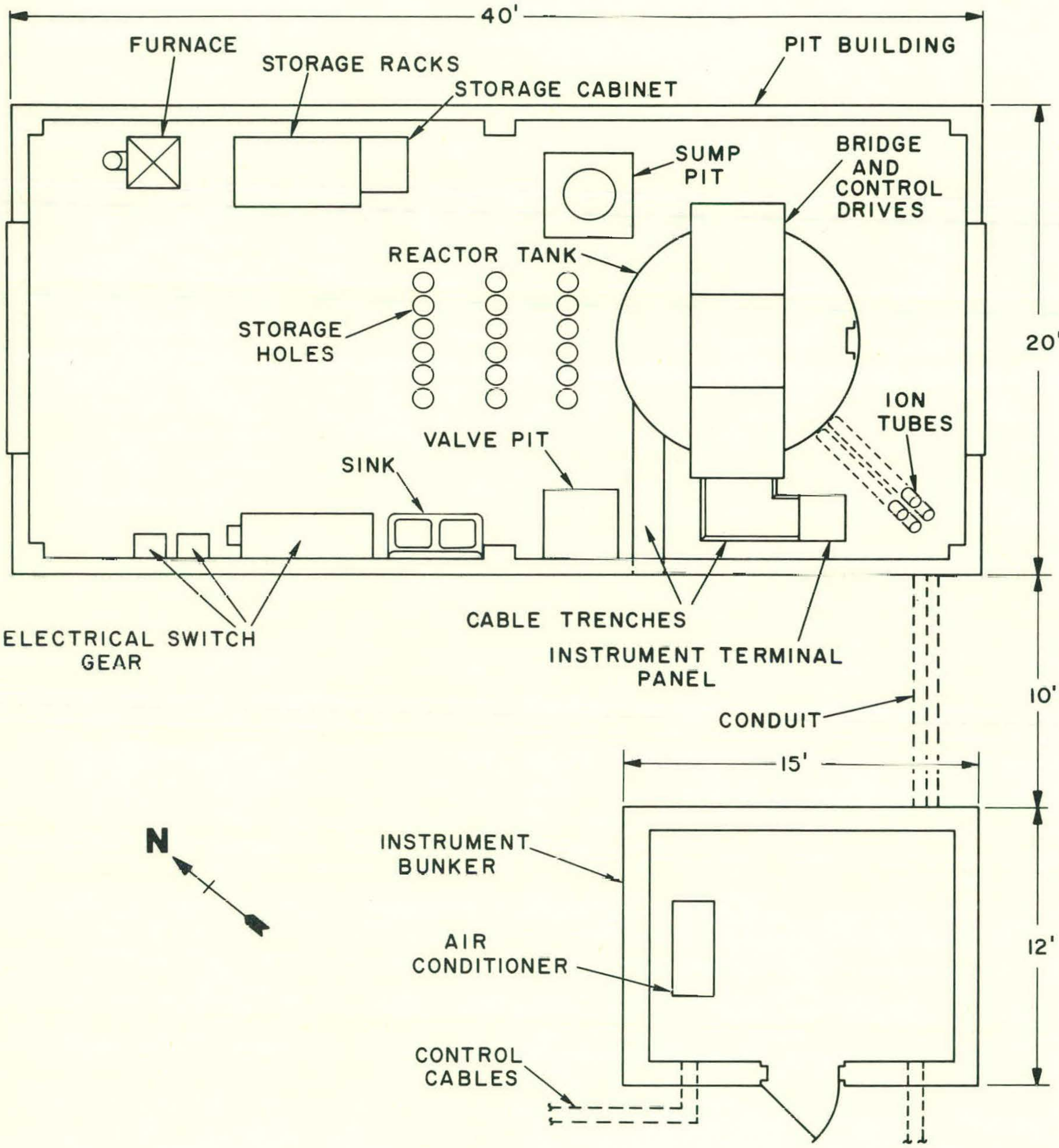

Fig. A-6. - Spert I Reactor Building - Floor Plan

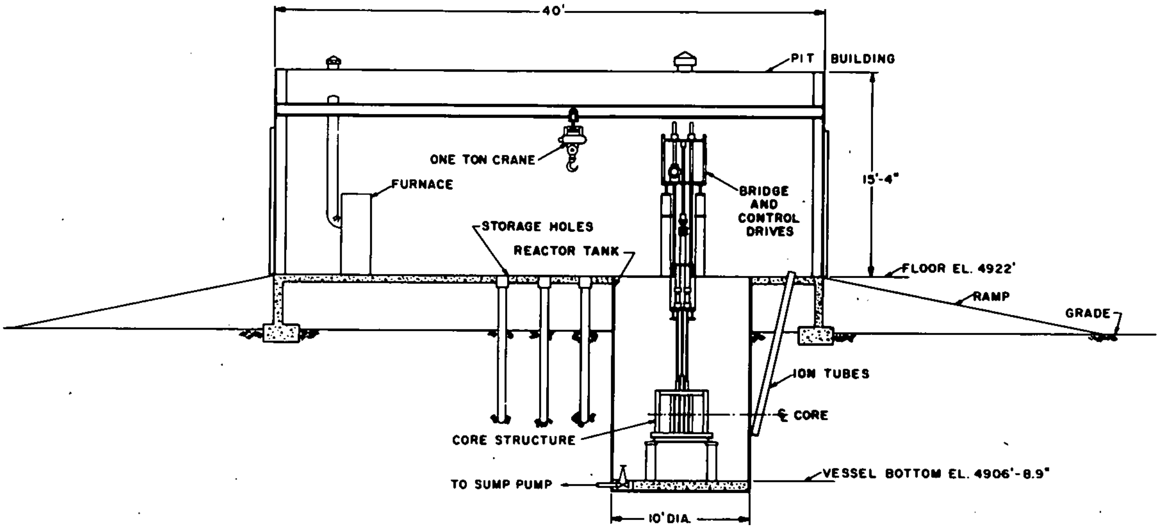

Fig. A-7 - Spert I Reactor Building - Elevation

Two additional pits are located in the reactor building floor. One, the valve pit, is 3-ft square by 6-ft deep and houses a manual valve on the process water inlet line and an air-operated valve on the deionized water reactor fill line. The other pit extends 18 ft below the building floor and houses the sump pump. The sump pit connects with the reactor drain line and facilitates draining of the reactor tank.

A one-ton electric hoist spans an 8-ft width in the center of the building and is operable over a length of the building from the reactor upper bridge structure to the northwest building access door.

The utilities supplied to the reactor building are deionized water, process water, 110-psig compressed air, 120/240-volt single-phase electric power, and 480-volt three-phase electric power. An isolation transformer is utilized to supply constant-voltage electric power for instrument use.

A cold water sink, which drains to the sump pit, is provided in the reactor building.

The building sump pump discharges into a leaching pond located 40 ft north of the reactor building. The leaching pond has a capacity of approximately 2500 cu. ft and is surrounded by an earth dike. There is no other provision for contaminated waste holdup.

Communication between the reactor building, the instrument bunker and the Spert I Control Room at the Control Center is provided by a battery-powered intercom system.

An 8-ft-square by 11-ft-high insulated frame structure is located 40 ft southwest of the reactor building. It was formerly used as a guardhouse but now contains the Health Physics constant-air-monitor instrument, protective clothing, and portable radiation monitors.

Located adjacent to this frame structure is a portable emergency power generator, housed in a galvanized iron enclosure. The emergency power generator provides power for the area monitor system, the constant-air monitor, the portable special air monitor and certain area lights. In the event of a power failure, an automatic transfer switch transfers Health Physics instrumentation load to the generator. The standby plant is rated at 2 KVA, 2 Kw, 1.0 P.F., 115/230 volts, single-phase, 3 wire, 60 cycle. The rotating armature-type generator is driven by a propane fuel engine capable of carrying 115% of the generator-rated load for eight hours without overheating.

5. Instrument Bunker

The instrument bunker, shown in the foreground in Fig. A-5 is a pumice block structure 15-ft long, 12-ft wide, and 9-ft high, covered with 1-1/2 ft of earth. Earth fill is also placed between this building and the reactor pit to protect the instruments from radiation damage during nuclear power bursts. Two 4-in. and one 3-in. conduit runs carry signal cables between the reactor building and the instrument bunker. Housed in the bunker are the necessary relays required for operation of the reactor equipment and the experimental instrumentation amplifiers, power supplies, drivers, etc., for transmitting signals over cables between the reactor area and the Control Center. The 7 kw heat load generated by this equipment is removed by an air conditioner.

To protect the electronic gear, a thermostatically controlled switch will cut the power to the bunker if the interior temperature reaches 85°F, provided that the reactor is not in operation. The compressor for the air conditioner is located in an enclosure adjacent to the bunker.

6. Spert I Control Room - Control Center Building

The Spert I Control Room is located in the Control Center building. The Control Room contains the reactor control console, which provides for remote control of the reactor and for experimental data recording. Various plant operations, including operation of the sump pump, reactor-fill pump, reactor-inlet water valve and electrical outlets, are also controlled from this Control Room. The reactor control console is discussed in detail in Section III-6 of Appendix A. An air conditioner supplying both the Spert I and Spert II Control Rooms protects the electronic instruments from overheating.

II. Reactor Components

1. General

The Spert-I reactor, a natural-circulation, open-pool-type reactor, consists of a reactor tank, core structure, lower support bridge, upper support bridge, fuel assemblies, and control- and transient-rods, with their associated mechanical drive units and electrical control circuitry. This section of the report is a description of the reactor tank, upper and lower support bridges and the fuel assemblies. A detailed description of the control system including the control- and and transient-rods and their associated drives, electrical circuits and nuclear instrumentation is given in Section III of Appendix A.

2. Reactor Vessel

The reactor vessel was fabricated in 1955 by the Western Steel Company of Salt Lake City, Utah and is embedded in the reactor building floor with sifted dirt as backing. The vessel is a 10-ft high, carbon steel tank, with a wall thickness of 1/4 in. Five external 3-in. x 4-in. x 1/4-in. angle irons, positioned every three feet from the top of the tank down, act as stiffening rings. Two carbon steel plates, 15 in. x 1 in. x 4 ft, are welded to opposite sides of the tank rim to act as bridge support pads. The floor of the reactor tank and the reactor-core-support base structure are integrally formed into a 9-in. thick, 3000 psi, reinforced-concrete pad. The core-support base structure consists of four 8-ft-long, 4-in. WF 13 beams, forming a 2.5-ft square, with 2.75-ft-long arms. Reinforcing is provided by two courses of ASTM A-305 and A-15 No. 6 medium-grade reinforcing bars on 10-in. centers, welded to the reactor tank wall and to the base structure. The base structure was leveled using mild steel wedge shims, tack-welded to the tank floor and to the base structure. Four, 1/2-in. thick by 14-in. square, mild-steel plates are welded to the corners of the square formed by the core support base. Sixteen core support column anchor bolt holes were located and drilled from a template in these plates after the core support base structure was installed in the tank. The core support column anchor bolts are 1 - 8 NC-2 and are made of 18-8 stainless steel. Twenty additional bolts are welded to the base structure for use as instrument bracket supports. Fig. A-8 is a photograph of the bottom of the reactor tank just prior to pouring the concrete.

Also pictured in Fig. A-8 is the sump form, fabricated from 1/4-in. mild-steel plate. This form houses the 2-1/2 in., schedule-40 18-8 stainless-steel drain pipe, which connects to the sump pit. A 2-1/2 in. gate valve with a screened flange is installed at the termination of the reactor drain line in the reactor tank. A mild-steel cover is installed to protect the reactor drain valve from damage during severe transient tests.

Two 4-in. mild-steel channels, rolled to a 10-ft diameter, are welded inside the tank. One, located 1 ft from the top, serves as a cable run. Clips welded to the channel serve as cable retainers. The other, 34-11/16 in. from the top of the reactor tank, serves as the working platform (grating) support. The working platform is pictured in Fig. A-9.

The carbon steel tank was sandblasted and spray-painted with two coats of Phenoline primer and three coats of white Phenoline paint to control rusting and deposition of corrosion products in the core. The concrete floor was hand brushed with two coats of Phenoline prime and three coats of white Phenoline paint to seal the surface for ease in decontamination.

A 42-in.-high guardrail, built in six removable sections of 1-1/4 in., schedule-40 pipe and painted yellow, surrounds the tank.

A carbon steel ladder is welded to the tank with the bottom 9 in. sunk in the concrete floor. Three brackets are welded to the tank to support a water-level indicator. Also welded to the tank are four periscope brackets, three pressure transducer brackets, a stirrer mount and a fill pipe mount.

An existing 6-in. tank-overflow line, which is directed to the leaching pond, is located 2 ft below the top of the tank. Since the support bridge is 3 ft below the top of the tank, this overflow line will not be normally used.

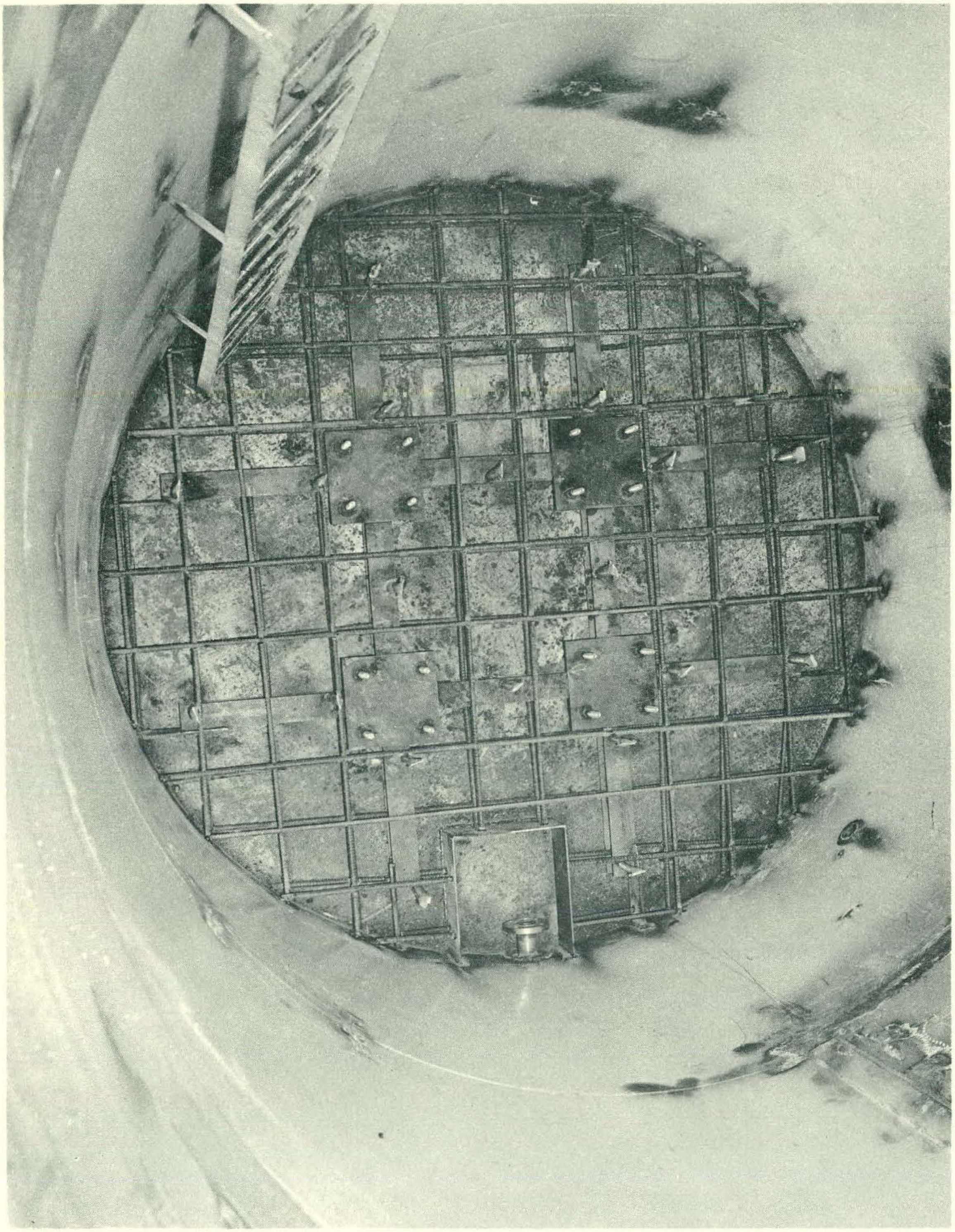

Fig. A-8 - Bottom of Reactor Tank Prior to Pouring Concrete

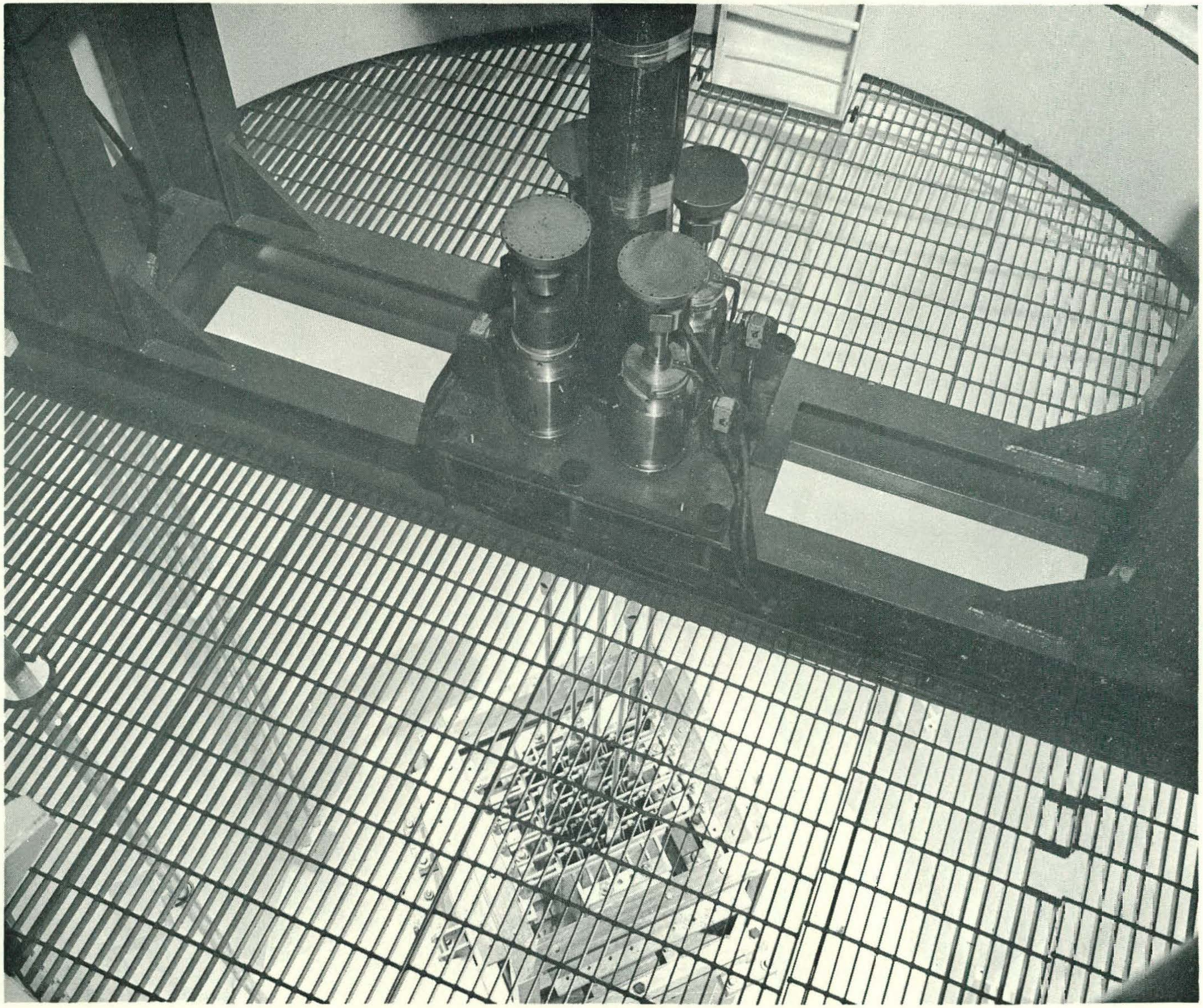

Fig. A-9 - Reactor Grating

3. Core Structure

The destructive test core structure, shown in Fig. A-10, consists of the following: the core support columns, shim plates, the grid support assembly, the grid assembly and the hold-down frame assembly.

The four support columns are 31-1/4-in. long, 6-in. WF 15.5 mild-steel beams. A 12-in. square by 1-in. thick and a 9-in. square by 1-in. thick, mild-steel plates are welded to the bottom and top, respectively, of each column. Each column is fastened to the core-support base structure by means of the anchor bolts and 5/8-11NC-2 heavy hex stainless steel nuts. The top plate has three 7/8-in. holes drilled through and three 3/4-10 NC stainless steel hex nuts welded to the underside. The columns were primed and painted with white Phenoline for corrosion resistance.

Two 6061-T6 aluminum shims, nominally 1-in. thick and 9-in. square, with three 1-1/4-in. holes drilled through, are inserted between each column and the grid support assembly. Leveling of the grid support assembly is accomplished by making minor variations in the thickness of the 1-in. aluminum shims.

The grid support assembly, shown in Fig. A-11, is 6061-T6 aluminum and consists of a 7/8-in. plate, 40-1/4-in. square, from which a 28-1/4-in.-square center section has been removed. This plate is welded to a square frame of 1/2-in. plate, 6-1/4-in. deep. Welded to the corners of this square frame are four 9-in. square by 7/8-in. thick plates, with three 1-1/4-in. holes drilled through. These lower plates and the shim plates fasten to the support columns with twelve 3/4-10 NC-2 by 5-in. long, stainless steel hex head cap screws and spring-lock washers. Eight 1-8 NC-2 by 1-in. long "Helicoils" are inserted in the 40-1/4-in. square plate to permit fastening of the grid assembly to the grid-support assembly. Four lifting eyes are provided on the support assembly.

The grid assembly and hold-down frame assembly were pre-assembled and transported to Spert I (see Fig. A-12). The grid assembly consists of 6061-T6 aluminum plate, 4-in. thick by 40-1/4-in. square. A 27.5-in. square section has been removed from the center of the plate and replaced

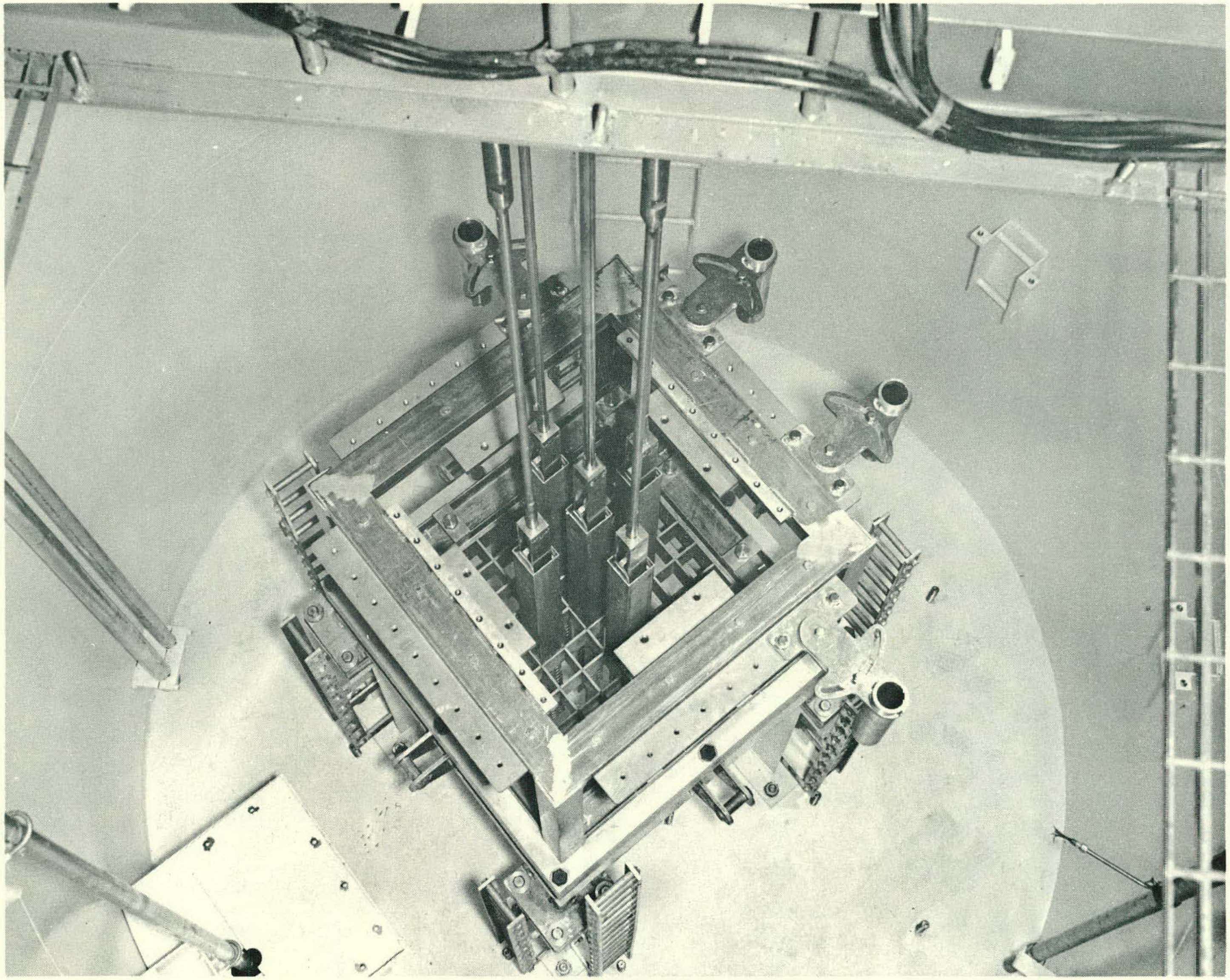

Fig. A-10 - Reactor Core Structure

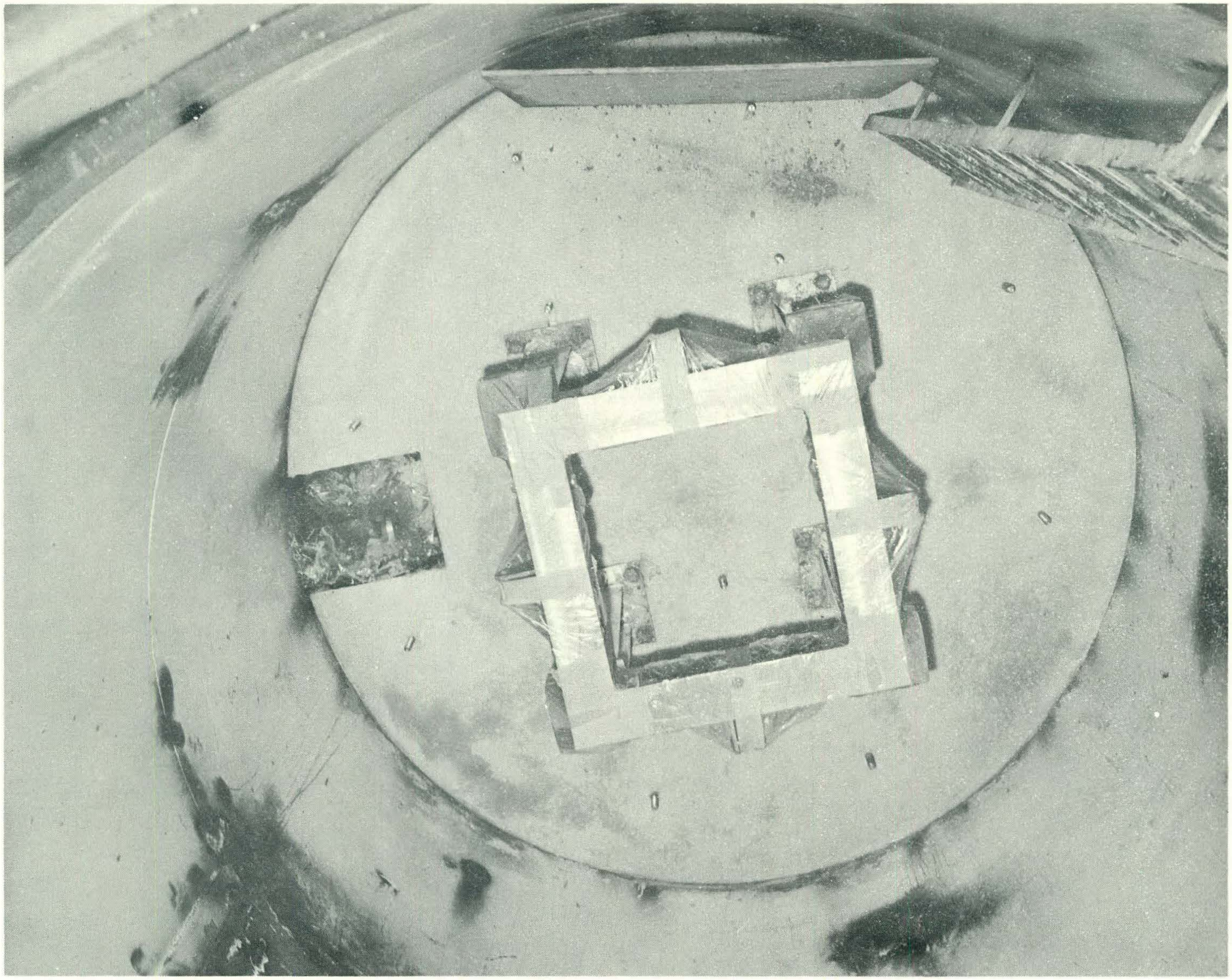

Fig. A-11 - Grid Support Structure

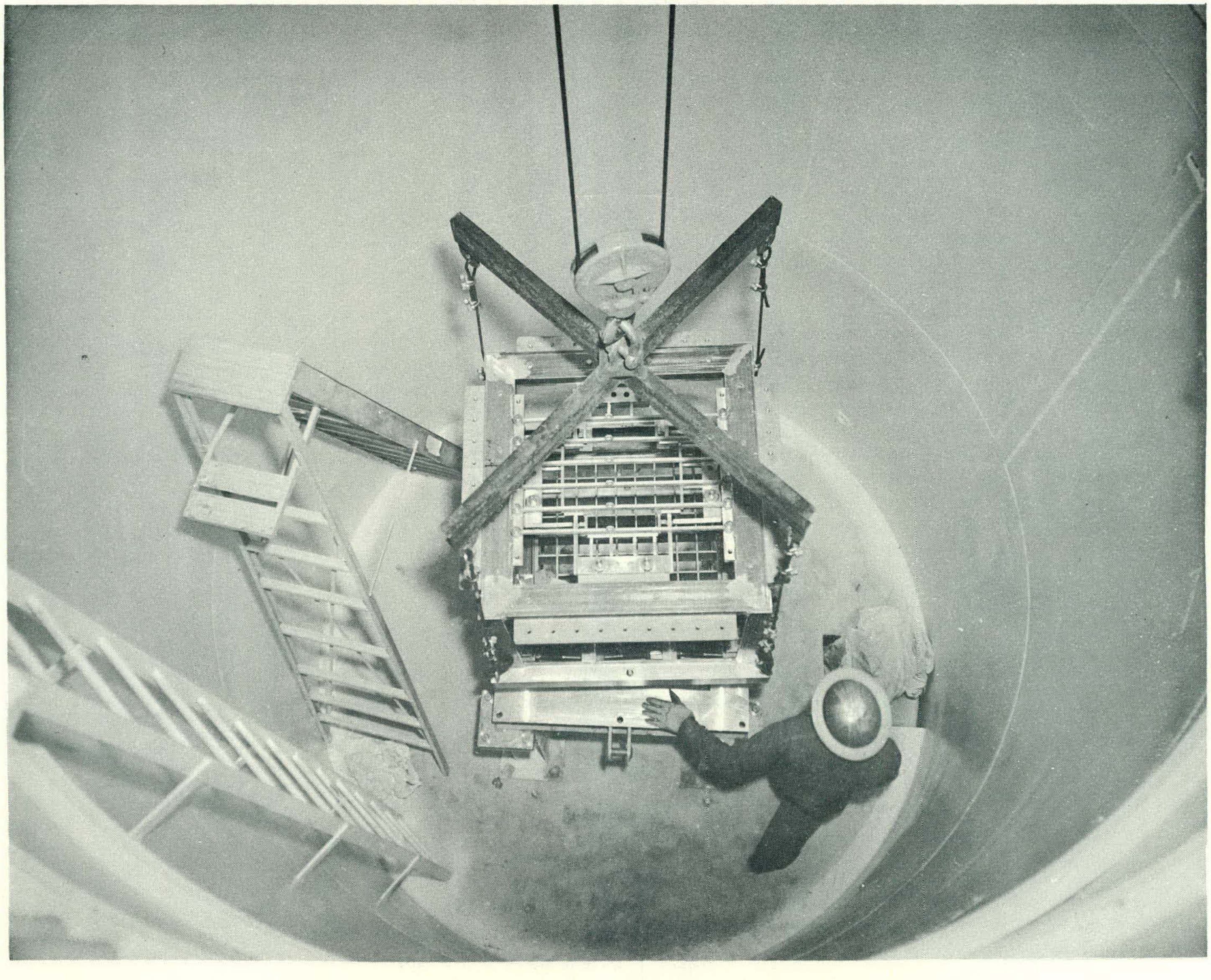

Fig. A-12 - Grid Assembly

with an interlocking egg-crate assembly of 0.300 in. by 4 in., 6061-T6 aluminum plate. The egg-crate is fastened with twelve 0.250 in. by 5/8-in. long stainless steel cap screws to the grid plate. The egg-crate assembly divides the square section into a 9 x 9 lattice of 3-in. by 3-in. cells. The lower end boxes of the fuel assemblies are retained laterally by the egg-crate. The grid assembly fastens to the grid-support assembly with eight 1-8 NC x 5-in. long hex head stainless steel cap screws.

The 6061-T6 aluminum hold-down frame assembly fastens to the grid assembly with twelve 1-8 NC x 3-1/2-in. long hex head stainless steel cap screws. The hold-down frame assembly holds the fuel assemblies securely in place by means of hold-down bars, which fit over the tops of the fuel assemblies, and by means of core clamps, which fit against the sides of the core-periphery fuel assemblies. The core configuration pictured in Fig. A-13 shows the core clamps for a 5 x 5 lattice with the corners removed. Other core clamps are available for different core configurations, such as 5 x 5 with corners in, 5 x 6, 6 x 6, and 7 x 7.

The hold-down frame assembly consists of 4 in. aluminum channel welded into a 35-3/4-in. square, with 3-in. aluminum-angle legs, each 27-3/8-in. high. Two 1-1/4-in.-wide by 23-in.-long aluminum bars are welded on the upper channel to support the hold-down bars. The hold-down bars are 6061-T6 aluminum and are fastened by 1/2-13 NC-2 stainless steel captive screws. The bars may be removed by using long-handled tools, thereby allowing any or all of the fuel assemblies to be removed while the core is under water. Four 6061-T6 aluminum core clamp support plates fasten to the hold-down assembly with sixteen 1/2-13 NC 3-1/4-in.-long socket-head cap screws and hex nuts. Each support plate has nine 1/2-13 NC 3/4-in.-long Helicoil inserts, installed to provide support for instrument brackets, and two 3/4-10 NC 3/4-in.-long Helicoil inserts for support of the core clamps.

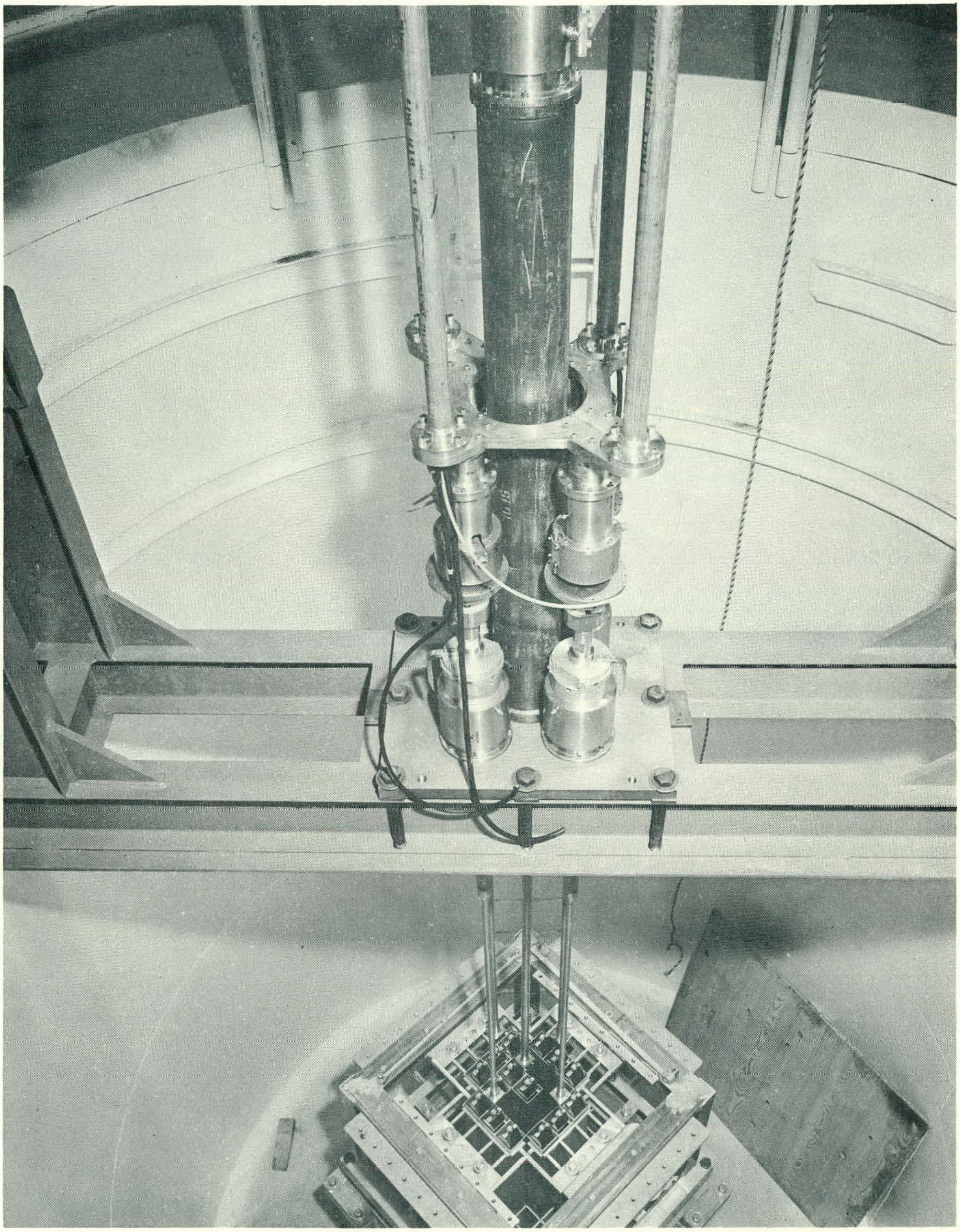

Fig. A-13 - Core Showing Core Clamps

4. Lower Support Bridge

The lower support bridge is shown being lowered into place in Fig. A-14. The bridge, of all-welded construction using 8-in.-I 23 mild-steel structural beams, is 1-ft 7-7/8-in. wide by 11-ft 6-in. long. The surface of the support bridge, upon which the control- and transient-rod shock absorber dash pots are mounted, is 2-ft 2-11/16 in. below the top of the reactor tank. The bridge is bolted to two 1-in. thick, mild-steel mounting plates, which are welded to the rim of the reactor tank. Each mounting plate has four 1-8-NC-2 by 4-in.-long bolts welded to it for the bridge fastening. The mild-steel dash pot mounting plate is 24 in. x 19-7/8-in. x 1-3/8-in. thick and has four 2-13/16-diameter holes to accommodate the control rods and one 4-1/16-in.-diameter hole to accommodate the transient rod. The mounting plate is shimmed with 1-in. mild-steel shims and fastened to the support bridge with six 1-8 NC by 12-in.-long and two 1-8 NC by 8-in.-long hex head steel bolts, and hex nuts and spring lockwashers.

5. Upper Support Bridge

The carbon steel upper bridge structure, which supports the control rods and transient rod drives, is pictured in place in Fig. A-15.

The upper bridge portion fastens to 53-in.-high stanchions, consisting of four mild-steel, 10-in.-WF 39 beams, with 1/2-in. plates welded to the top and bottom. The bottom plates are welded to the reactor tank rim. Each top plate has two 1-1/16-in.-diamter holes to allow fastening of the bridge to the support stanchions, using eight 1-8-NC by 2-in.-long hex head bolts and nuts. The upper bridge span is constructed of 10-in.-WF 39 beams and of 6-in.-I 12.5 beams. The span is 12-ft long by 2-1/2-ft high by 3-1/3-ft wide and is equipped with guard rails and gratings. A detailed description of the rod drives is contained in Section III-2 of Appendix A.

6. Fuel Assemblies

a. Standard Fuel Assemblies

The standard fuel assemblies for the Spert I destructive core, shown in Fig. A-16, are the Spert type-D assemblies. Basically, a

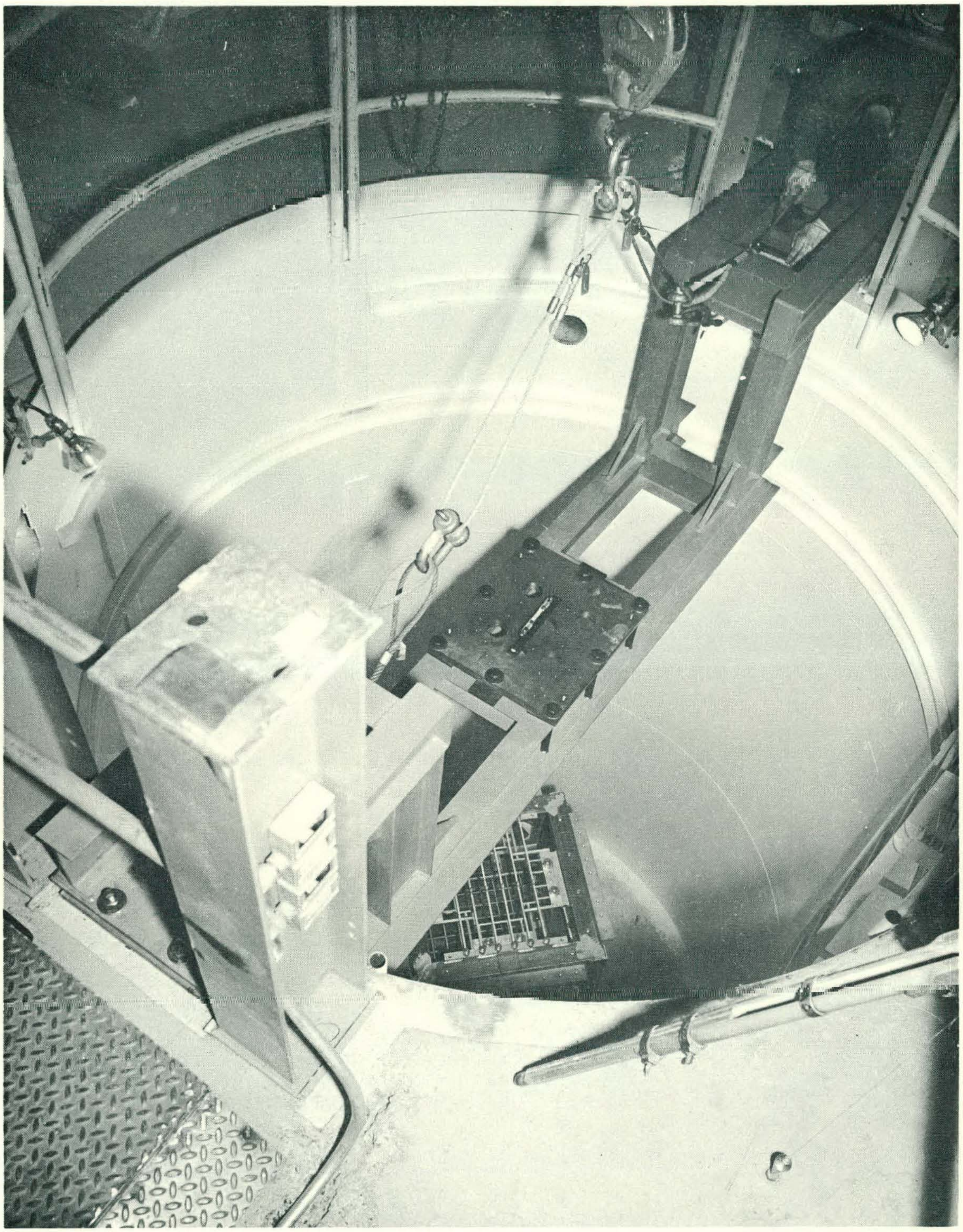

Fig. A-14 - Lower Support Bridge

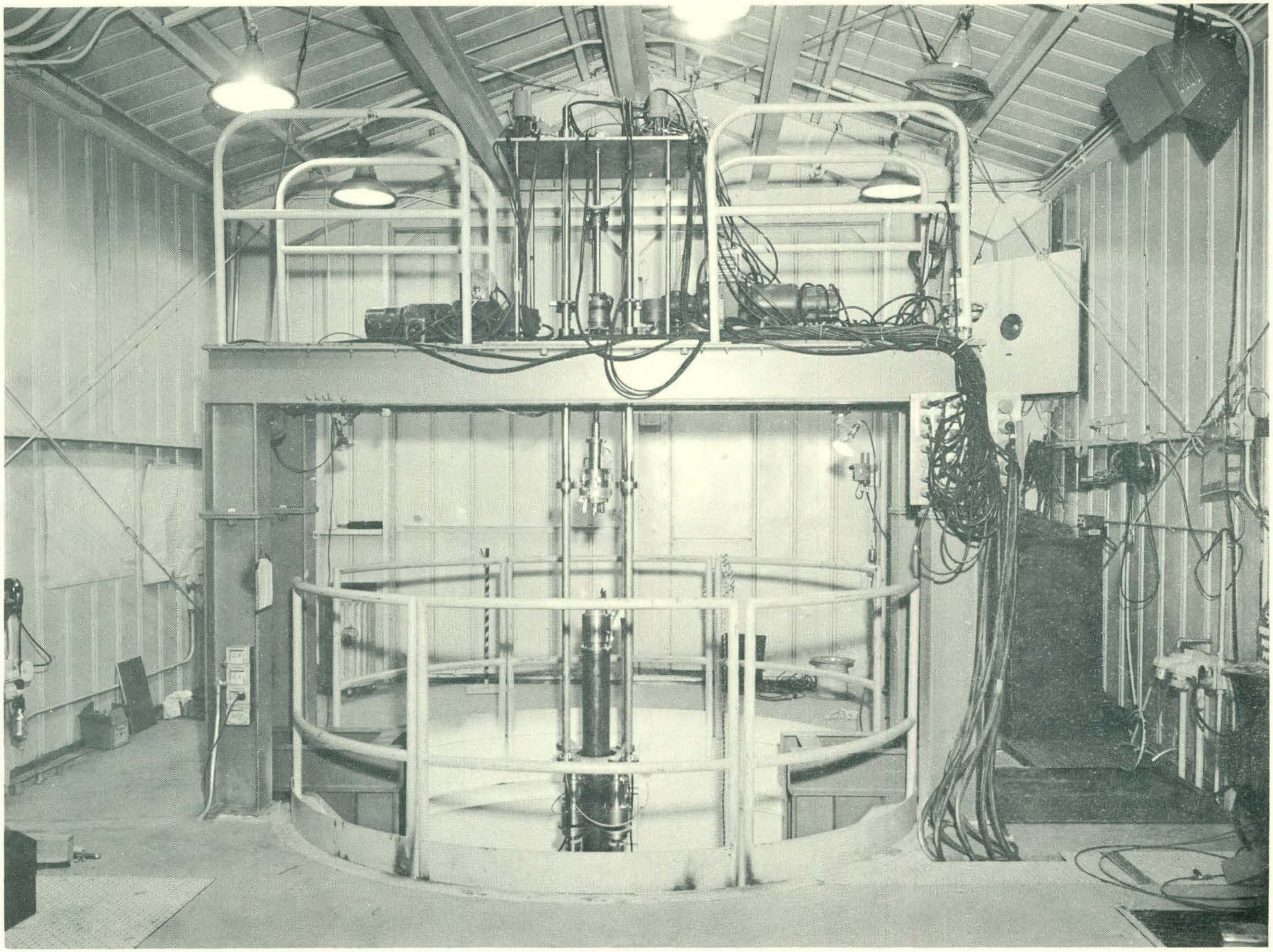

Fig. A-15 - Upper Support Bridge Assembly

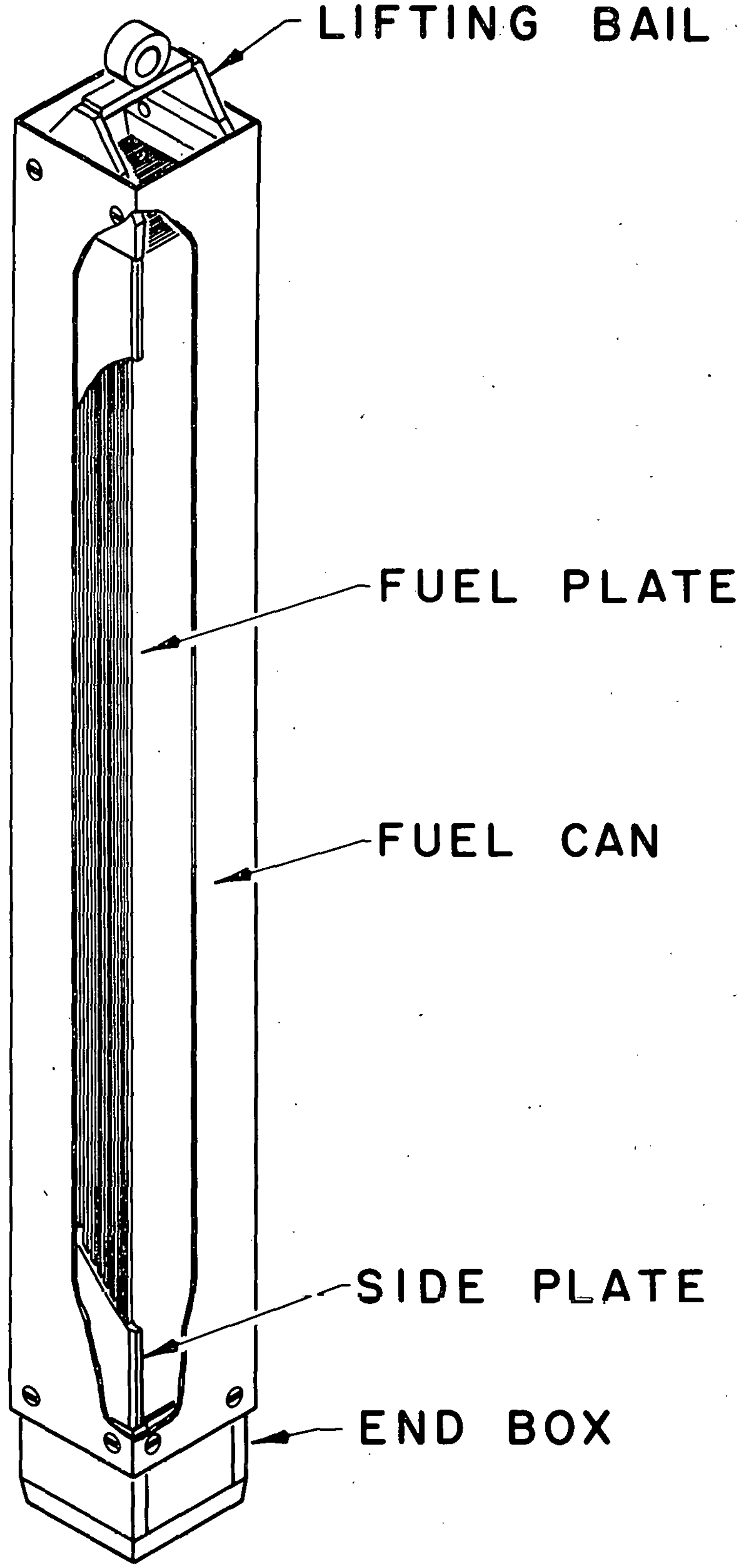

Fig. A-16 - Standard Spert Type-D 12-Plate Fuel Assembly standard Spert type-D fuel assembly consists of: a square end-box; a 2.996-in.-square, 0.060-in. wall, aluminum retaining can; two grooved side plates; 12 fuel plates; and a lifting bail. The lifting bail and end-box also serve to hold the plates in the assembly. Fuel plates can be taken from the fuel assembly by removing the lifting bail which is fastened to the aluminum retaining can by 4 machine screws.

The square end-box is machined from commercial 6061-T6 aluminum cast and the can is commercial square, 6061-T6 aluminum tubing purchased in accordance with Engineering Specification SPT-1012 contained in IDO-16745, which describes the Spert IV facility.

The active region in each fuel plate consists of 14 g of U-235 alloyed with aluminum melting stock to produce a core 0.020-in. thick by 2.45-in. wide by 24-in. long. The active region is clad with 6061 aluminum to produce a fuel plate 2.704-in. wide by 25-1/8-in. long by 0.060-in. thick. The water channel spacing between plates is nominally 0.179 in. Since the channel spacing can be changed by substituting different side plates or by removing fuel plates, the water channel spacing may be readily varied as experimental conditions dictate. Detailed fuel plate specifications are given in IDO-16745.

b. Control and Transient Rod Fuel Assemblies

Four special control rod fuel assemblies and one centrally located, transient rod fuel assembly are installed in the reactor core. These five assemblies are identical to standard fuel assemblies except for a slight modification of the end box to permit the attachment of lower blade guide assemblies. In addition, upper blade guide assemblies are inserted in place of fuel plates to position the control or transient rod blades through the active region of the reactor core. The upper and lower blade guide assemblies are constructed of 6061-T6 aluminum.

The upper blade guide assemblies and two control (or two transient rod) blades occupy 6 of the 12 fuel plate positions of the special fuel assemblies. The remaining six fuel plate positions contain standard fuel plates. Fig. A-17 shows the control and transient rod fuel assembly.

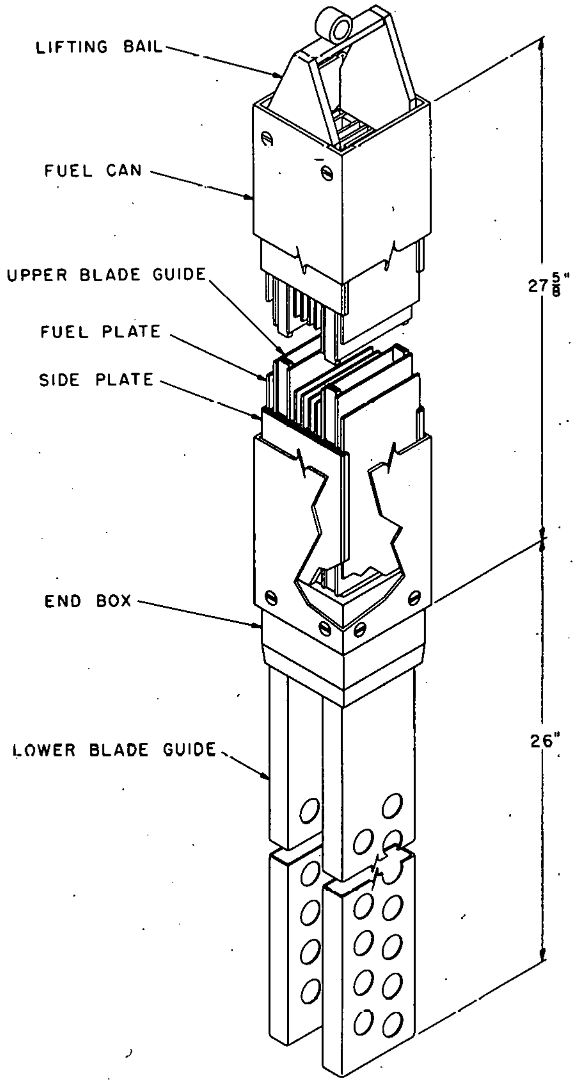

Fig. 17 - Rod Bearing Fuel Assembly

III. Reactor Control

1. Introduction

This section of the report is devoted to a discussion of the various components of the Spert I 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, must reflect somewhat the philosophy of operation of the facility. The purpose of Spert I 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. 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 approximately one-third mile of the reactor. Thus, 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 test types and the short test-time interval for most of the experiments led to the selection of a simple control system for Spert I, in which operation is strictly manual, with no servo or feedback loops in the control system. Because of the short time scale of the tests, the individual functions required to be performed during a transient test, such as ejecting the transient rod, starting data recording and photographic equipment, and insertion of control rods at a convenient time following completion of the test, are programmed on a sequence timer, with the test itself initiated by starting the timer. The reactor operator is always under the direct surveillance of at least one other qualified operator who provides backup and, together with all other persons in the control 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. 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.

The following subsections describe the control rod and transient rod drives, the control rods, the transient rod, the control system electrical circuits, and the reactor control console.

2. Control Rod and Transient Rod Drive Systems

a. General

The Spert I drive system was designed and installed in 1955 as a part of the original Spert I facility. Although numerous modifications have been made to various parts of the drive system in order to accommodate changes in requirements for control rod locations in the eight different cores which have been tested in Spert I, the basic drive system has not been changed. The drive system consists of magnetically latched control rods driven by a single, variable-speed, motor-transmission combination and a mechanically latched transient rod driven by a second motor-transmission combination. The drive units are mounted on a fixed bridge spanning the reactor vessel and independent of the core structure. A photograph of the drive system is shown in Fig. A-18.

The output of the drive motors is through chain and sprocket drives acting on ball nuts and screws which are connected to the rod shafts. By changing the variable-speed transmission gear-head, the control rod maximum withdrawal rate can be varied between 10 and 35 in./min. A total control rod travel of 23.7 in. is available.

The control rods are coupled to the rod shafts by means of four individual electromagnets and armatures. Although the control rod drives are fastened together by a plate, the individual magnets permit raising or scramming individual or various combinations of the control rods. De-energizing the magnets allows the control rods to fall and they are accelerated through the first two inches of their downward travel by means of small air pistons and plungers. Scram time as measured from initiation of scram signal to shock adsorber contact is about 300 msec.

The transient rod drive is coupled to the transient rod by a mechanical latch. The transient rod is "fired" when the latch is disengaged by air pistons acting on a release ring. The transient rod is also accelerated in its downward travel by means of an air piston. Transient rod travel time is about 200 msec from latch release to shock adsorber contact.

An indication of the control rods and transient rod positions while the drives are in physical contact with the rods is provided to the nearest 0.01 in. by Telesyn transmitter-receivers and register-indicators operating from the rod drives.

Both upper and lower limit switches are provided on the drives to prevent overtravel.

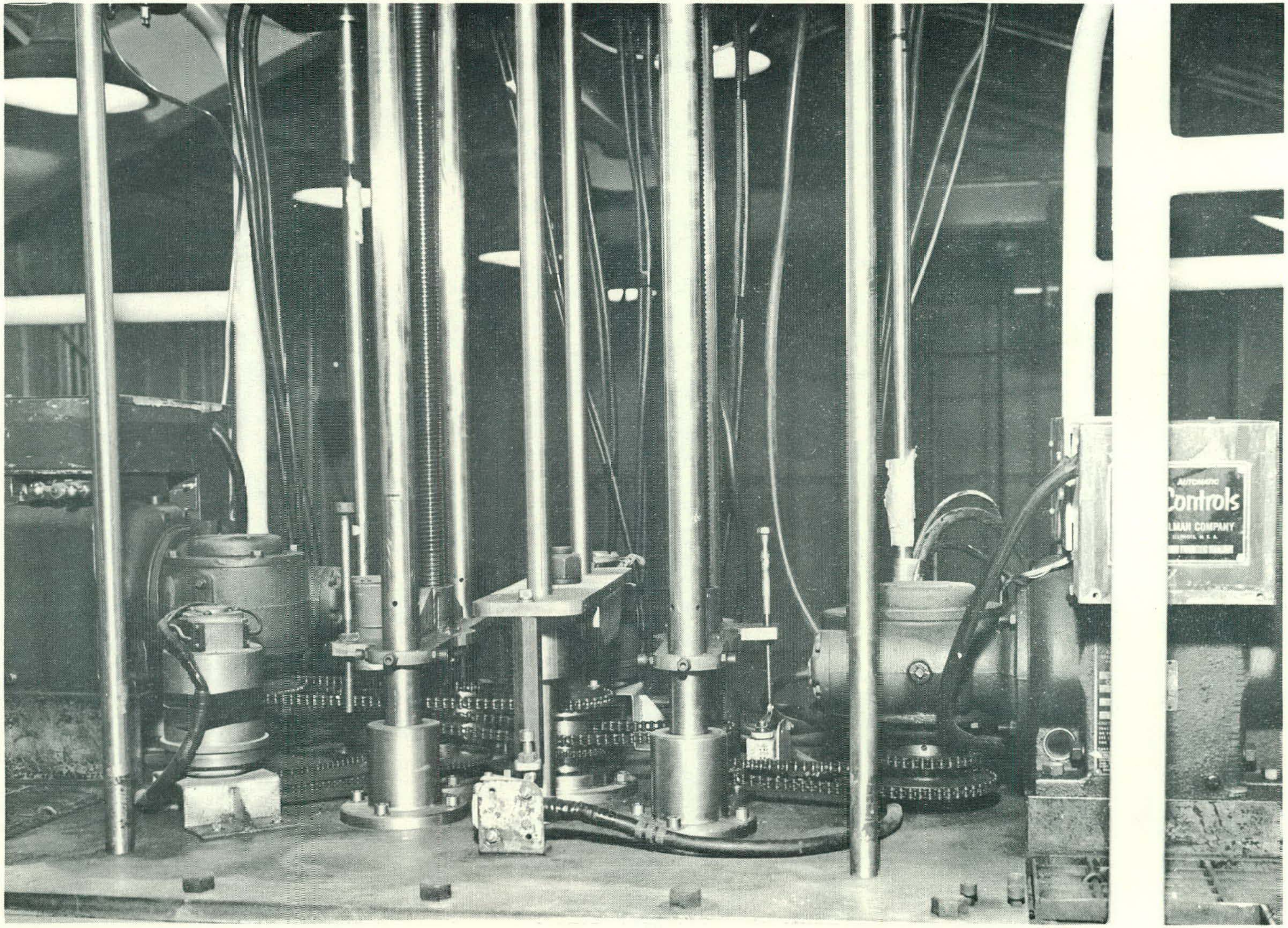

Fig. A-18 - Rod Drive System

b. Detailed Rod Drive Component Description

(1) Introduction

The drive consists of the following components which will be discussed in detail: the lower and upper support plates, drive motors and transmissions, drive sprockets and chains, ball nuts and screws, position indicators, rod shafts, magnets and air-driven pistons, limit switches and shock absorbers.

(2) Lower and Upper Support Plates

The lower drive support plate is a 3/4-in. thick, 304 stainless steel plate which is bolted to the upper support bridge. This plate supports the drive motors and transmissions, drive sprockets and chains, and control- and transient-rod bearings and bushings. The 1/2-in.-thick, 304 stainless steel, upper support (guide) plate is attached to the lower plate with six 38-in.-long, 304 stainless steel support rods. This upper plate aligns the drive system and supports miscellaneous equipment.

(3) Drive Motors and Transmissions

The drive system includes two 1/2 hp, 480 v, 3 ϕ, induction motors driving Graham variable-speed transmissions, Model 150 MW 18. One unit operates the four control rods, and the other operates the transient rod. The motors operate at 1150 rpm and are equipped with magnetic brakes.

(4) Control Rod Drive Components

The output of the control rod drive transmission is through a gear shaft which connects to a carbon steel sprocket and then through a single row roller chain to another sprocket connected to a 37-9/16-in. long ball screw. The ball screw has a 1.150-in. diameter ball circle and was manufactured by the Saginaw

Steering Gear Division of General Motors Corporation. A roller chain connects this screw to another identical screw with a 1:1 ratio. The screws are fitted with single row radial ball bearings and have bearing housings fastened to the upper and lower support plates.

One ball screw is connected by chain and sprocket to the control rod drive Telesyn transmitter, manufactured by the Singer Manufacturing Electrical Division, Model No. C-69405-2. The transmitter is coupled to a receiver at the Spert I control room which drives a digital readout of rod position to the nearest 0.01 in.

Each ball screw drives two control rod shaft extensions through a ball nut and yoke connection. The shaft extensions are 121-5/16-in. long and are made of 304 stainless steel. They are hollow to permit electrical power and compressed air to be delivered to the magnets and are guided by nylon bushings. A control adapter plate bolts to the ends of the four shaft extensions fixing them in a single unit.

A control rod magnet assembly is shown in a pictorial cutaway view in Fig. A-19. The four magnet assemblies are fixed to the control rod adapter plate. The electromagnets of the Spert I drives are cylindrical, with an outer diameter of 4-7/8 in. An axial section of the core and armature is of conventional "E-I" appearance. Each magnet has an individual power supply and the current to each magnet coil is adjusted to give the same release time to each control rod. Normal operating current is approximately 0.1 amp, with a corresponding release time of less than 50 msec. To scram the reactor, the magnets are de-energized and the control rods are allowed to fall

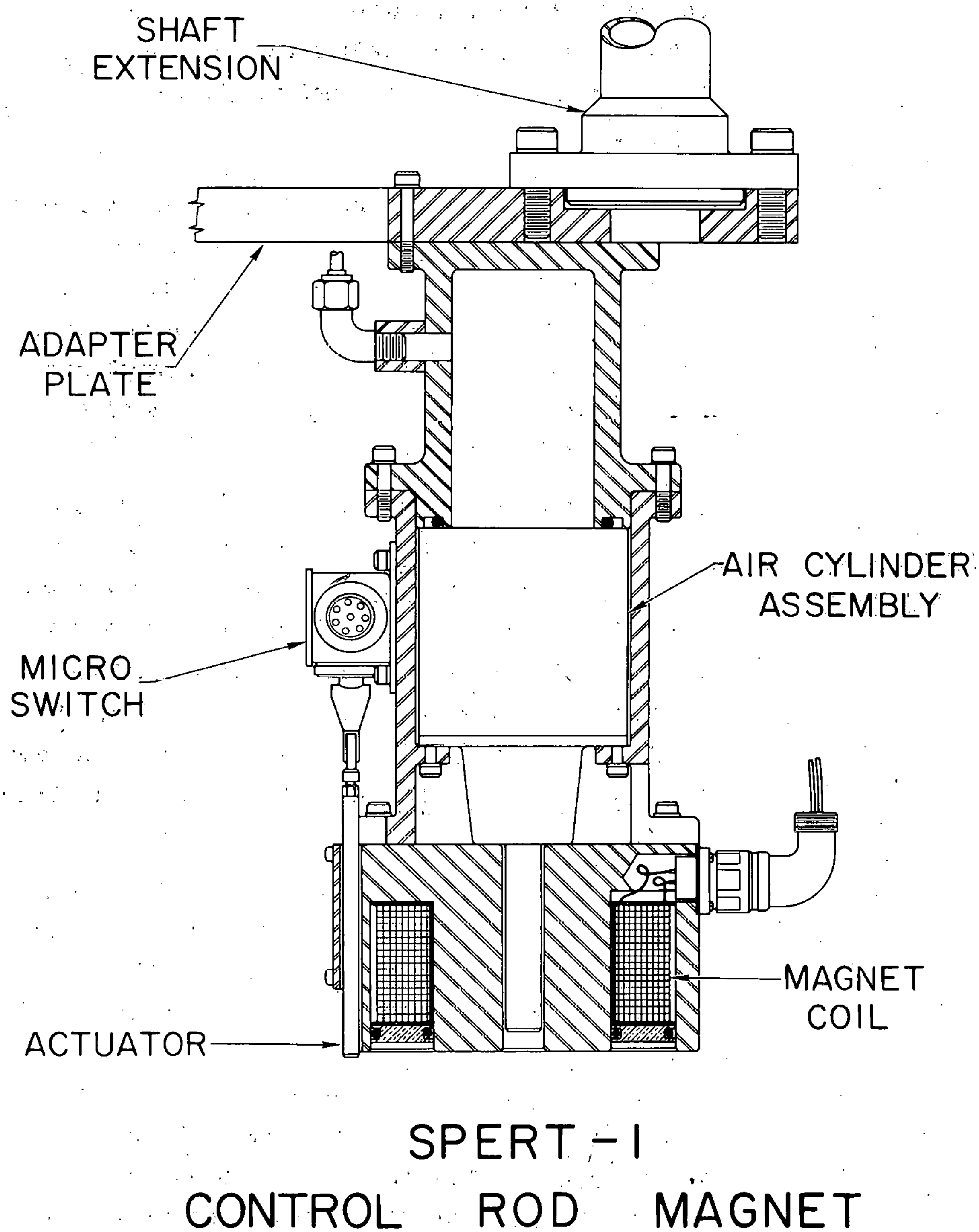

Fig. A-19 - Control Rod Magnet Assembly

into the core by gravity after an air-piston-assisted breakaway. Air is introduced into the cylinder assembly when the magnets are energized. The four control rod air pistons operate with 50 psig air pressure.

When a magnet is in contact with the armature, a micro-switch is actuated which lights the magnet contact light on the control console.

(5) Transient Rod Drive Components

The output of the transient rod motor transmission is connected through a chain and sprocket to a ball nut. The ball nut is fastened to a 304 stainless steel shaft which is attached to the lower support plate. This ball nut drives a 42-1/2-in.-long, 1.150-in.-diameter ball screw up and down to position the transient rod. The upper end of the ball screw is guided by a yoke which is fitted to two of the six support rods.

A transient rod ball screw guide is provided to prevent excessive sway of the transient rod. The guide is made of mild steel with a 1.144-in.-ID oilite bronze bushing, and fastens to the underside of the lower support plate with the same studs that fasten the bearing housings.

The transient rod ball screw has an extension fitted to its lower end which fastens to the latch assembly.

The transient rod latch is shown in a pictorial cutaway view in Fig. A-20. The latch is disengaged when 90 psig air is admitted to the latch cylinder and the latch ring is depressed. At the same time, a 50 psig air piston accelerates the rod out of the core. A key interlock switch which is mounted on the reactor control console is provided to prevent unintentional initiation of a power excursion. The

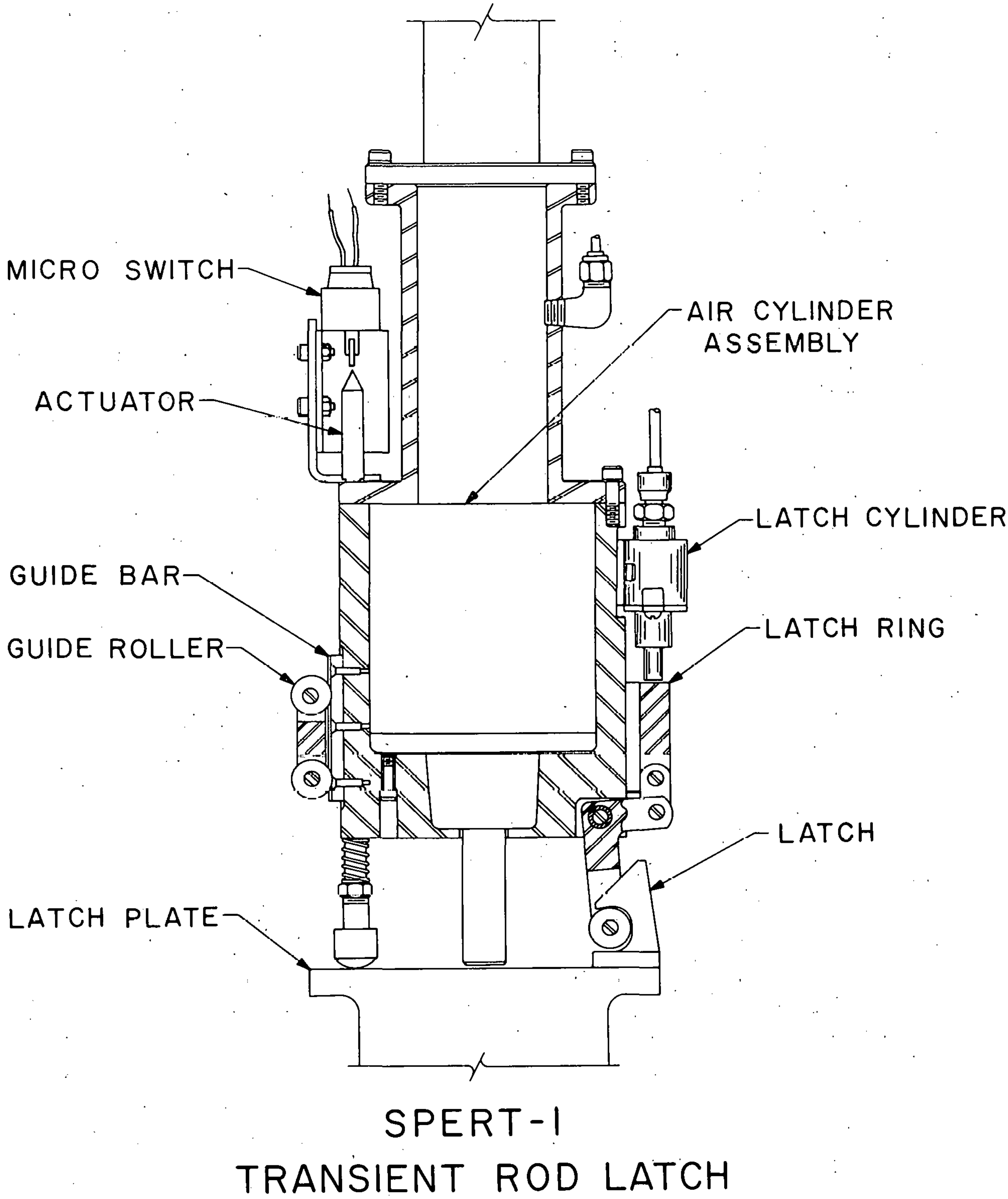

Fig. A-20 - Transient Rod Latch Assembly key switch "arms" (allows compressed air to be supplied to the latch mechanism) the reactor and is a necessary condition for release of the transient rod. A micro-switch actuates a contact light on the control console when the latch is engaged.

(6) Limit Switches

The various limit and contact microswitches are 4-pole, double-throw, 115 v, hermetically-sealed, type H-2.

(7) Shock Absorbers

Deceleration of the control rods and the transient rod is accomplished by dash-pot-type shock absorbers mounted to the dash pot support plate on the lower bridge structure. Essentially, the dash pot consists of a single 6061-T6 aluminum piston in a 304 stainless steel cylinder with a reservoir of SAE 90 W oil. The kinetic energy generated in scramming the rods is not great, and the shock absorbers provide deceleration over a total travel of approximately 1 in. Two music-wire springs, which are not strong enough to bounce the control rod after scramming, return the piston to the normal position after each compression. Microswitches indicate when the piston is up. Air is allowed to escape through 3/16-in. grease fittings during the compression of the shock absorber.

The control rod dash pots are fastened to the dash pot support plate with five 1/4 - 20 NC screws. The transient rod dash pot fastens to a mild-steel support, 52-9/16-in. long and 5-1/2-in. OD, with eight 1/4 - 20 NC screws. The support fastens to the dash pot support plate with eight 1/2 - 13 NC screws. Fig. A-21 shows the shock absorber assemblies.

3. Control Rods

There are four control rod units operating in the reactor core. Each unit consists of the following: the armature, upper drive rod, rod adapter, dash pot bumper, drive rod upper section, lower section, yoke, two poison (neutron-absorbing) blades and two aluminum-follower blades.

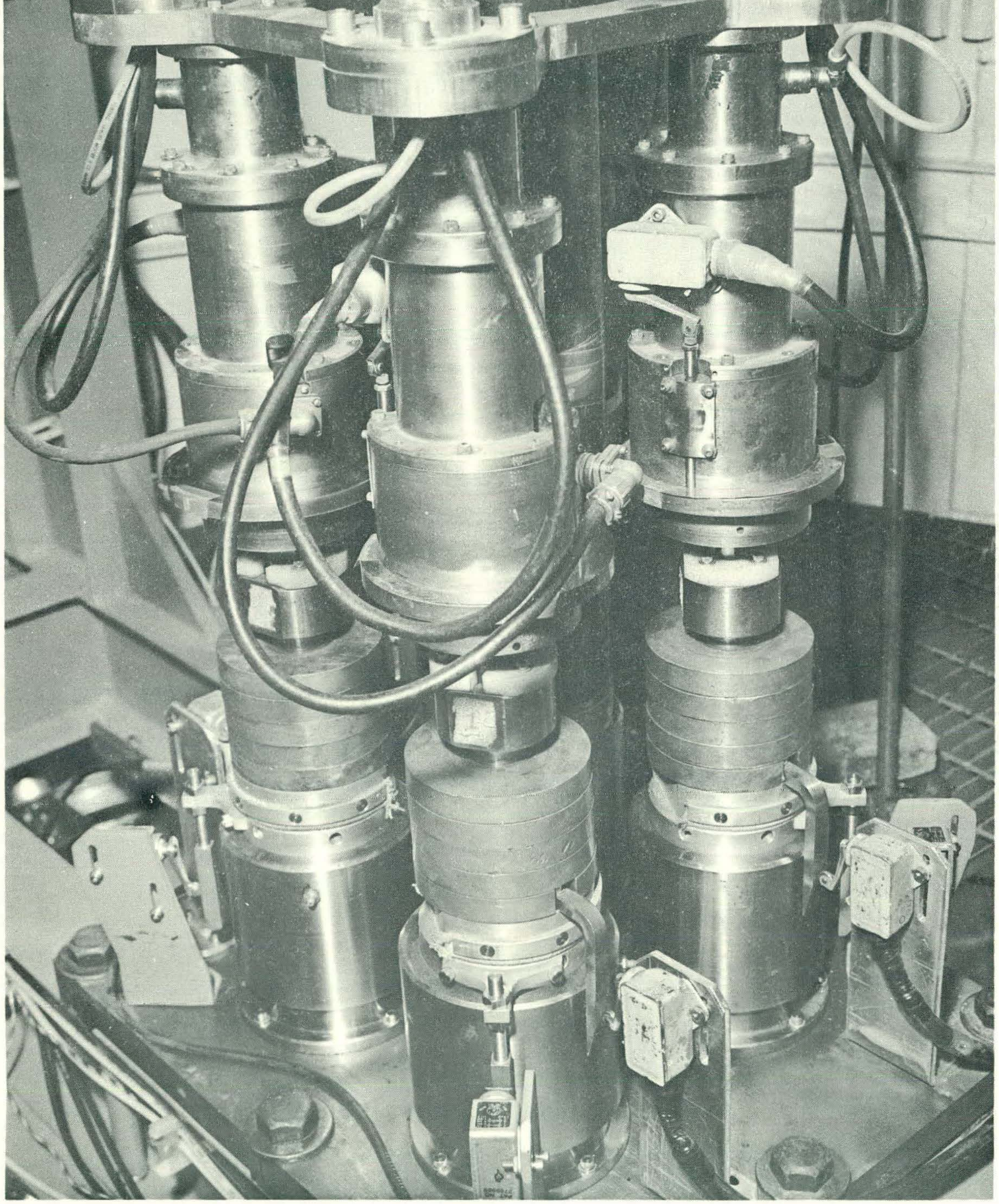

Fig. A-21 - Shock Absorber Assembly

The Armco-iron control rod armatures are mounted on swivel joints to permit positive mating with the control rod magnet assemblies. Thin, stainless shims are attached to the armatures to impede currents induced upon de-energization of the magnets, which would tend to decrease the release time. The armature is connected through a 304 stainless steel rod adapter to a 1.5-in.-diameter, 38-3/16-in.-long, 304 stainless steel drive rod upper section. This section is fitted with a 6061-T6 aluminum dash-pot bumper, which shoulders on the dash pot piston when the control rod is scrammed. Five lead weights, totaling 22 lb, are fitted to each control rod to provide added mass for proper functioning of the shock absorber units. A seat switch is actuated by a bayonet on the armature when the control rod is fully inserted in the core.

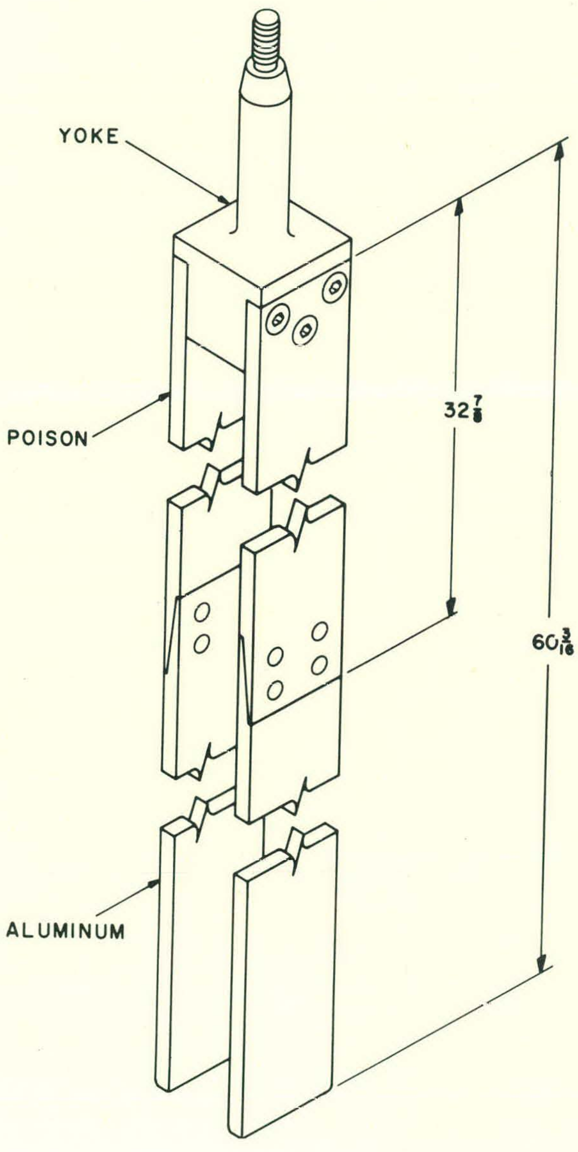

Fig. A-22 - Yoke and Blade Section of a Control Rod

A 3/4-in.-diameter, 46-23/32-in.-long, stainless steel lower section mates with the upper section of the armature by means of 5/8-11 NC-2 threads. This connection is pinned. A 304 stainless steel yoke mates with the lower section by means of 1/2-13 NC-2 threads. The yoke is also pinned.

The poison and follower section of the control rod consists of two flat blades secured to the yoke by three 5/16-18 NC x 15/16-in. flat head socket cap screws per blade. The yoke and blade section of a control rod is shown in Fig. 22. The cap screws are staked at assembly to prevent loosening of the screws and possible separation of the blade from the yoke.

Each blade of a control rod is made from two sections: the follower section is made of 6061-T6 aluminum, while the poison section is made of "Binal"* containing 7 wt% Boron. The sections are tapered on one end and riveted together at the tapers by 4-3/16-in.-diameter, 100° flat head aluminum rivets to form a blade 5/16-in. thick, 2-in. wide and 60-3/16-in. long. The upper (poison) section of the control rod blade is 32-7/8-in. long. The lower (follower) section serves as both a guide and flux suppressor as the control rods are withdrawn.

Each blade of the control rod operates in a guide slot that replaces three of the fuel plates in the standard type-D fuel assembly. The end boxes of the control rod fuel assemblies are equipped with lower blade guides which mate with the blade guides in the fuel assembly. The lower blade guides, which extend 26 in. below the grid assembly, have been perforated with 3/4-in.-diameter holes to facilitate escape of water from the guides when the rods are dropped. A description of the control rod fuel assembly is given in Section II-6 of Appendix A. Fig. A-23 is a view of the core, with control rod No. 4 withdrawn 10 in.

4. Transient Rod

The transient rod is used to initiate excursions by the sudden addition of reactivity. The transient rod unit consists of the following: the latch hooks and latch plate, two upper rods, a coupling, a connector, a drive rod, a yoke, two aluminum blades and two poison blades.

The three latch hooks are attached to a latch plate which makes contact with the dash pot piston. Two 114-25/32-in.-long, 1-in.-diameter 304 stainless steel upper rods fasten to the latch plate and pass through the dash pot by means of ball bushings and seals.

The two rods are welded at their lower end to a 304 stainless steel rod base plate. This plate attaches to a coupling with two 5/8-11 x 1-1/2-in. long socket head cap screws. The coupling is threaded on its lower end with 1-8 NC-2 threads and mates with the transient rod connector. A hex jam-nut is used to prevent loosening of the connector. The connector is drilled and tapped with 3/4-10 NC-2 threads at its lower end and mates with the

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*Trade name for the Sintercast Corporation Aluminum-Boron powder-metallurgy processed material.

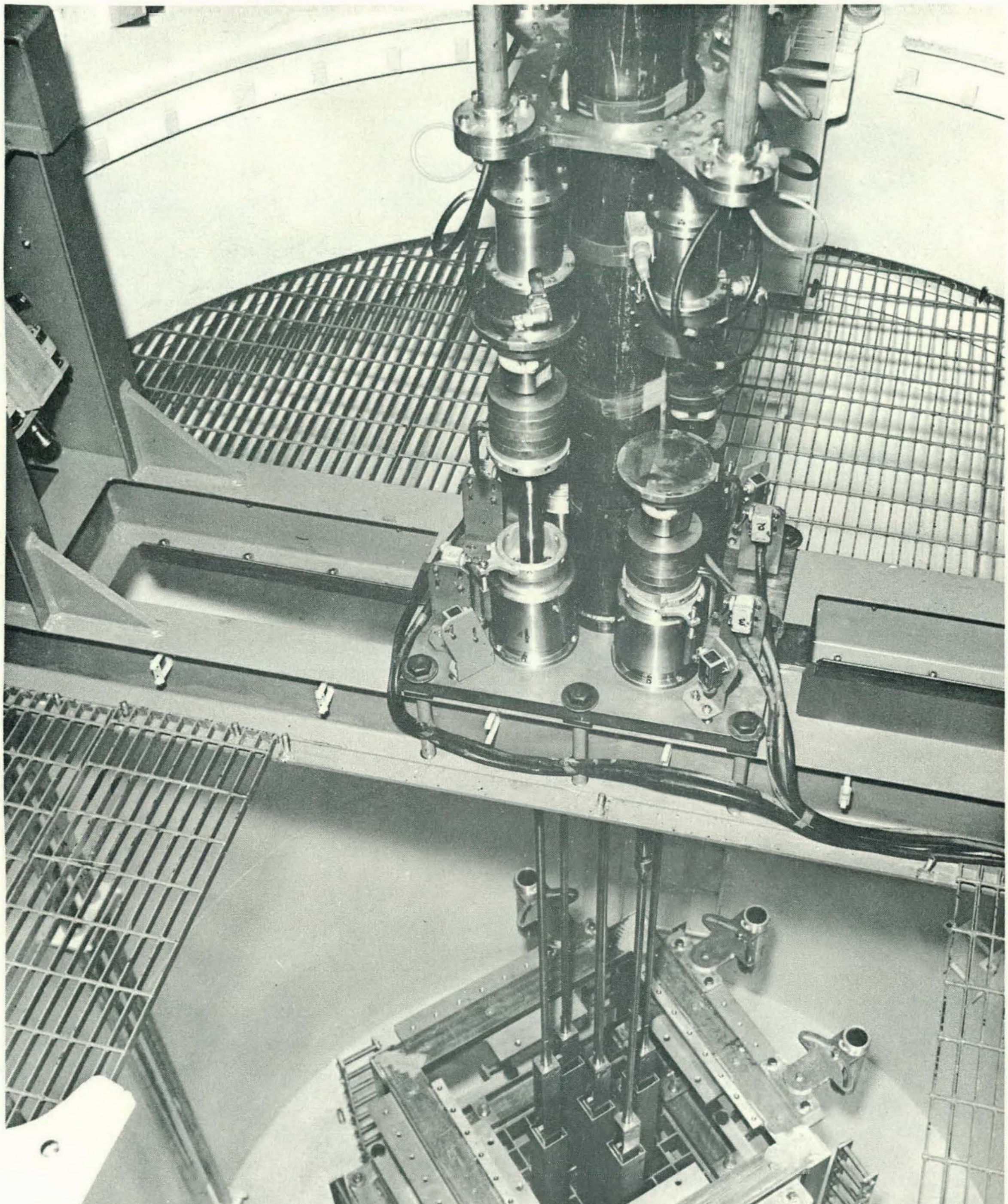

Fig. A-23 - Reactor Core Structure Showing Control and Transient Rods

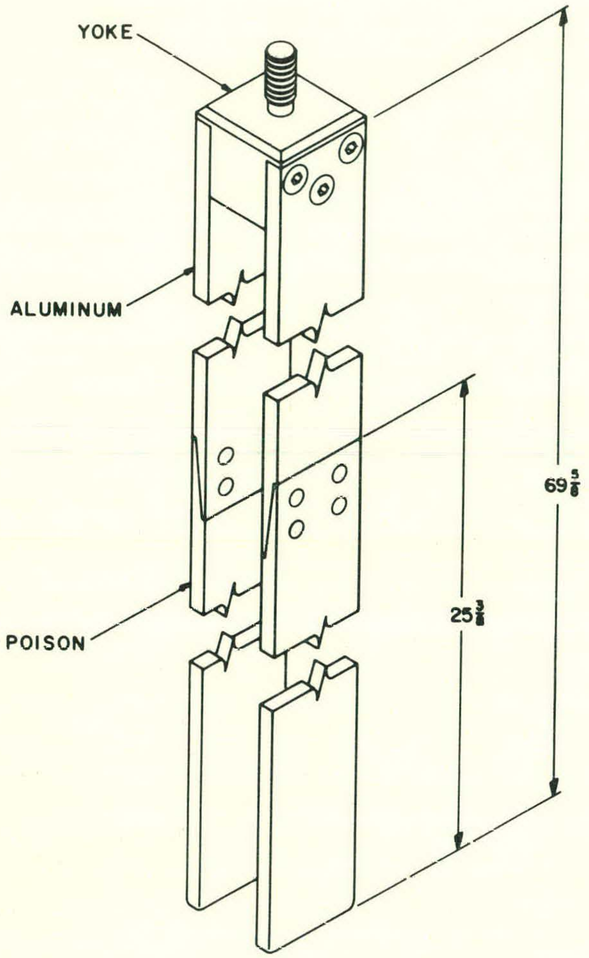

Fig. A-24 - Blade Section of the Transient Rod

55-3/4-in.-long, 1-in.-diameter, 304 stainless steel transient rod drive rod. The drive rod is drilled and tapped with 5/8-11-NC-2 threads at its lower end. This rod mates with the yoke, which is pinned.

The blade section of the transient rod consists of two flat blades secured to the yoke with three 5/16-18NC by 15/16-in.-long flat head socket cap screws which are staked in place. In addition, each transient rod blade has two 5/16-in.-diameter by 3/4-in.-long dowel pins fitted through the yoke.

Each blade of the transient rod consists of two sections: an upper 6061-T6 aluminum section riveted to a lower Binal poison section, in identical manner to the control rods. The blades are each 5/16-in. thick, 2-in. wide and 69-5/8 in. long. The lower Binal section is 25-3/8-in. long.

The poison section of the transient rod is normally positioned below the active region of the core; consequently, the rod is raised to decrease reactivity and lowered to add reactivity.

The transient rod is guided through the reactor core by the transient rod fuel assembly, which is identical to the control rod fuel assembly and is described in Section II-6 of Appendix A. Fig. A-24 shows the blade section of the transient rod.

5. Control System Electrical Circuits

a. Power Supply

The NRTS electrical power standard for applications up to 100 hp is 480-volt, 60-cycle, 3-phase. Three-phase, 480-volt power, which is used only for the rod drive motors, is obtained from the main bus of the reactor building motor control center through a 15 ampere, air-circuit breaker. This bus is fed directly from the Spert-I substation through a 400 ampere air-circuit breaker.

Power for the control relays and other portions of the reactor control system is obtained from control center building single-phase power circuit C40 through a 30 ampere air-circuit breaker. The neutral and power leads are designated ϕ_0 and ϕ_1, respectively. The control system is energized from the circuit breaker through the contacts of a hermetically sealed relay, which is designated the main power relay. Downstream from these contacts, the control power ϕ_1 becomes ϕ_{1c}. Control System power, designated ϕ_{1b}, is obtained from a circuit breaker in the instrument bunker.

As shown in Fig. A-25, the main power key switch on the control console controls the main power relay.

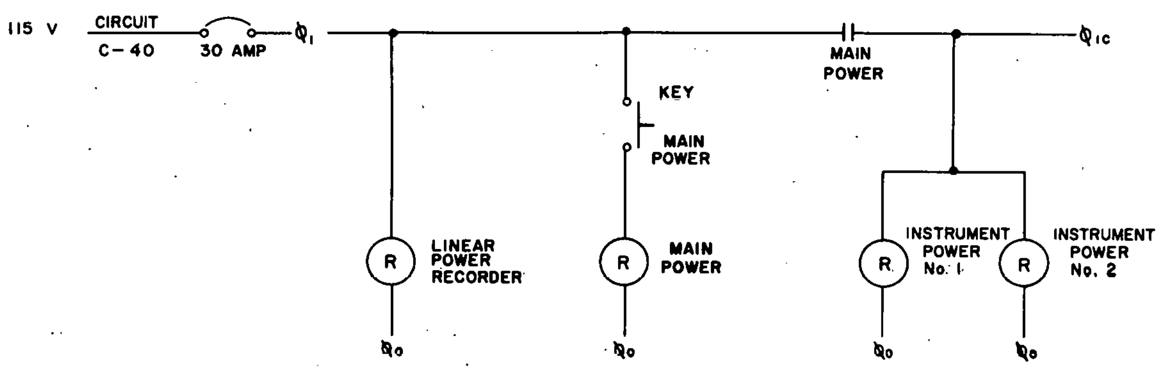

Fig. A-25 - Control System Power Supply Circuit

b. Rod Drive Insert-Withdraw Circuits

Standard NEMA size-00 reversing motor starters are used to control the 1/2 horsepower, 480-volt, 3-phase induction rod drive motors. Electrically, the two starters comprise four units, designated the control rod insert contactor, control rod withdraw contactor, transient rod insert contactor, and transient rod withdraw contactor, (Fig. A-26). Basic control of these contactors is from two pistol-grip, insert-withdraw switches on the control console. The "OFF" positions and control rod insert position are maintained by detents. The control rod withdraw and the transient rod insert and withdraw positions are spring-returned and must be maintained by the operator.

Main control power must be on before any rod movement can be performed. Also, because movement of the rods changes the reactivity of the reactor, the linear power recorder relay must be operating before the rods can be moved in order that any change in reactor power can be recorded.

No inhibitions are included in the control rod 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 a reversing starter has been eliminated as shown

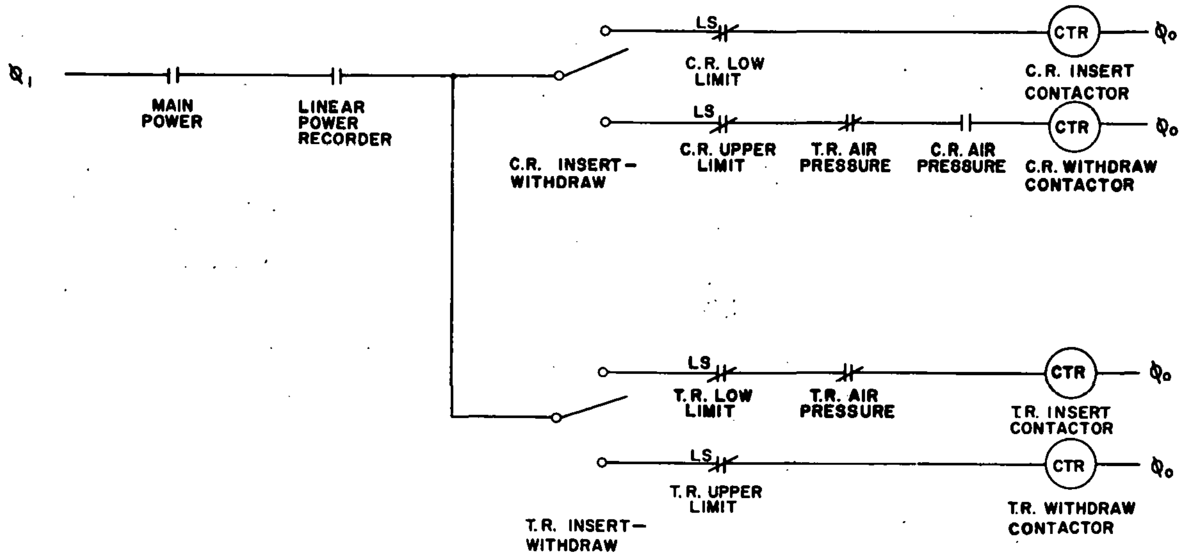

Fig. A-26 - Control System Insert-Withdraw Circuit in Fig. A-26 in order to allow interruption of the rod withdraw circuit and permit actuation of the insert circuit in case of malfunction of the withdraw circuit. Each starter is mechanically interlocked in standard fashion to prevent energizing both coils simultaneously.

Lowering of the transient rod drive is inhibited by the lower limit switch and by contacts of the transient rod air relay.

For normal withdrawal of the transient rod, the rod will be held in contact with its drive by the mechanical latch. When the transient rod is released, its insertion is assisted by an air-operated piston. The piston air is controlled by the transient rod air pressure relay, contacts of which have been included as inhibitions to insertion of the transient rod and to withdrawal of the control rod. These contacts prevent movement of the transient rod and withdrawal of the control rods once "FIRE-AIR" has been applied to the transient rod accelerating piston.

Each control rod is equipped with an air-operated piston which accelerates the rods when they are scrammed. The "SCRAM-AIR" pressure is monitored by the control rod air pressure relay, contacts of which are in the control rod withdraw circuit (Fig. A-26). With normal scram-air pressure (50 psig), the control rod air pressure relay is energized, which allows the control rods to be withdrawn.

Limit switch relay contacts also are included in both the transient rod and control rod withdraw circuits. The "upper limit" switches indicate drive positions corresponding to complete withdrawal of the poison section of the control rods from the reactor core, or to complete insertion of the transient rod poison section into the reactor core. When the upper limit positions are reached, the upper limit switches prevent further withdrawal of the rods.

The control rod drive is not equipped with mechanical latches, but uses electromagnets to couple the control rods to their drives. The magnets are individually controlled so that selective withdrawal of individual control rods is possible.

Because dropping the transient rod prior to the programmed schedule could cause a premature excursion, the transient rod piston fire-air is not applied until immediately before the planned transient.

c. Control Rod Magnet Control Circuits

Four full-wave, single-phase, filtered rectifiers supply current at about 15 volts for the four control rod magnets. The rectifiers are constructed from Offner driver amplifier power supplies. Current to each magnet can be monitored by a galvanometer and adjusted by a rheostat.

The scram circuit is similar to an ordinary motor starter circuit with multiple stop-button stations, except that two parallel relays are used. 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 six locations in the reactor area. Two extension-cord jacks are provided at the control console for additional hand-held scram buttons in the console room. Contacts of timer relay 7 operated by the sequence timer provide for programmed scrams. The pressure safety contacts prevent resetting the scram if the transient rod is armed.

Control rods are selected for withdrawal by closing appropriate "magnet selector" switches on the control console. The control rods must be in contact with the magnets on the drives in order to be withdrawn. Actual energization requires the instrument power to be turned on, 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.

Fig. A-27 shows the control system magnet control circuits.

d. Latch and Piston Air Control Circuits.

Scram-air is applied to each control rod piston by closing the control rod air switch on the control console, which energizes the control rod air relay and air valve. If the piston air pressure is 50 psig or greater, the control rod piston pressure actuated switch will close, energizing the control rod pressure relay. In turn, this relay must be energized in order that fire-air can be applied to the transient rod piston. Fig. A-28 shows the control rod piston air control circuits.

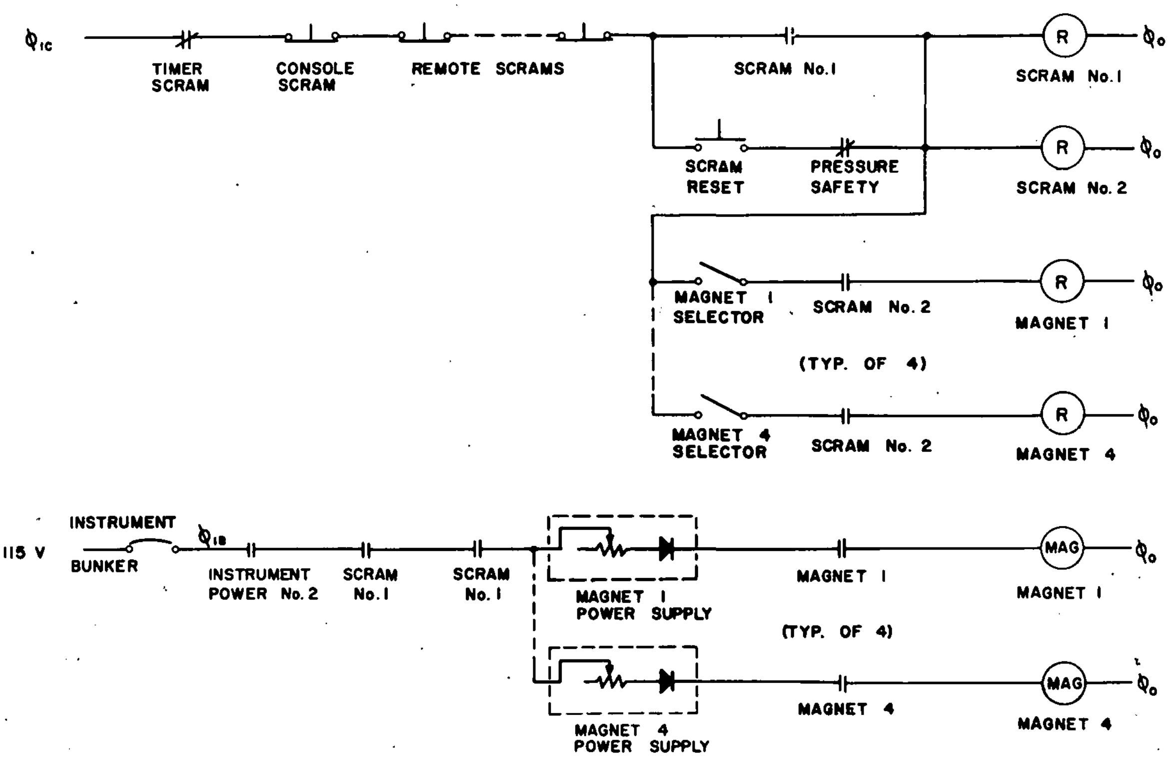

Fig. A-27 - Control System Magnet Control Circuit

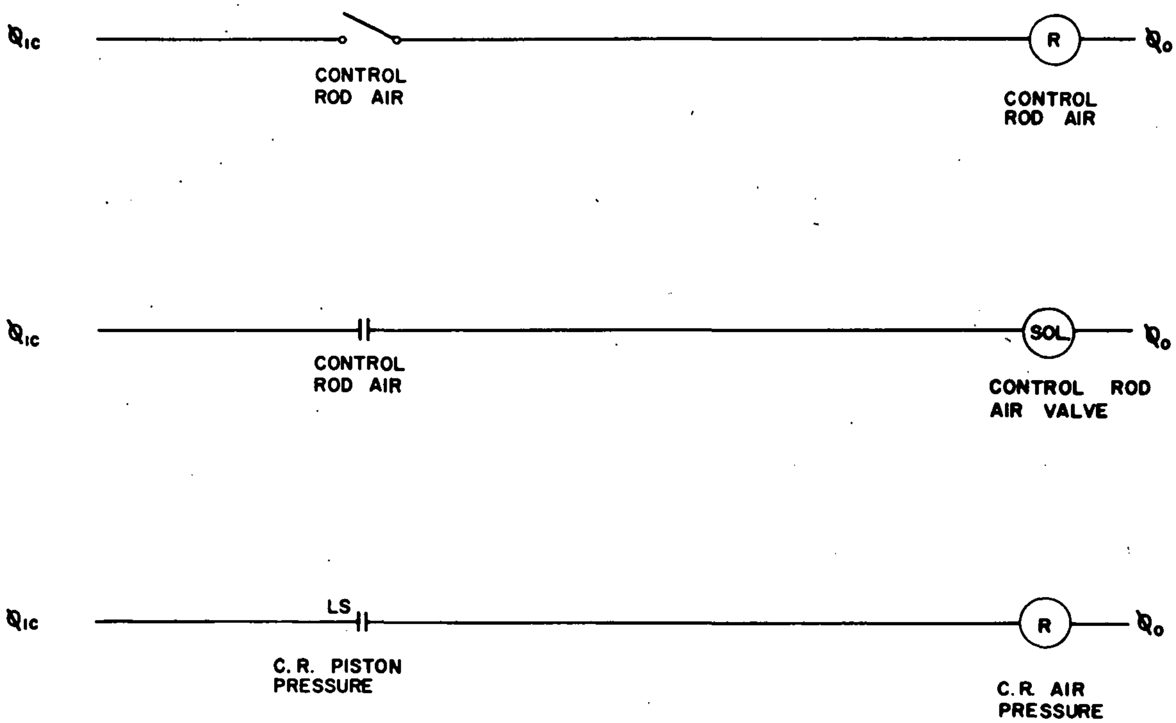

Fig. A-28 - Control Rod Piston Air Control Circuit

To obtain transient rod piston fire-air, control rod piston scram-air must first be on. With the transient rod at contact the transient rod piston air keyswitch is closed, energizing both transient rod pressure relays. These relays then energize the transient rod piston air valve and pressure safety relay as shown in Fig. A-29.

Arming the transient rod prevents any attempted control and transient rod motion except control rod insertion. If it is desired to abort the experiment, the transient rod can be disarmed by opening the transient rod piston air keyswitch.

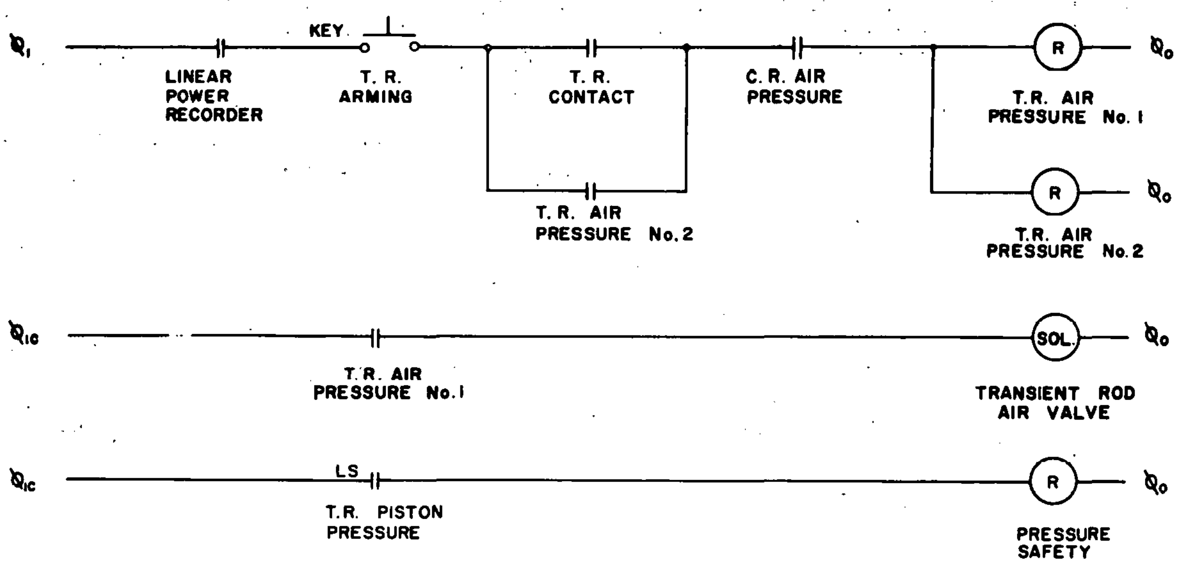

Fig. A-29 - Transient Rod Piston Air Control Circuit

When the transient rod is armed, the "TIMER START" keyswitch can be operated allowing the power excursion to proceed as programmed. As shown in Fig. A-30, the timer contact 6 energizes the "FIRE" relay which, in turn, energizes the transient-rod release piston solenoid valve, opening the latch and dropping the transient rod into the core. The transient rod piston accelerates the transient rod as it falls. Fig. A-30 also shows the transient rod latch control circuit.

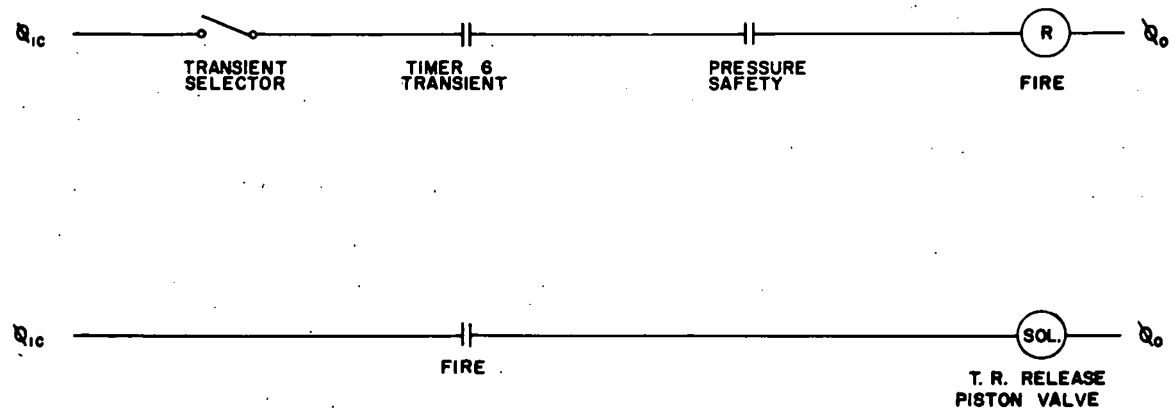

Fig. A-30 - Transient Rod Latch Control Circuit

e. Sequence Timer Circuits

The Spert I control system is provided with a Multiflex timer, manufactured by Eagle Signal Corporation, for sequence programming of experiments. The timer has a maximum range of 30 seconds and settings can be made with accuracy within 0.25% of full scale.

The basic timer circuit is shown at the top of Fig. A-31. This is an elaboration of what is referred to by the manufacturer as the "no voltage reset arrangement", protected against automatic restarting. Energizing the clutch solenoid of the timer by means of the "START" switch engages a clutch and lowers the contact trip bars to ride on a sliding plate. The synchronous motor then drives the sliding plate 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

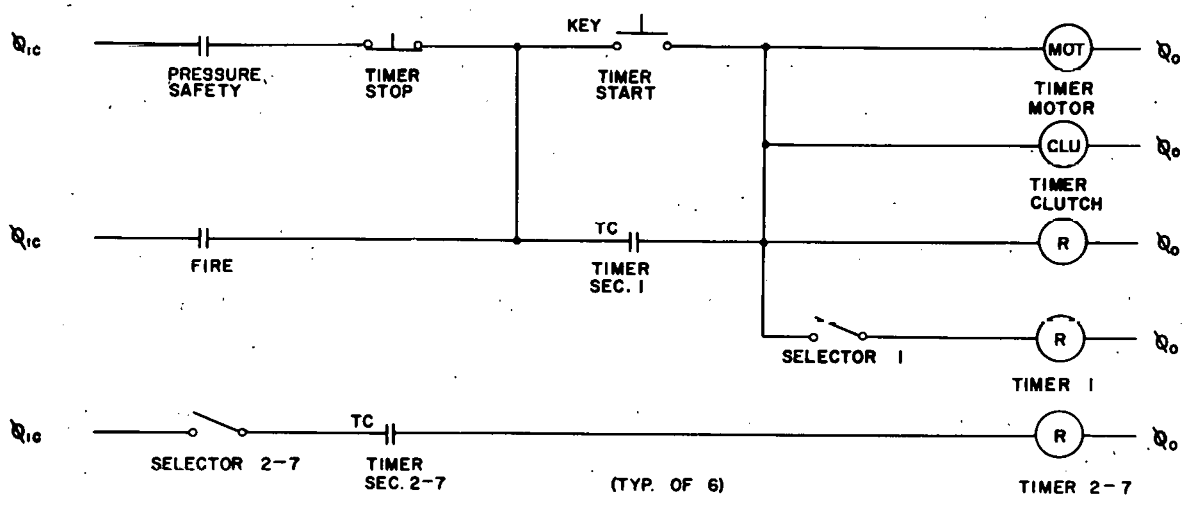

Fig. A-31 - Control System Sequence Timer Circuit disengages the clutch and raises the trip bars, allowing the sliding plate to reset by spring action to its original position. Section-one timer contacts are used as a holding circuit, enabling a momentary switch to be used for starting so that the timer does not repeat its cycle automatically after resetting. Used in this manner, the section-one closing contact always must operate when the clutch is energized, the section-one opening contact determining the length of the timer cycle.

Apparatus is not operated by the timer directly, but by seven timer-controlled relays. These relays are designated timer 1 relay, timer 2 relay, etc., and are controlled through selector switches on the control console. Timer relay circuits 2 through 5 and 7 are shown at the bottom of Fig. A-31. Timer 6 relay is the "FIRE" relay shown in Fig. A-30. Timer 1 relay circuit is shown with the basic timer circuit because of the internal use of the section-one contacts. Contacts of the "FIRE" (timer 6) relay are used to bypass, and thus prevent accidental functioning of the timer stop button during programmed transient power excursions.

f. Rod-Drive-Speed Control Circuit

The variable speed transmissions in the Spert I rod drives are equipped with reversible pilot motors for speed changing. These pilot motors are controlled by three-position toggle switches on the control console. Coupled to the speed-changing mechanism of the control rod drive transmission is a self-synchronous motion transmitter which drives a similar unit and a Veeder-Root digital counter located on the control console. Because the transmission speed ratio is not a linear function of the pilot motor shaft position no attempt has been made to choose gear ratios to make the digital counter direct reading. Calibration curves are required to interpret counter readings. Instead of a selsyn transmitter, the transient rod drive transmission has a potentiometer which can be used with a voltmeter to indicate the transient rod drive speed.

The speed control circuits with their console-mounted operating switches are shown in Fig. A-32.

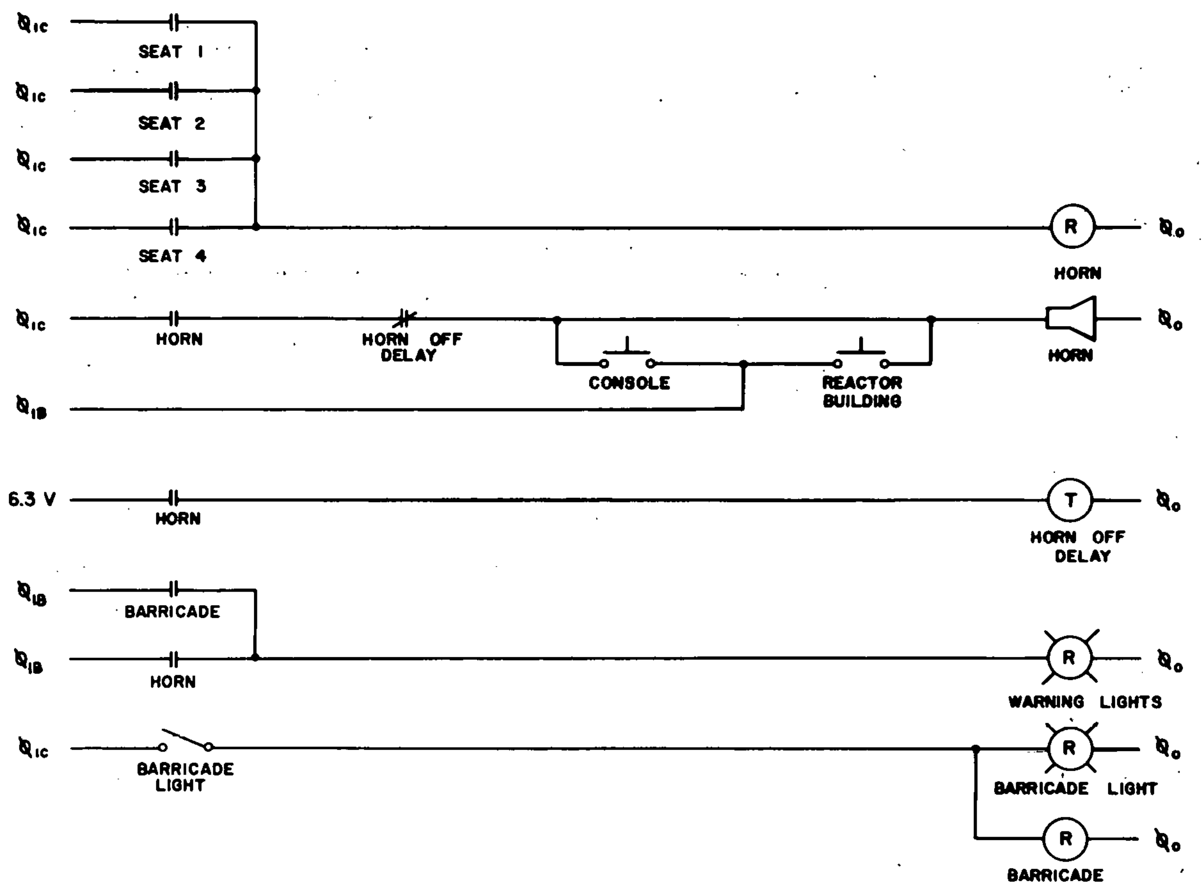

Fig. A-32 - Rod Drive Speed Control Circuits

g. Rod Drive Warning Horn and Lights

The reactor area has been provided with warning lights and a horn which are part of the reactor control system. Whenever control power is on (as it must be to operate the rod drives) and any rod is raised from seat position, the warning system is energized. The circuitry is shown in Fig. A-33.

The warning horn, mounted on the roof of the reactor building, is operated from instrument bunker power by either the horn relay or by the horn buttons, one on the control console, the other in the reactor building. When actuated by the horn relay, the horn is operated for 15 seconds by a time-delay relay.

During operation of the reactor, a signal from any seat switch, indicating drive motion, energizes the horn relay and red warning lights in the reactor area. Provisions are made such that the warning lights are

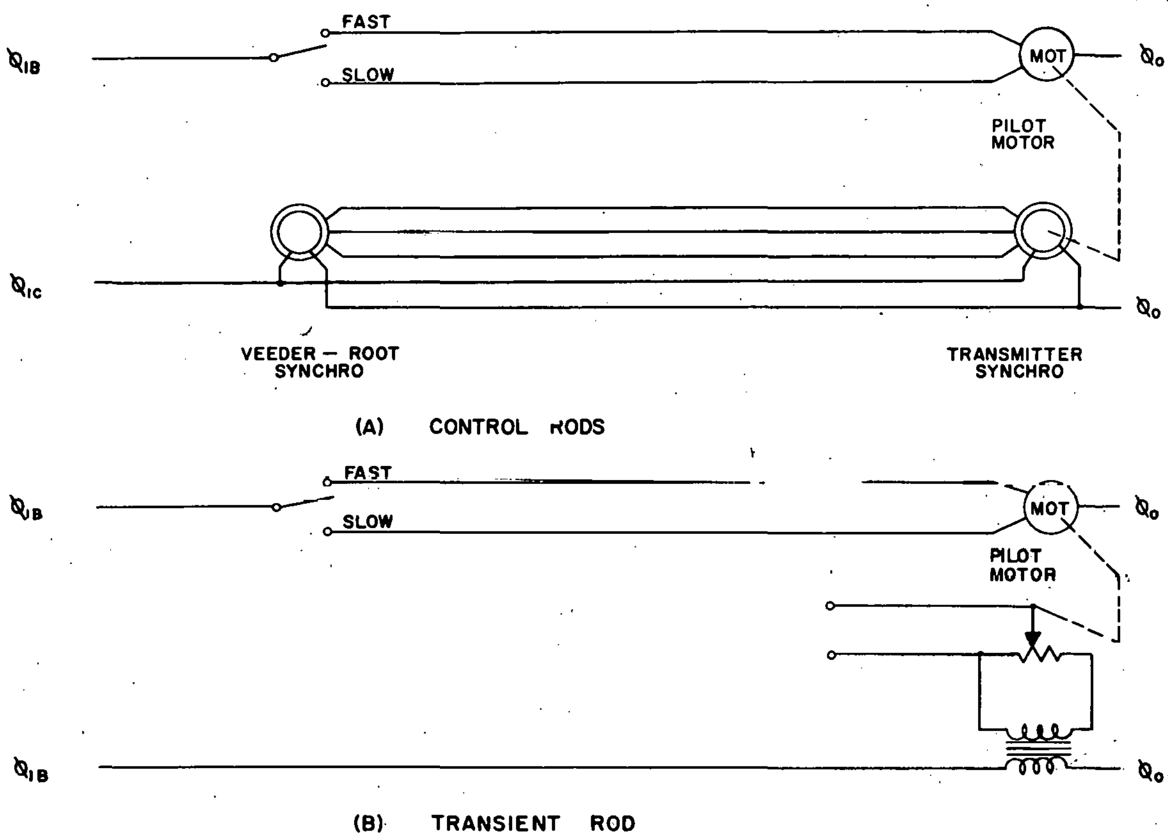

Fig. A-33 - Warning Lights and Horn Circuits

not energized by the manual horn buttons, and, as already mentioned, the horn operates from the horn relay for only 15 seconds. A red warning light is located at the barricade near the control center on the Spert I access road. This light is controlled from the control console, and is turned on after all personnel have been evacuated from the reactor area and the barricade has been placed across the access road. Turning on the barricade warning light also energizes the barricade relay which turns on all Spert I reactor area warning lights

h. Rod-Drive Sensing-Switch Circuits

Circuits for magnet contact switches, rod seat switches and shock absorber extension lights are shown in Fig. A-34. Oil-filled shock absorbers are provided to decelerate the rods when scram occurs or when a step transient is initiated. As the rods are withdrawn, the shock absorbers extend upward several inches in order that the shock of

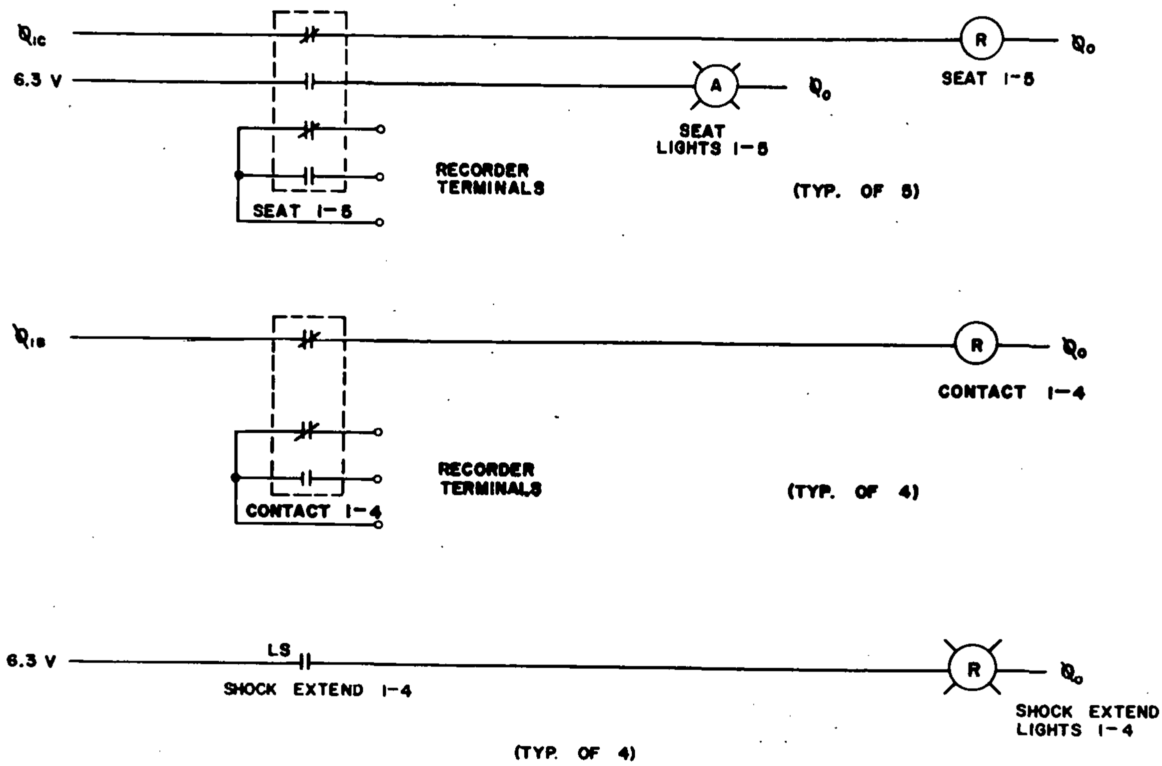

Fig. A-34 - Rod Drive Sensing Switch Circuits

subsequent dropping of the rods may be properly dissipated. Control console lights indicate the position of the shock absorber extensions.

The seat switches operate control relays and control console indicator lights. 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, 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.

Magnet contact switch circuits are identical to seat switch circuits except that no control relay is provided for the transient rod. Also, no continuity testing feature is provided. Rod drive limit switches (not shown) merely operate the control console indicating lights, no provision being necessary for the recording of limit switch signals.

6. Reactor Control Console

Fig. A-35 is a front view of the reactor console. For the convenience of the operator in distinguishing the functions of the various controls on the reactor console, the lower section is divided into seven panels. From left to right, the first three panels contain controls for the four neutron pulse-counting systems (Fig. A-36), a log-rate meter, and three 26-channel recording oscillographs (Fig. A-37). The fourth panel, Fig. A-38, contains the reactor vessel bulk-water-temperature recorder, an auxiliary recorder which can be used to record power, temperature or pressure, and radiation-level meters, which monitor the reactor building doorways, the area directly above the reactor vessel, and the outside atmosphere near the Health Physics guard house.

The fifth panel, in addition to the logarithmic power recorder (Figs. A-38 and A-39), contains the controls and informative lights associated with the auxiliary facilities. The controls on this panel consist of ON-OFF push button controls for the sump pump, electric mixer, electric immersion heaters, the reactor-fill pump and inlet valve, and the reactor tank drain valve. Red and green lights indicate the operational status of each item of equipment. The reactor tank liquid-level indicator is located below the auxiliary equipment controls.

Immediately above the auxiliary equipment controls are two rows of toggle switches (Fig. A-39), which control 12 electrical outlets in the reactor building and the high voltage supplies to each of the four pulse-counting systems. A light adjacent to each switch indicates when the switch is in the ON position.

The lower section of the sixth panel (Fig. A-40), contains the control switches, indicators, etc., associated with operation of the reactor, and the upper section (Fig. A-41), contains a closed-circuit television receiver, with controls for remote television camera operation.

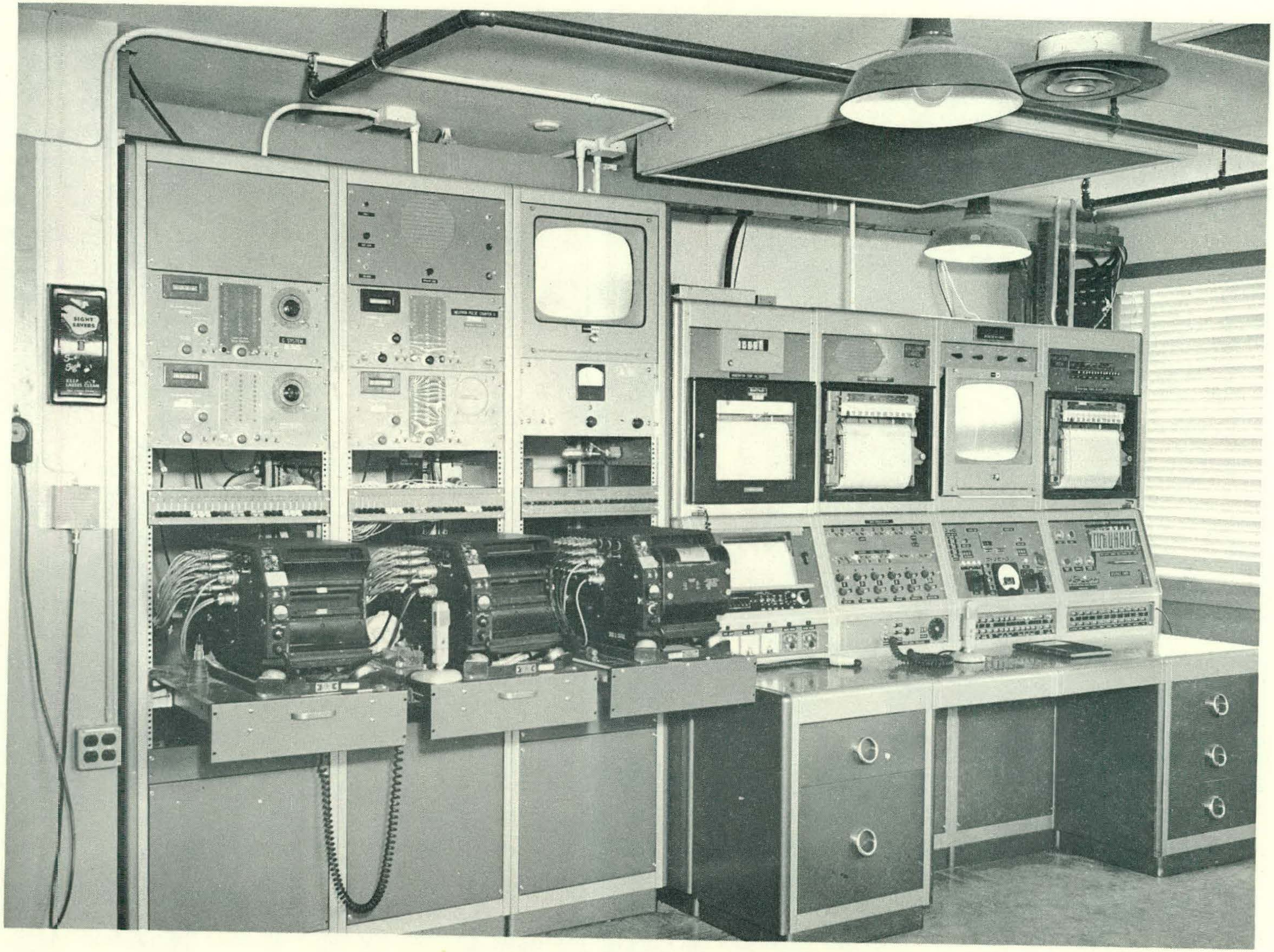

Fig. A-35 - Reactor Control Console

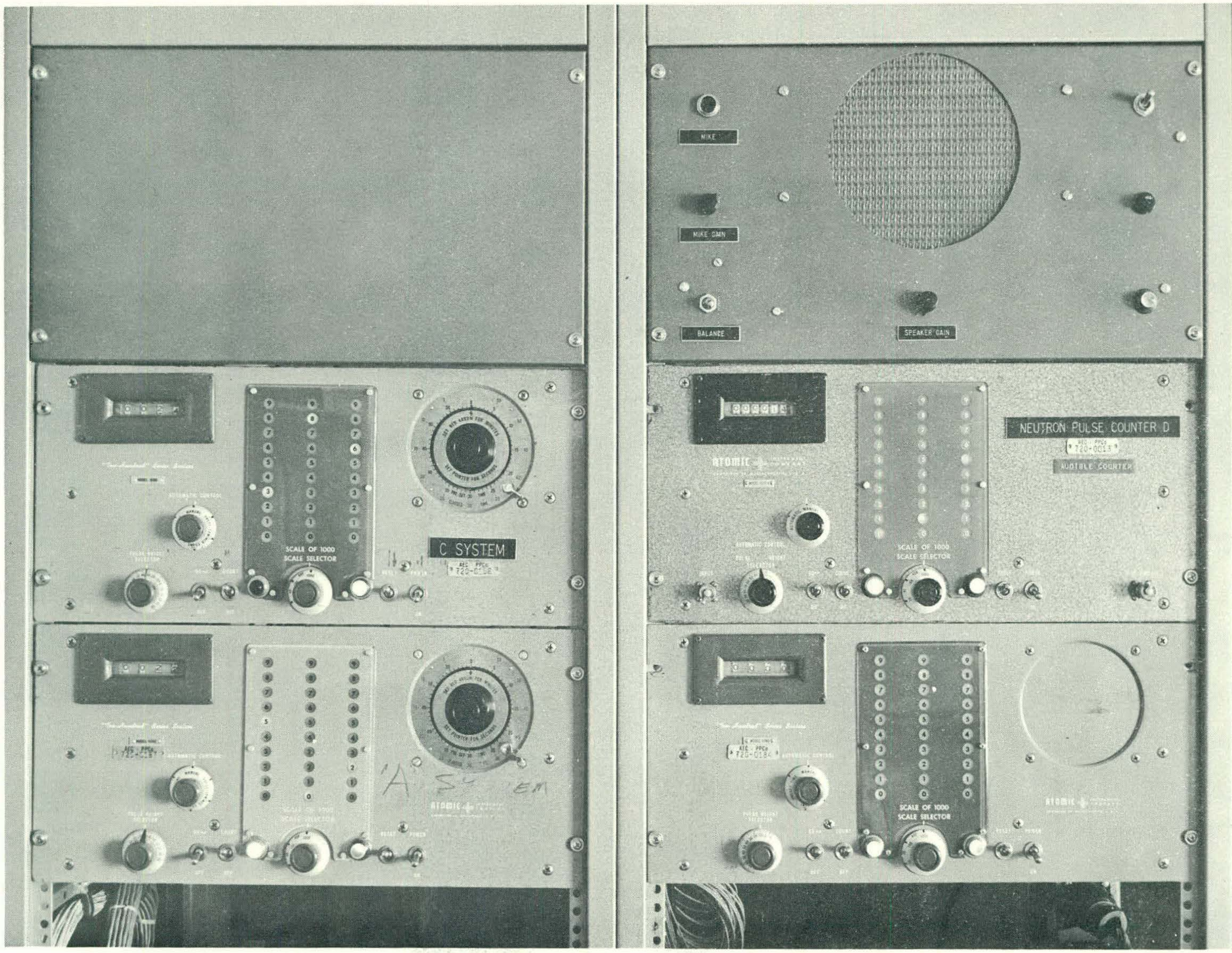

Fig. A-36 - Pulse Counters

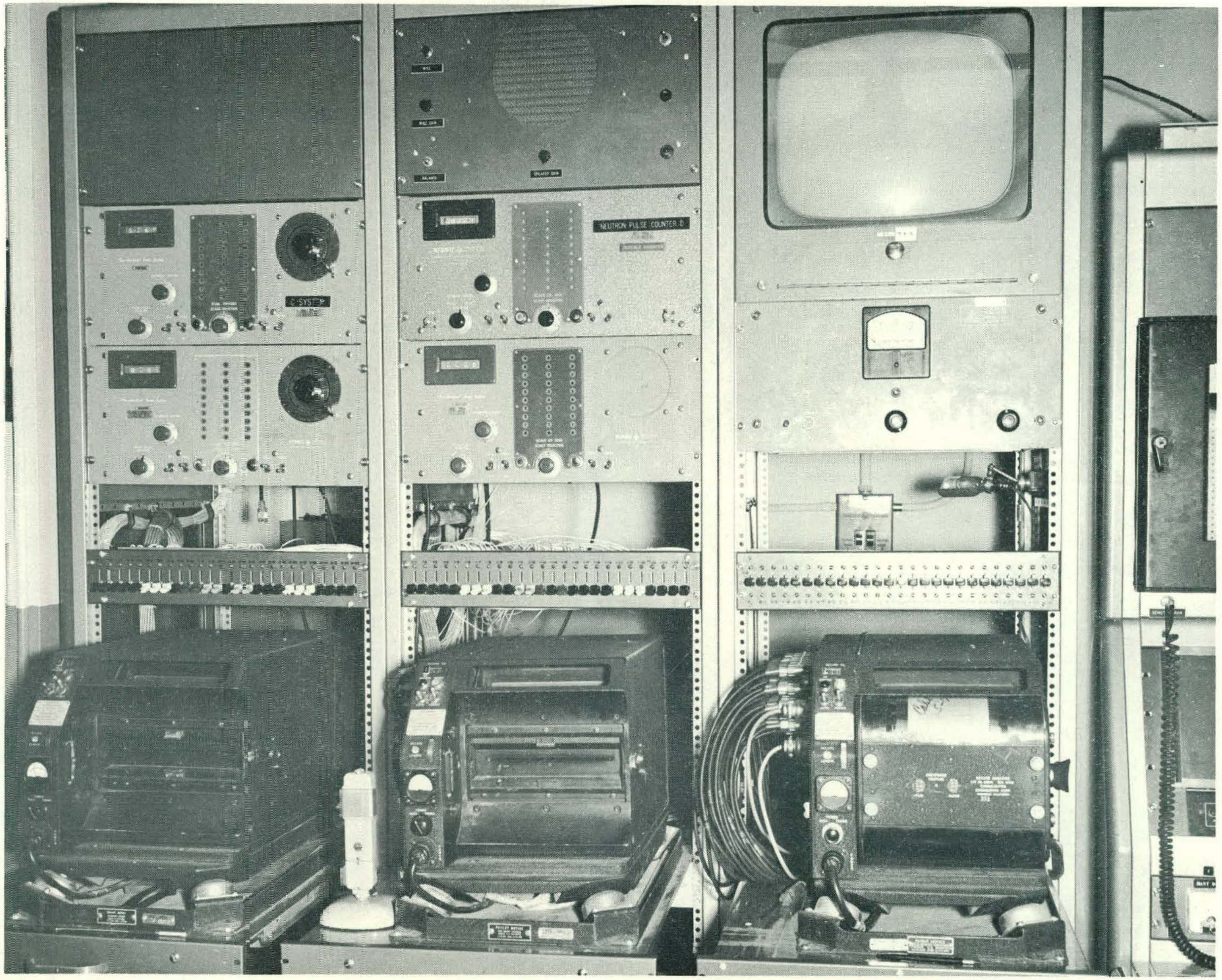

Fig. A-37 - Recording Oscillograph

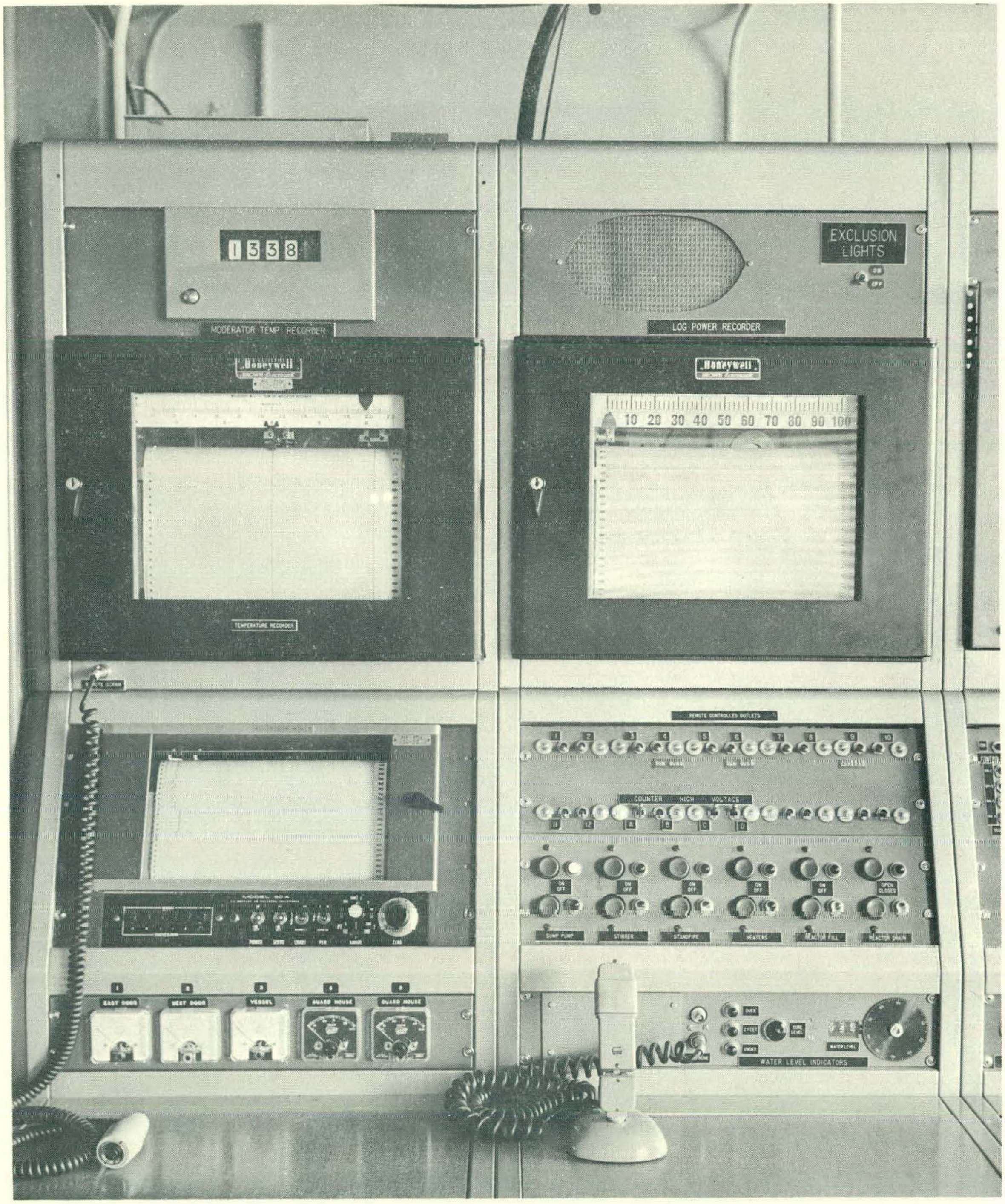

Fig. A-38 - Reactor Control Console - Panels 4 and 5

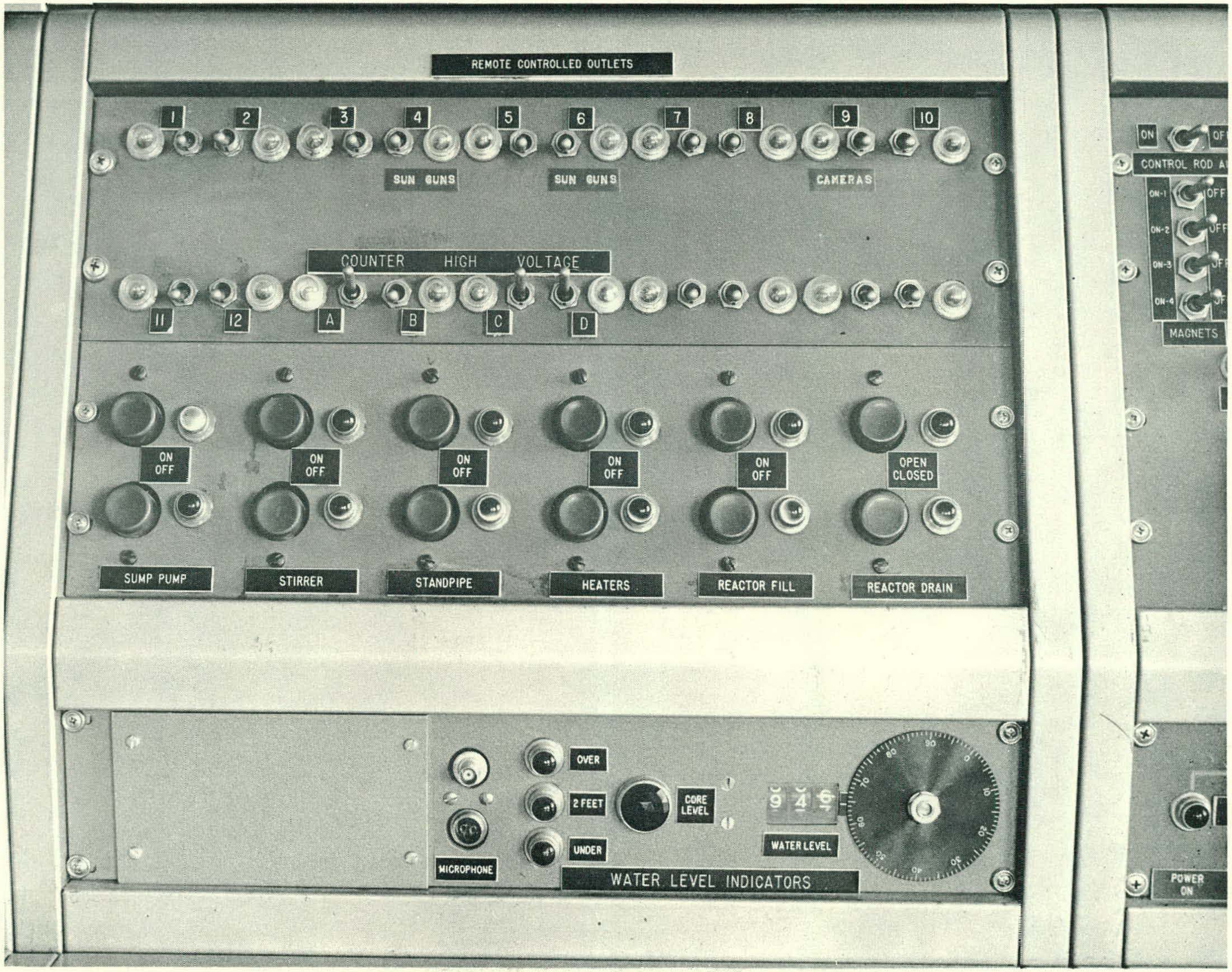

Fig. A-39 - Reactor Control Console - Panel 5

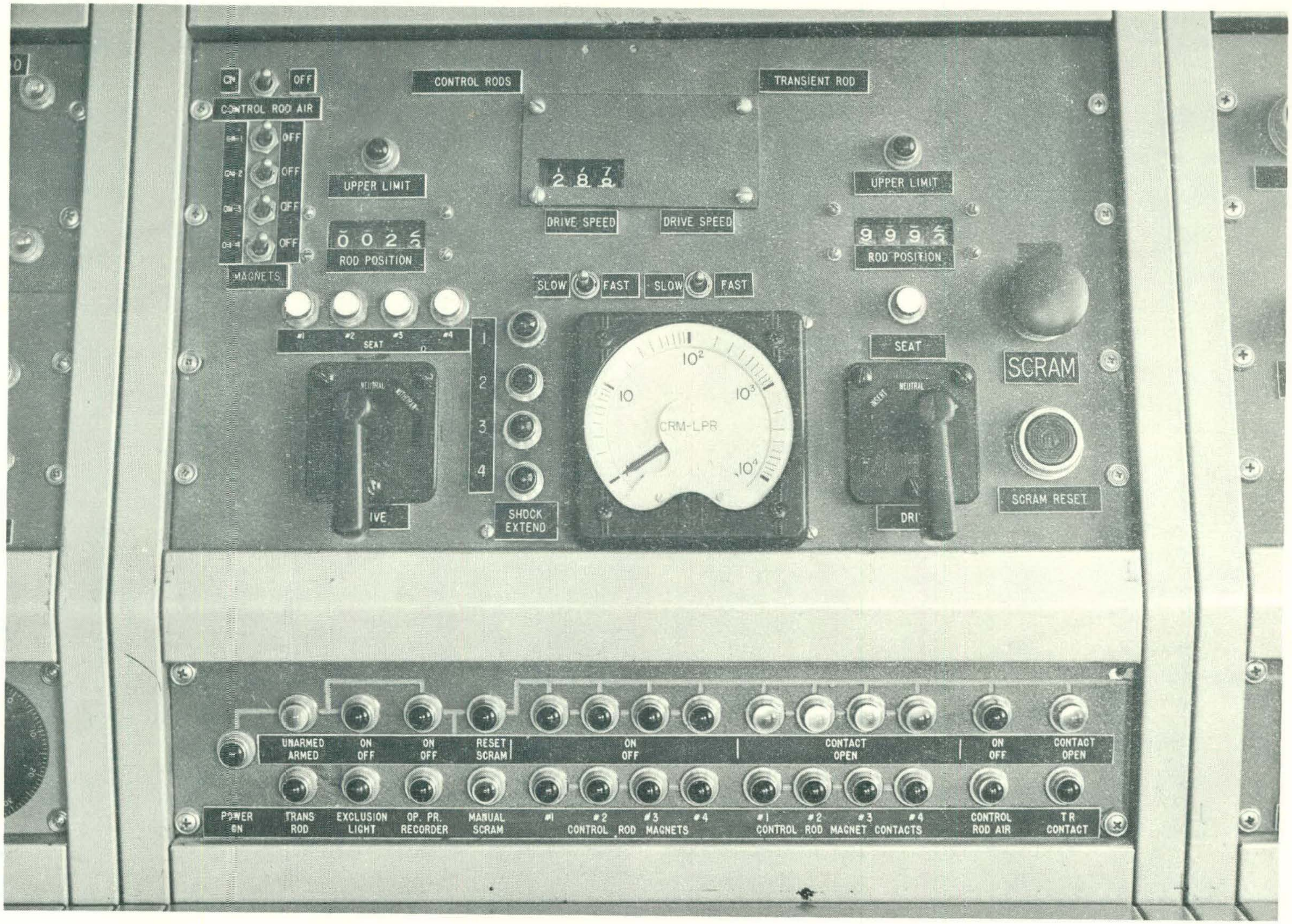

Fig. A-40 - Reactor Control Console - Panel 6

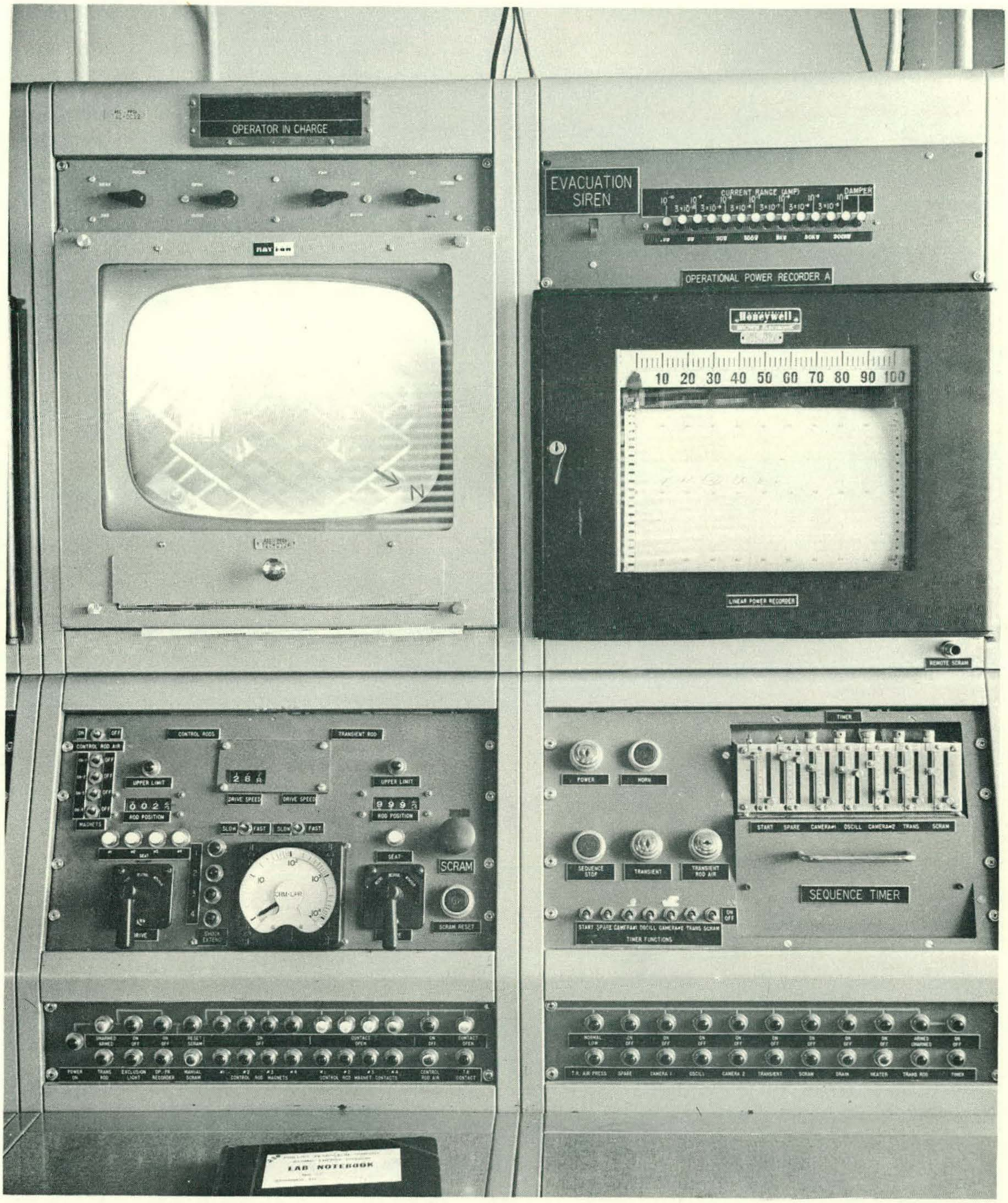

Fig. A-41 - Reactor Control Console - Panels 6 and 7
(The "evacuation siren" switch shown on the panel is no longer functional and has been removed)

Two three-position pistol-grip switches, located on the left and right of center of the panel (Fig. A-40), are provided for controlling the movement of the control rods and the transient rod, respectively. Each switch is equipped with Withdraw-Neutral-Insert positions. Spring returns on the switches return the control rod switch from the Withdraw to Neutral positions and the transient rod switch from Insert to Neutral. Control rod Insert is a maintained position. Although only gang-operation of the control rod drives is possible, individual rod insertion or withdrawal may be achieved by controlling the magnet current supplied to the individual magnets. Four toggle switches, one for each of the control rods, are located in the upper left corner of the panel; these control the supply of current to the individual magnets. Digital indicators, located immediately above the pistol-grip switches, continuously indicate the rod drive positions to the nearest 0.01 inch. A fifth toggle switch is provided to control the air supply to the control rod air piston accelerators. Withdrawal of the control rods is not possible without this air supply on.

A 4-in. dial count-rate meter is located in the geometric center of the panel, and provides the operator with visual indication of the power level during start-up operations.

Immediately above the count-rate meter is a three-place digital counter, indicating the drive speed of the control rod drives. The transient rod drive speed, although variable, is not monitored at the control console. Selection of the desired drive speed is achieved by toggle switches located directly beneath the speed-indicating digital counter.

A manual scram button and a scram reset button are located at the right-hand edge of the lower section of the panel. Indicating lights on the panel include the seat lights for each of the four control rods and the transient rod, the upper limit lights for the control rod and transient rod drives, and the shock extend light for each of the control rods.

A small procedure panel (Fig. A-40) equipped with red and green lights and located below the control panel supplies the following information to the operator:

1. Power on
2. Transient rod - armed or unarmed
3. Exclusion light - on or off
4. Operational power recorder - on or off
5. Scram - scram or reset
6. Control rod magnets (1 - 4) - on or off
7. Control rod contact (1 - 4) - contact or open
8. Control rod air - on or off
9. Transient rod contact - contact or open.

The right-hand panel of the console (Fig. A-41) contains the linear power recorder and its range selector in the upper portion, the sequence timer, and in the lower portion, informative lights pertinent to initiating a transient (see Fig. A-42).

A principal feature of this panel is the sequence timer which, in accordance with a pre-set schedule, starts cameras and recording equipment, initiates ejection of the transient rod, and at the desired time inserts the control rods and shuts off experimental equipment. In addition to the timer, the panel is equipped with key-interlock push buttons for arming the transient rod; i.e., supplying air to the air piston used to accelerate the rod, and for starting the reactor transient; i.e., starting the sequence timer.

If for any reason the reactor operator desires to interrupt the transient experiment after pushing the transient start button, a sequence stop button is provided which will stop the sequence timer if the transient rod has not yet been released. However, if the transient rod has been released, pushing the sequence stop button will not disrupt the programmed control rod scram nor will it prevent an immediate manual scram.

Toggle switches and indicating lights have been provided for the control of individual items of equipment such as cameras, recorders,

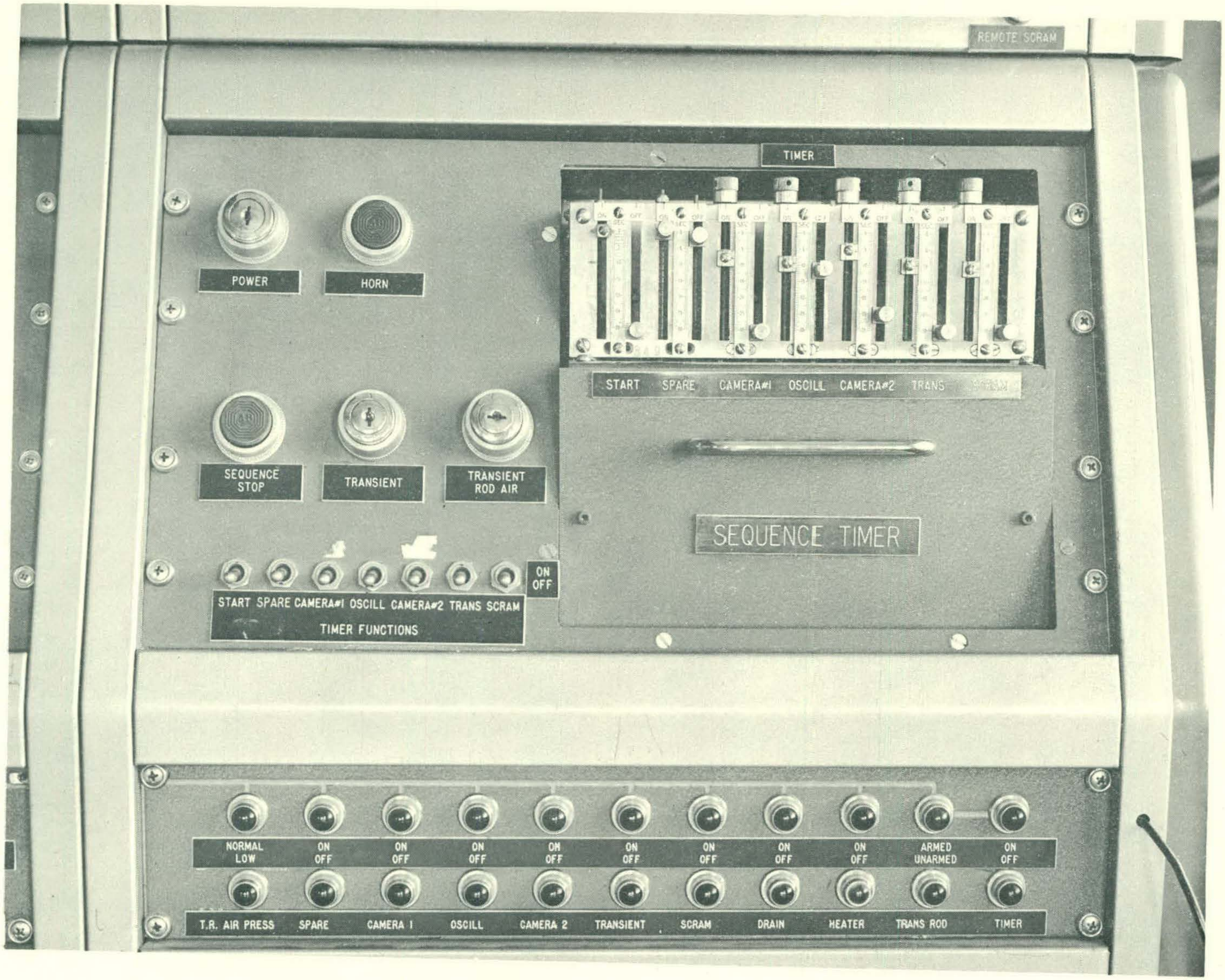

Fig. A-42 - Reactor Control Console - Panel 7

etc., connected to the sequence timer. Individual experimental equipment when not in use may be disconnected from the timer circuit by means of the toggle switch. Indicating lights inform the operator which pieces of experimental equipment are in the timer circuit.

IV Operational Instrumentation

1. General

The instrumentation for the Spert I reactor is divided into two categories: operational or reactor control instrumentation, and transient or experimental instrumentation. This section of the report deals with the detailed operational instrumentation; details of the transient instrumentation and associated problems are given in Section III of the body of the report.

Included in the operational instrumentation are the systems used to determine the nuclear status of the Spert I reactor, the water-temperature and -level indicators, and certain Health Physics equipment. A schematic block diagram of the operational instrumentation is shown in Fig. A-43.

2. Nuclear Operational Instrumentation

The nuclear operational instrumentation for the Spert I reactor includes pulse-neutron and ion-chamber systems.

The pulse systems are the primary neutron detectors in use during the initial loading of the reactor and in the approach to critical, monitoring the power level until it is high enough to be detected by the ionization chambers. The power range covered by the pulse counters extends from a source level of about 5 milliwatts to a level of the order of 10 watts. During the initial fuel loading operation, at least two pulse systems must be operating, with a third system in use as backup to provide continuous information during the time counting rates are being recorded on the other pulse systems. A log count-rate meter is included in this backup system. A neutron-monitoring system, which is audible in both the instrument bunker and reactor building is connected to the most sensitive pulse system. During any nuclear operation, at least two pulse systems must be in operation.

In the power range from about 1 w to 10 Mw, the power level is determined by two B^{10}-lined gamma-compensated ionization chambers, which are connected to an operational linear recorder and an operational log recorder. For all reactor operation, both linear and log systems must be in operation.

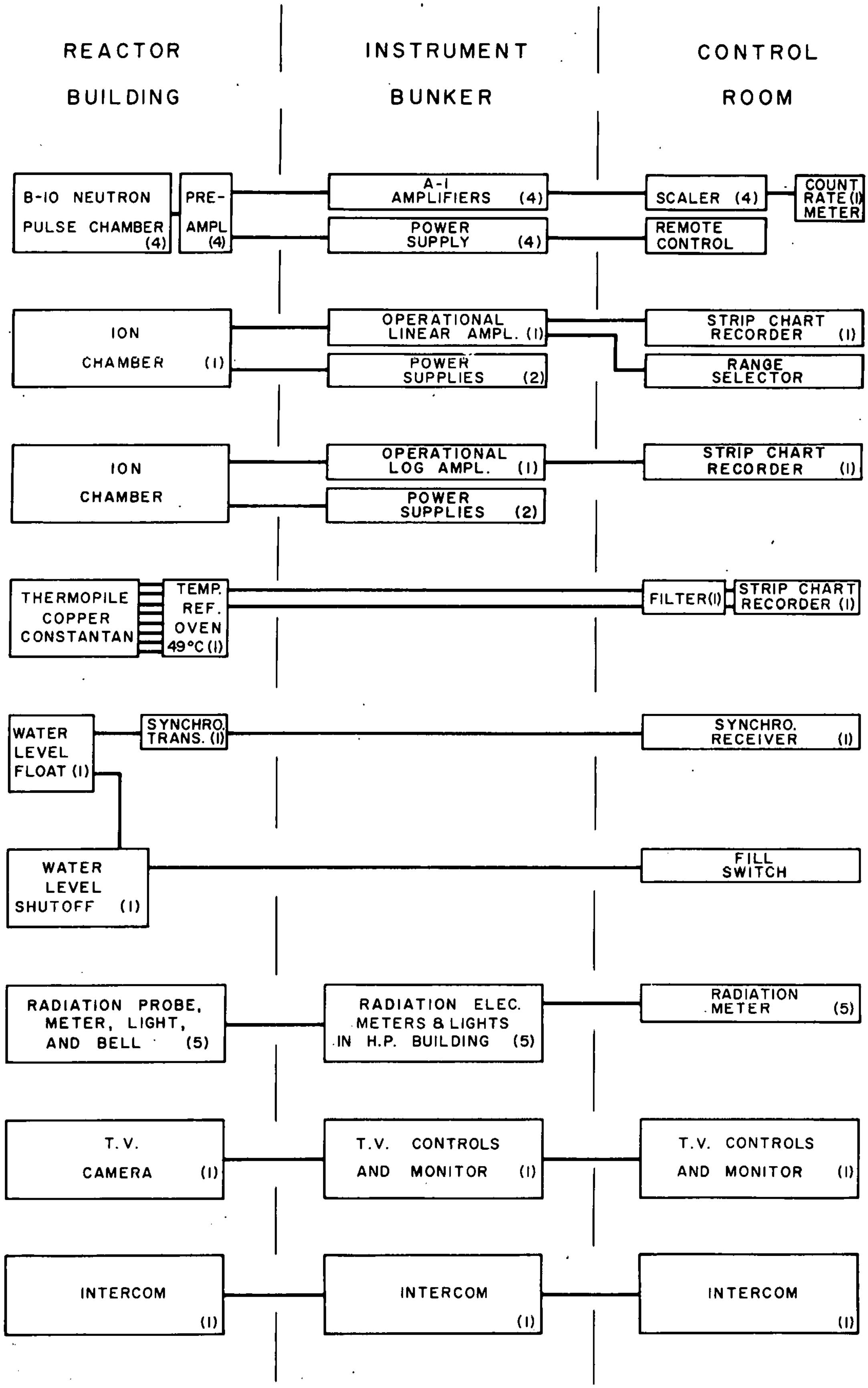

Fig. A-43 - Operational Instrumentation Block Diagram

a. Neutron Pulse-Counter System

Fig. A-44 illustrates a block diagram of a typical pulse-counting channel, including the proportional counter neutron detector, power supplies, preamplifier, amplifier, scaler, and count-rate meter.

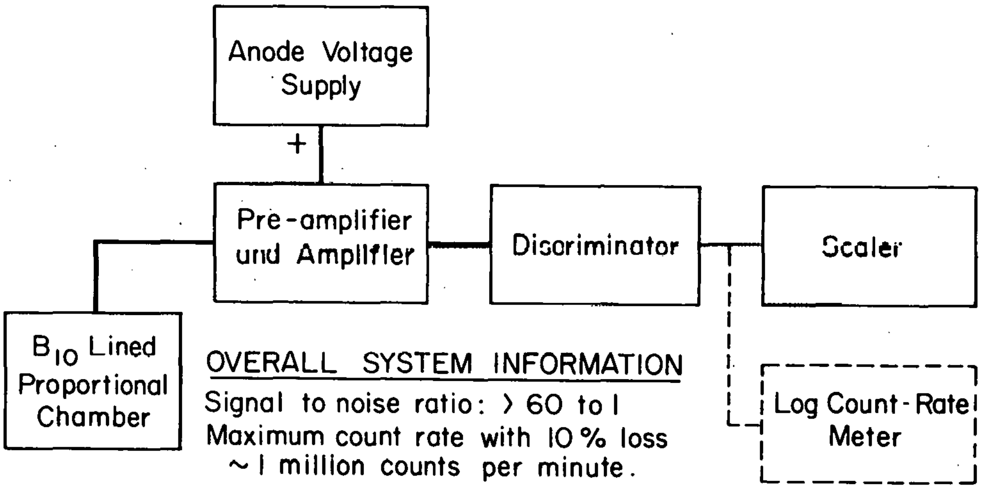

Fig. A-44 - Block Diagram of a Typical Pulse-Counting Channel

(1) Proportional Counter

The B^{10}-lined proportional counters are gas-filled with 95% O_2 and 5% CO_2, at a pressure of 20 cm of mercury. The voltage applied to the counters is approximately 900 volts and they are specified to have a neutron exposure lifetime of 10^{16} nvt. A typical voltage-dependent sensitivity curve is shown in Fig. A-45.

Counters from three manufacturers are used: General Electric, Reuter Stokes, and Westinghouse. The proportional counters manufactured by General Electric and Reuter Stokes both have a neutron sensitivity of 4 cps/nv; the Westinghouse counter has a neutron sensitivity of 3 cps/nv. The relative counter sensitivity was determined for a given source and set of amplifier parameters.

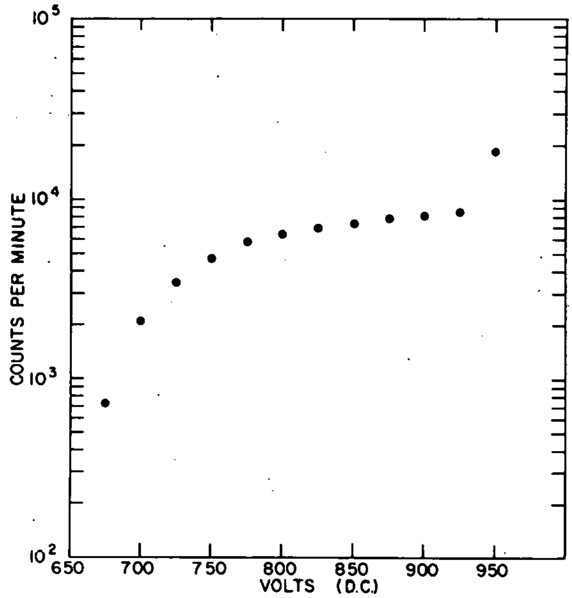

Fig. A-45 - Typical Voltage-Dependent Sensitivity Curve for Pulse Counters

Waterproof aluminum containers are used to house and position the proportional counters inside the reactor tank near the core. A polyethylene sleeve insert is used to insulate the counter from the aluminum container. The containers are mounted on brackets near the edge of the core structure or in the corner of the grid in specially designed dummy fuel cans.

(2) Preamplifier

For two of the systems, "A" and "B", Baird Atomic vacuum tube preamplifiers are used. These preamplifiers are located in the instrument bunker approximately sixty feet from the ion chamber near the reactor. The system was adapted to this usage by modifying the input to the preamplifier to permit application of the driven-shield technique.

The other two systems, "C" and "D", use Spert-designed and - constructed transistorized preamplifiers, incorporating a technique in which the cable is always terminated in its characteristic impedance. This permits essentially any length of cable to be used between the proportional counter and the preamplifier without pulse distortion. The width of the pulse from the transistorized preamplifier is much narrower than the pulse from a vacuum tube preamplifier, which allows a higher counting rate without saturation.

(3) Linear Amplifier

The Baird Atomic, Inc., linear amplifiers used are tube-type amplifiers which have been used previously at Spert. The discriminated output is transmitted to the scalers.

(4) Scaler

The scalers used are manufactured by Baird Atomic, Inc. These are vacuum-tube type, mechanical register scalers which have been used previously at Spert. The mechanical registers are the limiting factor in the maximum count rate which the pulse systems are capable of counting without error. If a higher count-rate is desired an available seven-decade transistor scaler without a mechanical register may be used. The frequency rating of the input decade counting is 1 Mc and all the other decade-counting strips are 100 kc.

(5) Count Rate Meter

The count-rate meter used for giving a rapid indication of radiation intensity is a transistorized log-rate meter having a range extending from 10 cpm to greater than 10^6 cpm. This meter was manufactured by Tracerlab, Inc.

b. Ionization Chamber System

Fig. A-46 illustrates a block diagram of a typical ion chamber power-level channel, including the ion chamber detector, power supplies, the electrometer ammeter, and recorders.

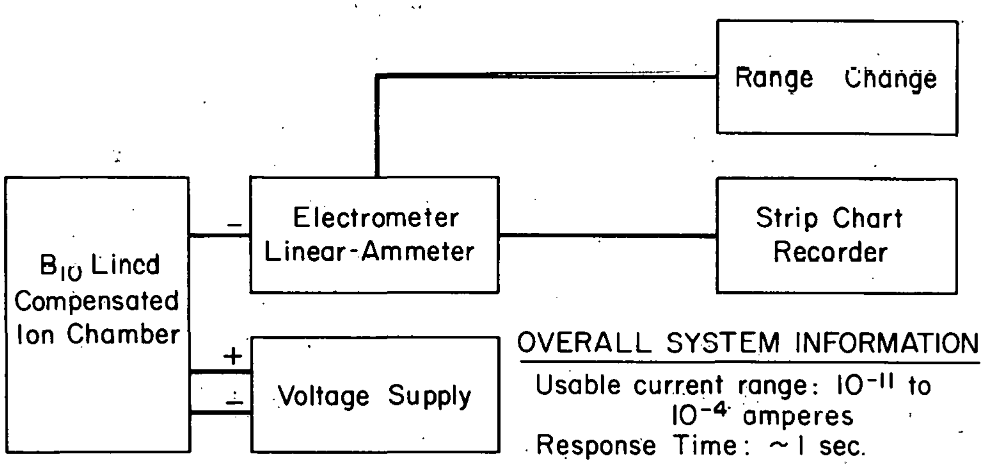

Fig. A-46 - Block Diagram of a Typical Ion Chamber Linear Power-Level Channel

(1) Ionization Chamber

The ion chambers used have typical neutron sensitivities of approximately 10^{-14} amps/nv and gamma sensitivities of approximately 10^{-11} amps/R/hr. A typical neutron sensitivity curve is shown in Fig. A-47. The operational linear chamber is gamma compensated.

The ion chambers are housed in heavy-wall aluminum containers designed to withstand an 8000 psi sustained pressure pulse. The containers can be placed in the lower grid structure or positioned by means of ladders on the core support legs.

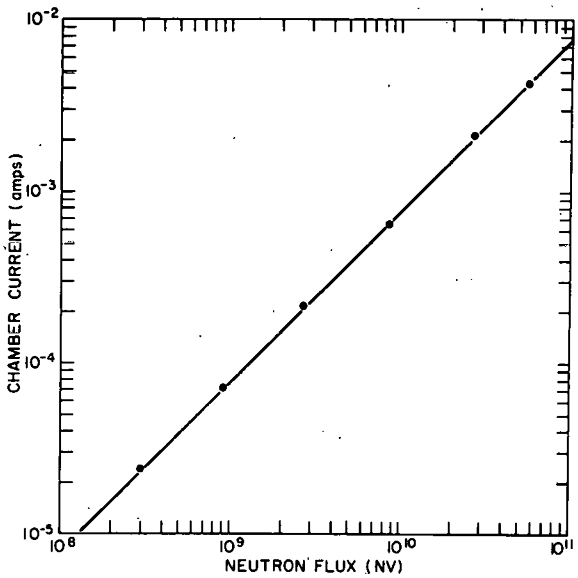

Fig. A-47 - Typical Ion Chamber Neutron Sensitivity Curve

(2) Electrometer Ammeter

The Spert-constructed linear electrometer ammeter converts the current from the ion chamber into voltage. The ammeter has current ranges from 10^{-10} through 10^{-4} amp, inclusive, with the desired range selected by a switch on the control console (see Fig. A-41).

The log electrometer was manufactured by Keithley Instruments Company. This device produces an output voltage which is proportional to the log of the input current. The range of the instrument is from 10^{-11} to 10^{-5} amp.

(3) Recorder

The recorders used for the operational linear and operational log are both Minneapolis-Honeywell Brown recorders. The range of these recorders is 0 - 10 mv. The electronics in the recorder are calibrated with a Rubican potentiometer and the accuracy of the calibration is better than 0.25%. The standardizing circuit in the recorder utilizes an Evenvolt power supply to provide a constant bucking voltage.

3. Bulk-Water-Temperature Indicator

The bulk-water temperature in the Spert-I reactor is measured by a thermopile consisting of four copper-constantan thermocouples connected in series. The thermopile is located on the reactor tank wall, approximately at the core centerline. The thermopile leads extend to a reference junction maintained at $49.2^{\circ}C$. The signal from the thermopile is filtered to eliminate 60-cycle a-c noise and transmitted wihout amplification to the control center. The temperature is read in millivolts on a multi-range Minneapolis-Honeywell Brown recorder equipped with five ranges and an automatic range selection. The range span is 0 to 10.2 mv.

4. Water-Level Indicator

The water level in the Spert-I reactor tank is measured by means of a float contained in a 15-ft-long aluminum pipe, which is attached by brackets to the inside tank wall. The float is connected by a wire to a counter weight contained in a similar pipe. The wire runs over a pulley geared to a Telesyn self-synchronous motion-transmitting system, which operates a Veeder-Root digital counter at the control console and has a least-count of 0.01 ft. The float actuates a microswitch set to stop the deionized water fill pump when the level reaches a predetermined height.

5. Radiation Detectors

The gamma radiation levels directly over the reactor tank and at other points in the reactor area are detected by five Area Radiation Monitors manufactured by Tracerlab, Inc. The signals from these chambers are indicated in the Spert I control room on five dial meters calibrated in mr/hr. In addition, the gamma radiation level is recorded in the control center Health Physics office on a multipoint Bristol recorder. A warning bell at the reactor building alarms when the gamma radiation level reaches a predetermined set point, normally

10 mr/hr, and a red light on the Area Radiation Monitor in the excess radiation field is activated. The bell is audible in the reactor control room over the intercommunication system.

A Nuclear Measurement Corporation Constant Air Monitor, located in the Health Physics frame structure adjacent to the reactor building, continuously samples the air from the reactor building for particulate or gas radiation. The radiation level of the filter accumulated sample is recorded on the CAM strip chart recorder.

V. Auxiliary Equipment

1. Water Treatment System

a. Introduction

Water for the Spert site is supplied by two deep-well pumps, one at the control center designated as No. 1 well pump house and one 500 yd southeast of the control center designated as No. 2 well pump house. The No. 1 well has a 20-stage, 10-in. submersible pump with a capacity of 400 gpm at 500 ft of head; the No. 2 well has a 15-stage, 10-in. line shaft pump with a capacity of 550 gpm at 545 ft of head.

Either or both pumps operate intermittently on storage tank level controls to supply two interconnected storage tanks having a total capacity of 75,000 gal. Low-level alarms on the tanks, connected with the ADT system at the fire department, are set to alarm when the water level in the tank drops to 5-1/2 ft.

Water is distributed to the various reactor areas by two parallel booster pumps, which, in conjunction with a pressure control valve, maintain 70 psig pressure on the Spert distribution line. Under normal operating conditions only one booster pump is in service; whenever supply pressure drops below 40 psig, however, the second booster pump can be put in operation manually.

Well water is supplied to Spert I at 70 psig. The well water is used for emergency fill in the reactor, utility purposes, and feed for the water-treatment equipment which provides demineralized water to the reactor. Table A-I presents a typical analysis of Spert well water.

The water-treating equipment for the Spert-I facility is located in the terminal building. The system, shown in Fig. A-48, consists of a water softener, one demineralizer unit, and associated piping and controls.

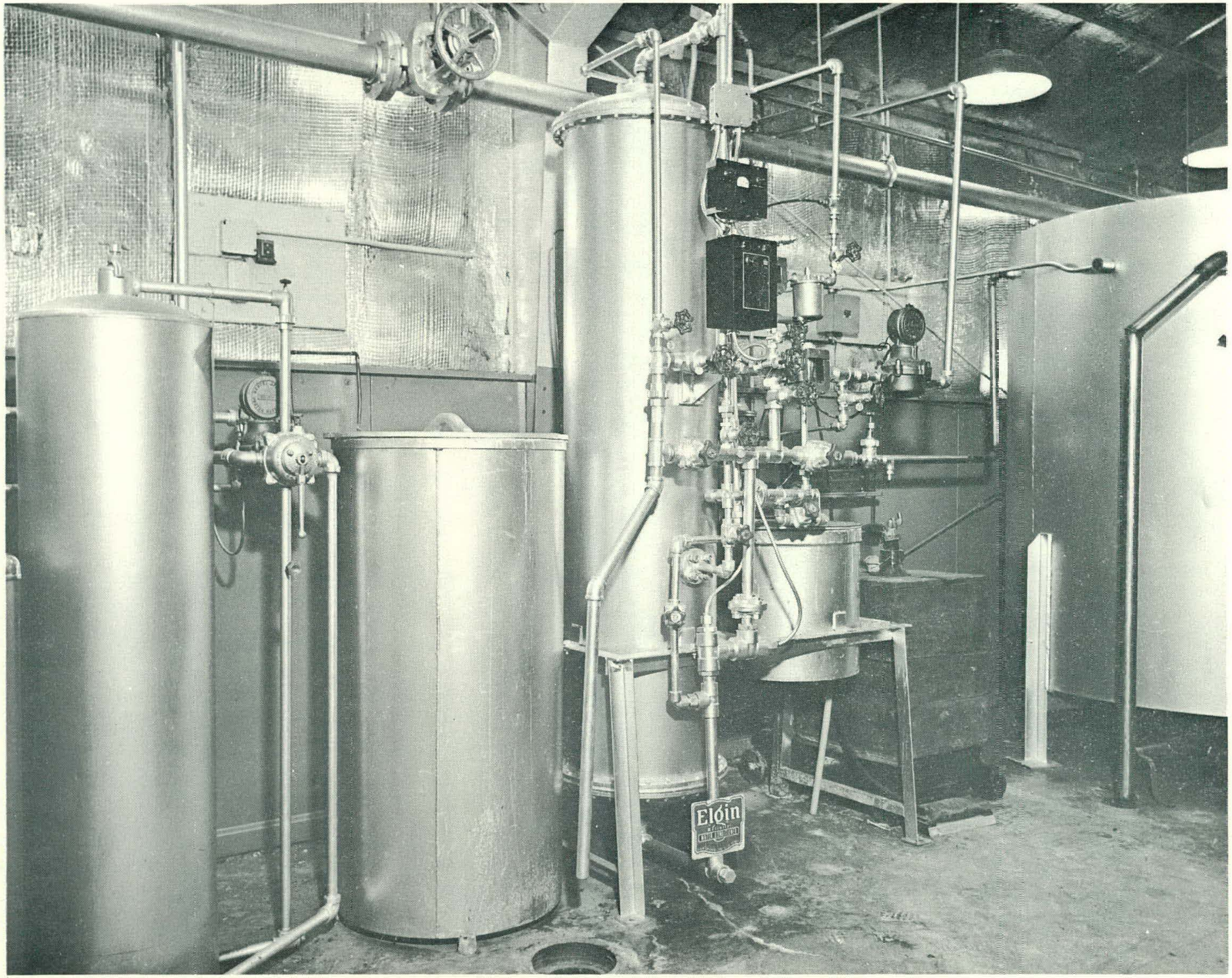

Fig. A-48 - Water-Treating System

TABLE A-1

Analysis of Spert Well Water

	ppm
Ca	39
Mg	14
Fe	0.04
F	0.
Mn	0.01
Na	8.8
K	27
B	0.05
SiO_2	26
NO_3	1.2
HCO_3	158 ppm as $CaCO_3$
Cl	16
SO_4	7
Dissolved Solids	205
Hardness	147
pH	8.2
Specific conductance at 25°C	332 micromho/cm

b. Softener

An Elgin Zeolite water softener, using a carbonacious zeiolite and NaCl as regenerant, is installed in the system to remove water hardness, thereby permitting regeneration of the deionizer with sulfuric acid. The softener has a capacity of 16,000 gal/cycle and a maximum flow rate of 10 gpm.

c. Deionizer

The deionizer is an Elgin Ultra Deionizer, single-column, mixed-bed unit, having a capacity of 6700 gal of deionized water with resistivity of 50,000 ohm-cm at a maximum rate of 18 gpm. Break-through of the column is determined by a conductivity cell located in the effluent line. A pneumatic valve, operated by the cell, automatically stops the flow of water to the storage tank when the conductivity reaches a preset maximum.

Deionized water is stored in the terminal building in a 1000 gal storage tank. The tank is equipped with a high-level cutoff to the water treating equipment and a low-level cutoff to the reactor fill pump.

d. Transfer Pump

Deionized water is delivered from the storage tank to the reactor building by means of a Peerless single-stage, centrifugal pump with a capacity of 120 gpm at 120 ft of head. The pump is driven by a 5 hp Reliance motor. The deionized water reactor fill line terminates 3 in. above the reactor tank concrete floor. Approximately 8 hr are required to fill the reactor.

2. Compressed Air System

Compressed air is used in the Spert I reactor for the control rod and transient rod piston accelerators and for general plant use.

The air compressor is located in the terminal building and is a Schramm Model KB. Loss of air pressure while the reactor is operating will cause an immediate scram.

3. Reactor Equipment

Auxiliary equipment located in the reactor building provides support for the experimental program and plant operations.

a. Sump Pump

The reactor building sump pump is located below grade in the reactor building and is used to drain the reactor tank. Reactor water flows by gravity to the sump pit. The sump pump is an Aurora Model 1-1/2 MSM manufactured by the Atlas Equipment Company and may be operated on automatic control or on manual control from either the reactor building or the Spert I control room. Approximately 60 minutes are required to drain 12 ft of water (the normal operating level) from the reactor vessel. If the pump fails or if the reactor drain valve is damaged, portable sump pumps can be used directly in the reactor vessel.

b. Mechanical Stirrer

A bracket is welded to the reactor tank wall to support a portable Lighting Mixer. The 6-in. propeller is positioned at the core centerline approximately 2 ft from the tank wall. The stirrer is used to obtain a uniform temperature distribution of the bulk-water moderator.

c. Television Camera

A remotely controlled, closed-circuit television camera is mounted between the legs of the lower support bridge, approximately 8 ft above the core. Another complete closed-circuit television system is available to observe the interior of the reactor building before and after a transient test.

d. Periscopes

Three periscopes serve as light transmission lines for high-speed cameras operating during destructive test transients. The periscopes are placed in positions permitting observation of the reactor core from several angles. They are constructed of aluminum and are able to withstand two atmospheres of external water pressure. The lenses in the objective end are constructed of non-browning glass.

Four brackets are welded to the reactor tank wall to position two periscopes to observe the north and east sides of the core. Additional brackets are installed to position the other periscope for observation of the top of the core.

The upper horizontal ends of the periscopes extend from 2 to 5 ft over the building floor, thereby permitting the cameras to be positioned away from the edge of the reactor tank and shielded from the high-radiation field.

e. Cameras

High-speed cameras will be used for the destructive test transients. Plans for the installation of eight cameras have been completed. The cameras are: three 16 mm Waddell cameras Model 16-3, with speeds variable from 350 to 8400 fps; four 16 mm Bell and Howell, with maximum speeds of 128 fps; and one 16 mm Fastax camera, with a speed of 3000 fps.

A special aluminum waterproof camera box with a plexiglass window has been constructed for one of the Bell and Howell cameras. The box is equipped with radiation shielding to protect the film and will be submerged in the reactor tank during the transient tests.

Experience has shown that the best photographs can be obtained using underwater lights of the Par 64 or Sun-gun type. A transformer is used to boost the voltage to 230 volts for these lights. Strobe lights have also proved effective for closeup shots through the periscopes. A detailed discussion of the photographic apparatus and its intended use is included in Section IV-C.

4. Emergency Shutdown System

It is conceivable that an electrical or mechanical malfunction of the control rods or physical damage (such as fuel plate buckling) to the core could prevent insertion of the control rods to shut down the reactor. Therefore, it is necessary to provide some means that is independent of the control rods for removing excess reactivity so that the reactor can be shut down for repair. The method selected to accomplish this is the addition of a water soluble poison such as gadolinium nitrate.

A pressurized, soluble-poison injection system has been designed to shut the reactor down should the control rods be jammed in an up position after a severe transient. The system consists of a compressed-air supply, compressed-air tank, poison-solution tank, and solenoid-operated valves. The poison solution tank contains the amount of gadolinium nitrate solution that is necessary to account for feed line holdup and shut down the reactor to at least $2 subcritical. The poison solution is pressured into the reactor tank from the solution tank by using compressed air and a suitable arrangement of solenoid valves. Fig. A-49 is a schematic diagram of the emergency shutdown system.

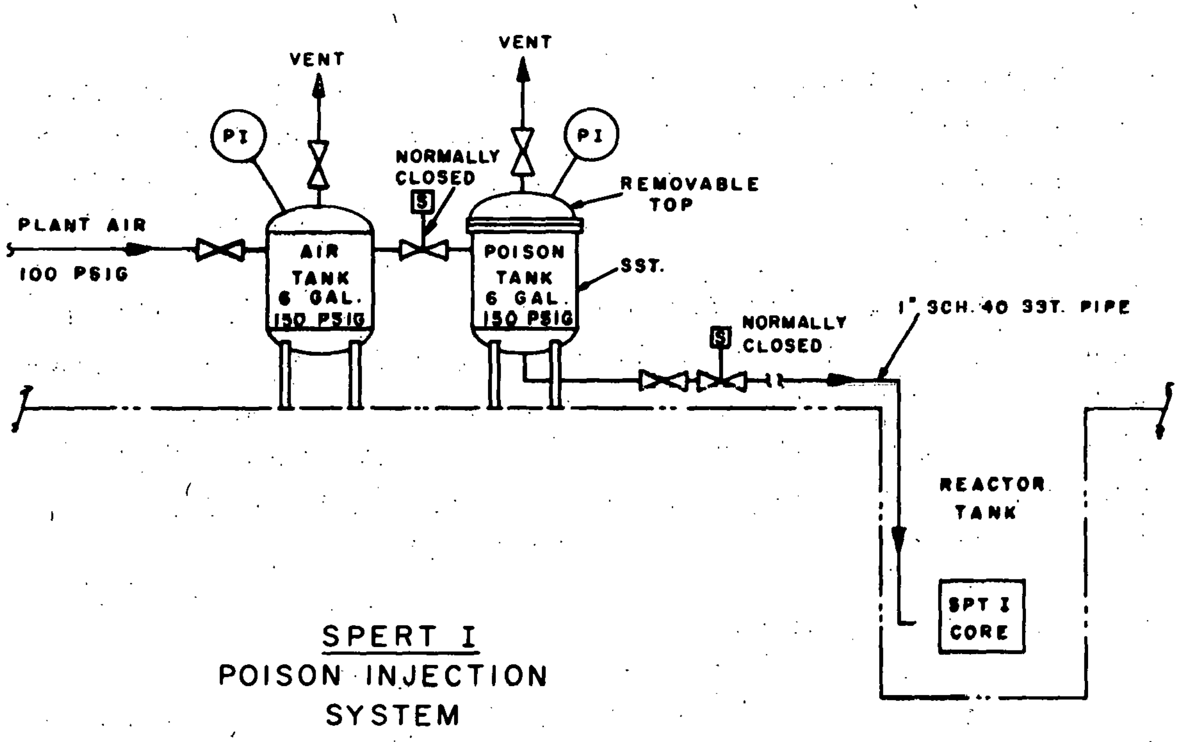

Fig. A-49 - Schematic Diagram of Soluble Poison Injection System

APPENDIX B

NUCLEAR OPERATION TESTING PROCEDURE*

I. Organizational Responsibility

Basic responsibility for nuclear and non-nuclear operation of a Spert reactor resides in two groups: (a) the Engineering Section, which has responsibility for all non-nuclear operation, including plant modification, maintenance and repair, and (b) the Nuclear Test Section, which has responsibility for all nuclear operation, instrumentation and data collection.

Preparations for nuclear operation of the reactor are carried out by the Engineering and Nuclear Test Sections. The Nuclear Test Section is responsible for ensuring that the required operational instrumentation in functioning properly. This consists of at least two neutron pulse counters producing an audible signal and two neutron current recorders (one log scale and one linear scale) visible to the operator. In addition to the operational instrumentation, all of the test data instrumentation and recorders are checked out at this time. The Engineering Section is responsible for ensuring the proper functioning of the reactor equipment and for the installation of special test equipment. During all non-nuclear operations, an engineer, designated the "Engineer-in-Charge", is responsible for the safety of the facility.

After the Engineer-in-Charge has verified that all plant preparations are complete and that no non-Spert personnel are in the reactor area, he verbally transfers the control and responsibility of the reactor to the Nuclear Test Section by notifying and obtaining concurrence of both the Nuclear Test Sestion Chief (or his designated representative) and the reactor operator. This concurrence and acceptance is noted in the reactor console log book by the reactor operator.

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*The operational procedures summarized in this section have been abstracted from the Spert Standard Practices Manual, which describes in detail the administrative and operational practices followed at Spert, including reactor operations, handling of fissionable and radioactive material, radiation, safety, emergency action plans, etc.

II. Evacuation of Reactor Facility

The reactor area must be evacuated throughout all critical operations. The reactor operator initiates routine evacuation of the reactor area by instructing the assigned Health Physicist (HP) to proceed with routine evacuation. The following steps must be performed in the following sequence to effect a routine evacuation.

1. The HP will notify the Security Guard, who has control of the reactor access road, that the reactor area is being evacuated and that no non-Spert personnel may be admitted until further notice.

2. The HP will then announce the order to evacuate over the reactor area intercom system and the reactor operator will sound the warning horn at the reactor area three times from the reactor console.

3. The Engineer-in-Charge or his delegated representative will obtain the plant operations log book and remove it from the reactor area.

4. The HP will inspect the reactor area, including both the inside and the outside of all buildings in the area, to ensure that all personnel have left the area.

5. The HP will then set up the road block and signal the reactor operator to turn on the exclusion light. When the light comes on the HP will acknowledge.

6. The HP will then report to the Security Guard that the reactor area is closed to all personnel, and check that all personnel have left the area as indicated by the checkout cards at the Security Guard Post.

7. The Engineer-in-Charge will report to the control room.

8. The HP will report to the control room that the area is clear.

The Nuclear Test Section Chief, or his designated Group Leader, is in charge of the facility during reactor operation, and is the responsible supervisor in the control room. In the event of a plant emergency, command reverts to the Engineer-in-Charge after the reactor has been safely shut down.

III. Operational Procedure for a Transient Test

With the required instrumentation on and operating, the reactor operator announces over the intercommunications system his intent to start nuclear operation, and will listen for possible response from the reactor area before proceeding with any reactivity addition. Upon permission from the responsible supervisor, the reactor operator will turn the console power on and perform the control rod scram check by individually withdrawing each control rod approximately 5 in. and scramming it by the manual scram button.

A sequence timer is used for programming certain events of the test. Prior to the test, the occurrence of events is programmed on the timer and a dry run of the timer is made to ensure proper operation. (Activating the timer turns the data-recording instrumentation on, turns the cameras and other equipment on, fires the transient rod, scrams the control rods, and turns the transient recording instrumentation and other equipment off in the proper sequence and at the proper time. Application of the sequence timer is an experimental convenience; it in no way takes control of the reactor from the reactor operator, who can scram the reactor at any time.)

With these and the other preliminary checks made by the operator, the initiation of a reactor transient proceeds as follows:

1. Criticality is obtained.
2. The transient rod poison section is inserted into the core to the extent that it will more than compensate the reactivity addition required for the transient test.
3. The predetermined reactivity addition required for the test is made by withdrawing the control rods a corresponding amount above the critical position.
4. Following a readiness check with control room personnel and approval by the supervisor, the operator arms the transient rod.
5. The sequence timer is turned on and the sequence of events previously described occurs.

IV. Reactor Shutdown

Upon completion of the transient test, the reactor is safely shutdown and the console keys are returned to a locked depository. The responsible supervisor in the control room will designate a person to remain on duty at the console until re-entry of the reactor building has been made.

The minimum conditions that must exist before the reactor can be said to be safely shutdown are:

1. All control rods and transient rod seated.
2. All control rod and transient rod drives at lower limit.
3. Instrumentation must indicate a shutdown reactor.
4. In the best judgement of the responsible supervisor in the control room, the reactor is subcritical and no foreseeable events will lead to criticality.
5. The console power must be turned off.

After a test or series of tests occurring in any one day, when it has been established that the reactor is safely shutdown, the responsible supervisor notifies the Engineer-in-Charge that the reactor has been shutdown and will at this time transmit any other information pertinent to the plant or reactor conditions to the Engineer-in-Charge. The responsible supervisor will then request the Engineer-in-Charge to assume control and responsibility for the reactor. With concurrence of the Test Section Supervisor, responsibility and control of the reactor reverts back to the Engineering Section.

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