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August 1963

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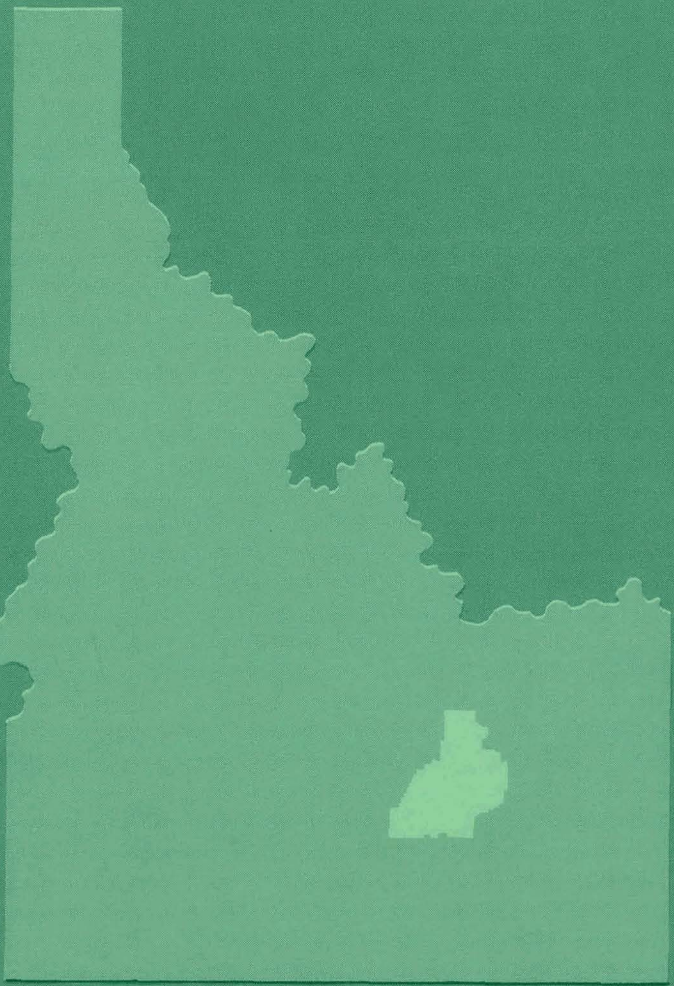
J. E. Grund and B. E. Norton



PHILLIPS PETROLEUM COMPANY
PHILLIPS 66
ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION

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Available from the
Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

Printed in USA

IDO-16906
AEC Research and Development Report
Reactor Technology
TID-4500 (22nd Ed.)
Issued: August 30, 1963

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by

J. E. Grund and B. E. Norton

PHILLIPS
PETROLEUM
COMPANY

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

ACKNOWLEDGMENTS

We wish to acknowledge the aid of the many Spert technical personnel who aided in the preparation of this report. In particular, J. A. McClure provided valuable assistance in the preparation of Section V on the hazards evaluation and P. L. Mahoney, H. W. Faire, Jr., D. R. Barton, K. O. Johnson, and J. A. Norberg who prepared Appendix A, the detailed description of the facility. We also wish to acknowledge the earlier work done with the oxide core that was reported by A. H. Spano, J. E. Barry, L. A. Stephan, and J. C. Young. This earlier experimental work has been invaluable in preparation for the destructive test program.

SPERT I LOW-ENRICHMENT OXIDE CORE DESTRUCTIVE TEST PROGRAM SAFETY ANALYSIS REPORT

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, an oxide core destructive test will be performed in the fall of 1963 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 low-enriched oxide core; the experimental results of nondestructive 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. Discussion also is given of the supervision and control of personnel during and after each destructive test, and of the plans for reentry, cleanup, and restoration of the facility.

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 comprised of approximately 600 4%-enriched UO_2 fuel rods clad with stainless steel, and 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.

On the basis of the results obtained from previous nondestructive kinetic tests, an analysis has been made to determine the nature of the results to be expected for an assumed 1.8-msec-period test in which total core destruction occurs. An evaluation of hazards involved in conducting the 1.8-msec test, based on conservative 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 incurred.

SPERT I LOW-ENRICHMENT OXIDE CORE
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DESTRUCTIVE TEST PROGRAM SAFETY ANALYSIS REPORT

I. INTRODUCTION

1. GENERAL

Preparations are underway for a low-enrichment oxide core destructive test at the Spert I facility in the fall of 1963. These tests form an integral and important part of the Nuclear Safety Program being carried out at the National Reactor Testing Station by Phillips Petroleum Company for the U. S. Atomic Energy Commission under sponsorship of the Nuclear Safety Research and Development Branch of the Division of Reactor Development.

This report is intended to provide a summary of the objectives of the Spert I oxide core test program; a detailed description of the Spert I reactor facility, instrumentation and operating procedures pertinent to the destructive tests; the results and analysis of nondestructive 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 associated with the test program.

2. SPERT REACTOR SAFETY PROGRAM

The inherent capability of reactors in general to experience rapid increases in power has been recognized from the first as offering 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 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 recognition of the need to obtain an understanding of the physical phenomena involved in reactor excursions and the implications in regard to reactor accidents and consequences, Phillips Petroleum Co. was asked by the U. S. Atomic Energy Commission in 1954 to undertake a long-range reactor safety program. Instructions at the inception of the Spert program included not only basic nondestructive studies of the importance of various parameters in reactor kinetic behavior but also included planned integral-core destructive tests to investigate the consequences of reactor accidents [1, 2, 3].

3. PREVIOUS DESTRUCTIVE TESTS

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 excursions were performed in a reactor comprised of MTR-type fuel elements [4, 5, 6]. While only minor damage was obtained as a consequence of these initial 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 19×10^9 watts, an estimated nuclear burst energy of about 135 Mw-sec, 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, it constituted the only planned destructive test performed until recently.

Beginning in April of 1962, a series of kinetic tests were performed with a highly enriched, plate-type, aluminum-clad core to be used for destructive testing in Spert I [7]. These preliminary tests provided data which could be extrapolated to predict performance of this core in response to very large reactivity perturbations which could be expected to produce destructive effects. Extrapolations of the data obtained in these tests indicated that for an excursion with an initial period of about 2 msec, the central region of the core should reach the vaporization temperature of aluminum. It was anticipated that violently destructive effects and very high pressures would be produced when the vaporization threshold was reached, but it was also recognized that an unpredictable threshold in the formation of large transient pressures might manifest itself for tests of almost any period in the range from 4.6 msec (the shortest period test with nondestructive results) down to 2 msec.

On November 5, 1962, a successful destructive test was performed having a 3.1-msec period, producing the predicted 2400-Mw peak power, and 33 Mw-sec nuclear energy release. Approximately 15 msec after the power peak had occurred, and after the power level had declined to less than 10% of the peak value, a very large (the order of 5000 psi) short-rise-time pressure pulse occurred which caused violent disassembly of the reactor core and damaged the core support structure and drive systems. The core damage produced was of the same order of magnitude as that which had previously been observed in the Borax I destructive test and the SL-1 accident [7].

The Borax and Spert tests demonstrated the importance of destructive core tests and confirmed the feasibility of conducting such tests with negligible risk to operating personnel.

4. OBJECTIVES OF THE SPERT DESTRUCTIVE TEST PROGRAM

A long-range destructive test program would be intended to be a comprehensive investigation of destructive reactor accidents for various classes of reactors. It would entail a series of controlled destructive tests designed to provide information on questions relating to (1) reactor kinetics and shutdown behavior; (2) the magnitude of the pressures generated and their mechanical effect of the reactor environs as related to the general problem of containment; (3) energy partition; (4) the extent of mechanical damage, radiation exposure, fission product release, etc, resulting from a given destructive burst; and (5) identification of the ultimate shutdown mechanism in a severe accident. For the first step in this overall program, the test series to be performed in Spert I will obtain data on the response of the oxide core reactor to large reactivity insertions, as distinct from test data bearing primarily on accident consequences exterior to the reactor vessel. Because of the large number of low-enrichment oxide systems in use and planned for use at the present time, scoping information on destructive excursions for such a system is of immediate interest.

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

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

(2) What is the pressure pulse developed within and without the reactor vessel, the impulse loading given to the reactor vessel and core-support structure, and the effect of the pressure pulse in the violent disassembly of the core?

(3) Is the energy source in the destructive accident entirely nuclear, or is there a significant contribution from other sources?

(4) 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?

The combination of the long thermal time constant in this fuel and the Doppler mode of reactivity shutdown results in almost insignificant transient pressures being developed for excursions with periods as short as 3.2 msec. It is expected, however, that as shorter period tests are performed, the cladding will ultimately rupture or melt and suddenly release UO_2 into the reactor water. It is possible then that a sudden increase in the rate of energy deposition in the water will result from the distributed source and may produce considerably larger destructive effects than were observed for the Spert I plate-type system. The pressure pulse which may result from rupture of the first few fuel rods may also, by its effect on other rods, initiate self-propagating failures throughout the core. The primary objective of this test program would be to determine the nature of the destructive effects produced when the high-energy-content UO_2 is dispersed into the reactor water as a result of cladding failure. It may be found that the reactivity insertions necessary to produce cladding rupture are so large that it is extremely difficult to produce the required excursion. Utilizing the oxide core and core structure which is presently on hand with only minor modification, it is estimated that the most severe test which could be performed will have a period of approximately 1.8 msec (corresponding to a reactivity addition of about 3.0$) and produce a nuclear energy release of greater than 200 Mw-sec. If such a test does not produce pin ruptures in this core, then the primary objectives of investigating the destructive effects arising from the sudden release of fuel to the water could still be satisfied by repeating the test with purposely defective fuel rods in the central region of the core.

5. SCHEDULE

The destructive test program to be performed in Spert I in 1963 will consist of (1) static measurements to establish the basic core properties; (2) fiducial, nondestructive kinetic tests, to establish base-point data for kinetic behavior for extrapolation to shorter-period, destructive regions and for comparison with other Spert Cores; (3) final preparatory kinetic tests, for checkout of instrumentation, etc; and (4) a destructive test, conducted with meteorological controls, special operating procedures, and personnel control appropriate to a violently destructive test.

Portions of the static and fiducial-kinetic tests have been completed previously [8] and will not necessarily be repeated. Assuming that approval for the conduct of the Destructive Test Series is received during the summer of 1963, the series will be initiated in September 1963. The test program is expected to be completed in the fall of 1963 barring unexpected difficulties or the onset of inclement weather.

II. BRIEF DESCRIPTION OF REACTOR FACILITY

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

1. 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 Figure 1); an earth-shielded instrumentation bunker, which is located adjacent to the reactor building and contains the transient electronic instrumentation used for transmitting data 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.

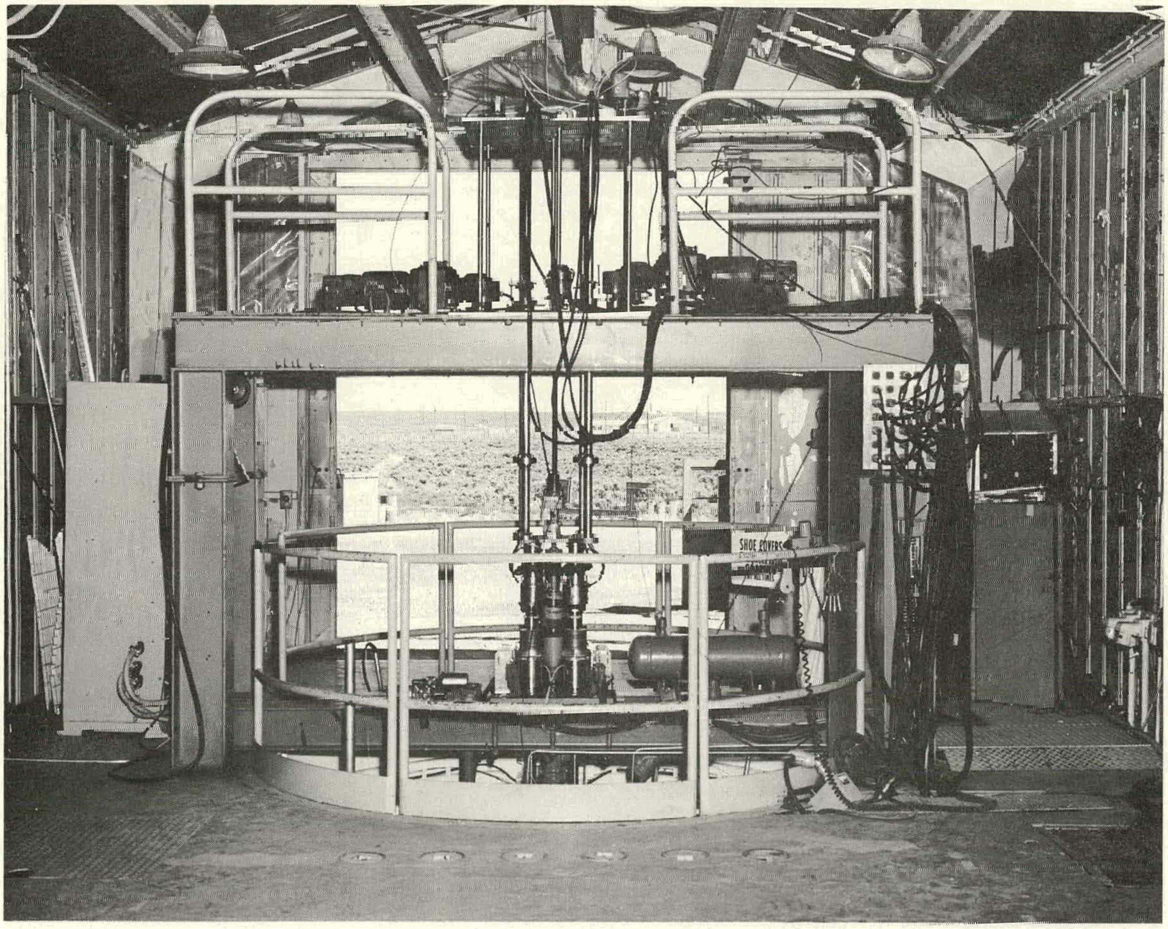

Fig. 1 View of interior of reactor building.

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

2. FUEL ROD

The fuel rods comprising the Spert I experimental core are those previously used in the Babcock and Wilcox N. S. Savannah critical assembly [9] and Spert I transient testing [8]. The fuel rod is a 6-ft-long, welded-seam, 304 stainless-steel tube, containing low-enrichment UO_2 powder, swaged-compressed to an effective density of approximately 87% of the theoretical density of UO_2. The fuel element design parameters are given in Table I.

TABLE I

N. S. SAVANNAH - TYPE FUEL ROD PARAMETERS [9]

Total length of fuel rod	71.5 in.
Active length	66.9 in.
Outside diameter of fuel rod	0.500 in.
Cladding thickness	0.028 in.
Fuel	Compressed UO_2 powder
Fuel enrichment	4.02 wt% U-235
Effective UO_2 density	9.45 g/cm^3
Mass UO_2 per fuel rod	1600 g
Mass U-235 per fuel rod	56.7 g
Mass U-238 per fuel rod	1353 g

3. CORE

The experimental core is assembled in the Spert I 10-ft-diameter open tank, which contains no provision for pressurization or for forced coolant flow. The core is comprised of fuel rods arranged with a center-to-center rectangular fuel rod spacing of 0.663 in. A top view of the core is shown in Figure 2 and a quarter section through the core is shown in Figure 3.

The octagonal core section is divided into four quadrants by a 3/4-in. thick, aluminum rod-guide cross, which houses the four control rod blades and the centrally located, cruciform transient rod. The fuel rods in the initial series of kinetic tests with an unconstrained core were supported by the upper and lower aluminum grid plates only, fitting loosely in 0.516-in. diameter grid-plate holes. Under equilibrium temperature conditions, the unclamped fuel rods could expand thermally in the axial direction without bowing, but, under transient conditions, radial bowing was experienced. For a constrained core, an additional grid, designed to prevent this radial movement of the fuel

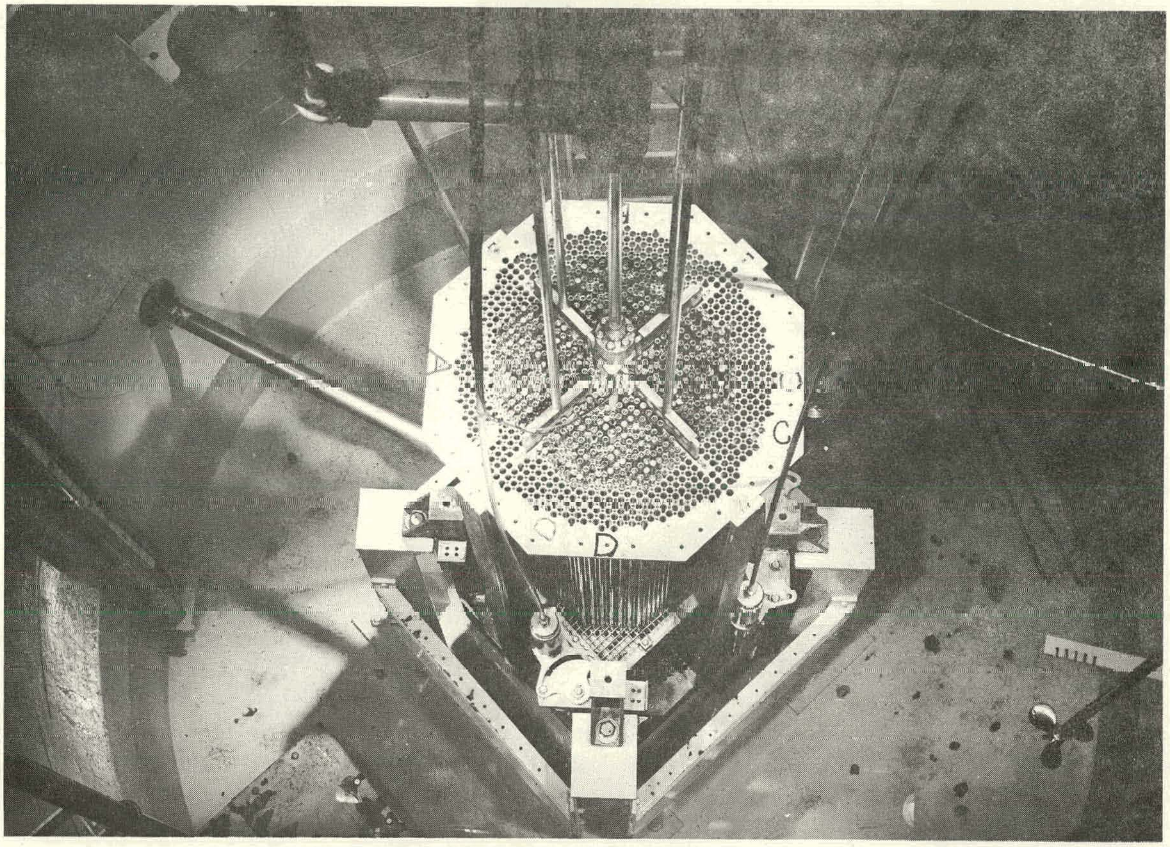

Fig. 2 Top view of Spert I oxide core.

rods, is installed about 17.4 in. above the bottom of the core, in the neighborhood of the peak of the axial flux distribution. The grid is comprised of two layers of combs of aluminum shims inserted between the fuel rods, with one layer of combs oriented at right angles to the adjacent layer in egg-crate fashion. An aluminum ring, placed at the perimeter of the core, prevents outward motion of the fuel rods.

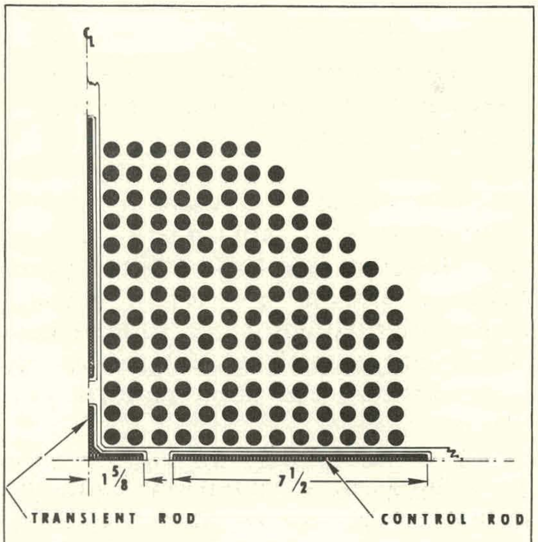

Fig. 3 Quarter section of oxide core containing 592 rods.

4. CONTROL SYSTEM

The poison section of the four gang-operated control blades (each 56 in. long x 7-1/4 in. wide x 3/8 in. thick) and of the transient rod (26-5/8 in. long x 2/3 in. wide x 3/8 in. thick) are constructed of an aluminum-boron alloy containing 7 wt% natural boron. The control blades and transient rod have follower sections which are constructed of aluminum.

The Spert I control rod drive system restricts the vertical movement of the control rods to a travel of 23.7 in. In an effort to maximize rod worth, the control rods are positioned axially so that the bottom tips of the poison move from a "fully inserted" position, 15.5 in. above the bottom of the active core region, to a "fully withdrawn" position, 36.5 in. above the bottom of the core [*].

The transient rod poison section, which normally is positioned below the core, can be raised a maximum of 26 in. into the core. Rapid ejection of this rod from the core is used to initiate step-induced power excursion tests of the oxide core.

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 multisection timer unit with associated relays is used to initiate selected experimental functions in a given sequence during a reactor transient, ie, the ejection of the transient 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.

5. OPERATIONAL INSTRUMENTATION

The Spert I operational instrumentation includes the neutron detection, reactor bulk-water temperature, reactor water level, and radiation-detection instrumentation.

Operational neutron instrumentation includes three B^{10}-lined pulse chambers with amplifiers and counters, a B^{10}-lined gamma-compensated, ion chamber connected through a linear electrometer to a 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 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

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[*] Core rod travel positions are measured from the "fully inserted" control rod position at the core elevation of 15.5 in. Thus, the critical rod position of 8 in. signifies that the rods are 23.5 in. above the bottom of the core.

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 monitored 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.

6. 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 operational items. Further discussion of the auxiliary equipment is given in Appendix A-V.

III. EXPERIMENTAL PROGRAM

1. REACTOR TEST SERIES

The oxide destructive test program may, for convenience, be divided into three series of reactor tests, designated the static, fiducial-transient, and destructive-transient series, ie, transient tests which are potentially destructive. Since this basic core has been extensively tested previously [8], a sizeable amount of the information to be obtained from the first two series of tests is already available. Consequently, these first two series of tests will not be as extensive as for a new, untested core and they will serve primarily as reproducibility checks on previous tests results.

The static tests are those required to determine the basic characteristics of the test core. Because of the testing previously performed with this core, only selected experiments will be repeated to determine that those characteristics are unchanged from those determined earlier. This series will include (1) the approach to critical loading and establishment of an operational core loading; (2) control and transient rod worth measurements and from these, determination of the excess and shutdown reactivities of the core; (3) flux distribution measurements; (4) isothermal temperature coefficient measurement; and (5) power calibration of the nuclear instrumentation.

The fiducial-transient tests will consist of step-initiated self-limiting power excursions, covering the reactor period range down to about 4 msec. The shortest period test previously performed with this core was 3.2 msec. These tests are intended to establish base-point data on the kinetic behavior of the core, for comparison with the behavior observed previously and for providing a basis for extrapolation to shorter period tests. These tests also will serve the purpose of proof-testing a large portion of the instrumentation and procedures to be used for the destructive test.

The destructive test series will be performed at periods in the order of 2 msec. All the 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 continuation of the destructive tests, the series will continue until a test occurs which results in sufficient damage to warrant termination of the tests.

2. 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 and to ensure the safety of Spert personnel and the NRTS. 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 1/3 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.

The Spert I reactor control system is not provided with automatic safety circuits due to the nature of the experimental program. 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 unanticipated situations or unsafe operating conditions.

3. TRANSIENT INSTRUMENTATION AND MEASUREMENTS

The transient instrumentation includes devices used to determine transient reactor power, energy release, water pressure, fuel rod temperatures, strain, acceleration and air pressure in or near the reactor core and vessel during a power excursion. The requirements of this instrumentation system are determined by the environmental, dynamic response, and sensitivity limitations imposed on the transducers. The transducers must be physically sized to be positioned in or near the core, have acceptable low-radiation sensitivity, be capable of operating in water or near a water environment, and have a frequency response of up to 20 kc or better. The nature of the tests also implies that precautions must be taken to provide a high probability of obtaining data from each test. For these reasons the instruments and leads are protected where possible from blast or missile damage and multiple instrumentation of the same or different types for the same measurement are used for simple redundancy.

In addition to the dynamic measurements mentioned above, photographic documentation of the tests will be obtained through use of cameras located at various vantage points surrounding the reactor vessel and building. Radiological measurements of direct dose, fallout or fission product release and dispersal of radioactive release to the environs also will be performed. After the tests, selected components of the reactor, fuel rods, core components, etc, will be metallurgically examined to determine extent and mode of failure.

A detailed description of the transient instrumentation and measurements is found in Appendix A-6.

(*) 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.

IV. BRIEF DESCRIPTION OF PREVIOUS EXPERIMENTAL RESULTS AND EXTRAPOLATION TO SHORTER PERIOD TESTS

The following is a brief discussion of the experimental results obtained on this oxide core previously studied in the Spert I facility. The results of these previous tests showed that, in such a long thermal-time constant and low-enrichment reactor, reactivity compensation for limiting power excursions is accomplished through a prompt mode of Doppler broadening and a delayed mode of moderator density effects. For a complete summary of the data, reference should be made to IDO-16751 [8], "Self-Limiting Power Excursion Test of a Water-Moderated Low-Enrichment UO_2 Core in Spert I". The step-transient summary data along with the summary and conclusions section of the above report are presented in Appendix C. Unless otherwise noted, all data given here pertain to the SA-592 constrained core, ie, the core which had an intermediate grid installed to restrict radial motion of fuel rods during a transient.

1. STATIC DATA

A compilation of the previously measured core parameters is listed in Table II. A complete discussion of the static data is given in IDO-16751.

TABLE II

CORE STATIC PARAMETERS

Initial critical mass (unconstrained core)	28.4 kg
Initial critical loading (unconstrained core)	500 pins
Final operational loading	592 pins
Excess reactivity	2.8$
Reduced prompt neutron lifetime (ℓ/β_{eff})	3.6 msec
Average uniform void coefficient (moderator)	-0.024 ¢/cm^3
Average uniform void coefficient (reflector)	-6 x 10^{-4} ¢/cm^3
Isothermal Temperature coefficient (20°)	-0.3 ¢/°C
Area-weighted peak to average thermal flux ratio	4.9

2. TRANSIENT MEASUREMENTS

The fuel rod support structure in this core originally consisted of an upper and lower grid plate arrangement which permitted, under equilibrium temperature conditions, free thermal expansion of the unclamped fuel rods. Step-induced transient experiments initially were performed on the unconstrained core. The results of these transient tests revealed that transient fuel rod bowing occurred with reactivity effects which were positive. The addition of a constraining grid virtually eliminated the effect of the fuel rod bowing on the burst behavior down to the shortest period obtained. Since a more complete coverage of this phenomenon is given in IDO-16751 and since the present oxide core will have this constraining grid in place, this discussion will be concerned only with the constrained core data.

In the constrained core test series, the range of initial asymptotic periods investigated was from approximately 10 sec to 3.2 msec. The tests were initiated at ambient temperatures of about 20°C and at power levels of a few watts.

In Figure 4 the maximum power obtained in the constrained core is shown as a function of the asymptotic reciprocal period. Included in this figure for comparison purposes is the data for the Spert I D core, an aluminum plate-type core, which underwent destructive testing in the summer and fall of 1962.

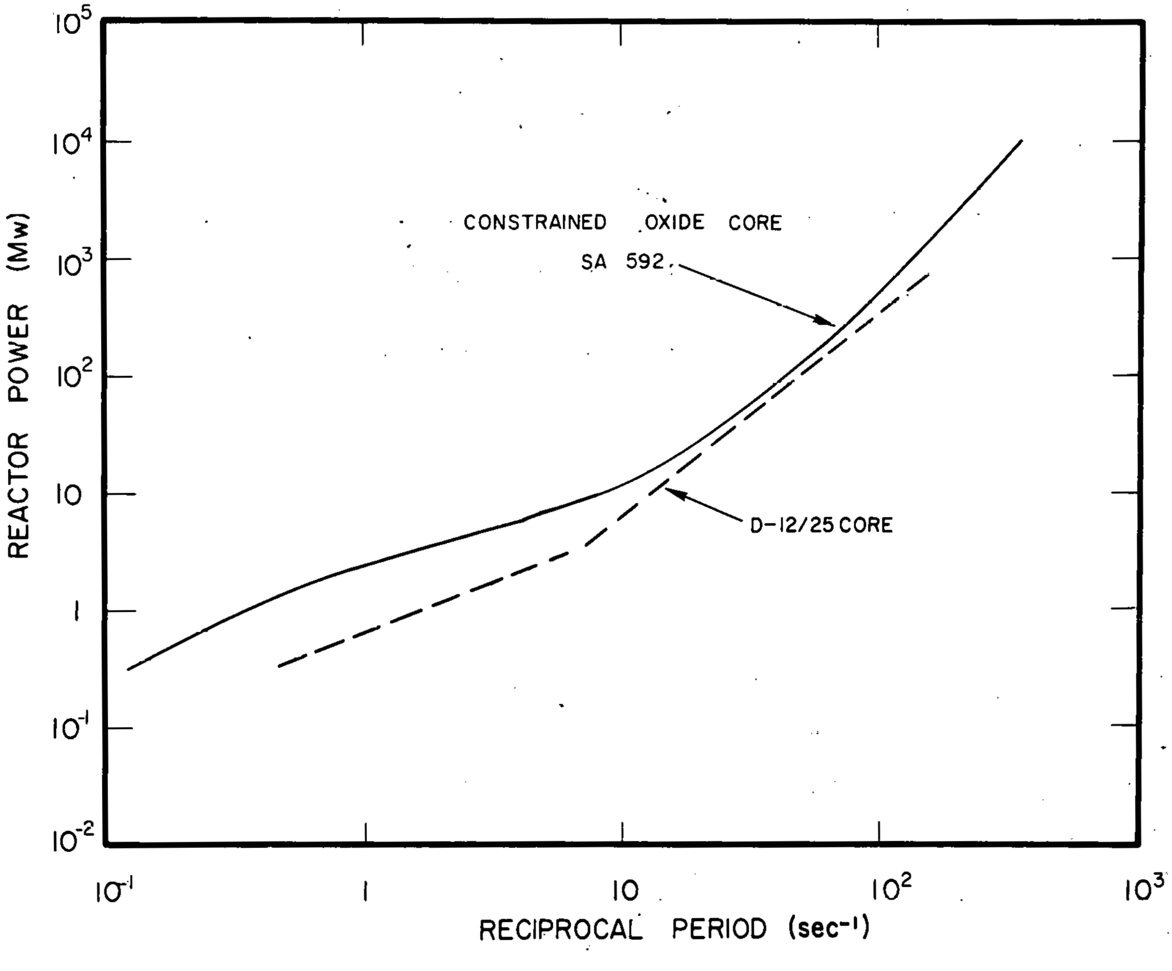

Fig. 4 Maximum power vs reciprocal period for the constrained oxide core and the D-12/25 core.

Energy to the time of peak power as a function of asymptotic reciprocal period for both the oxide core and the D core is shown in Figure 5.

Cladding surface temperature rise at peak power plotted against reciprocal period (Figure 6) presents an interesting comparison of the oxide core with

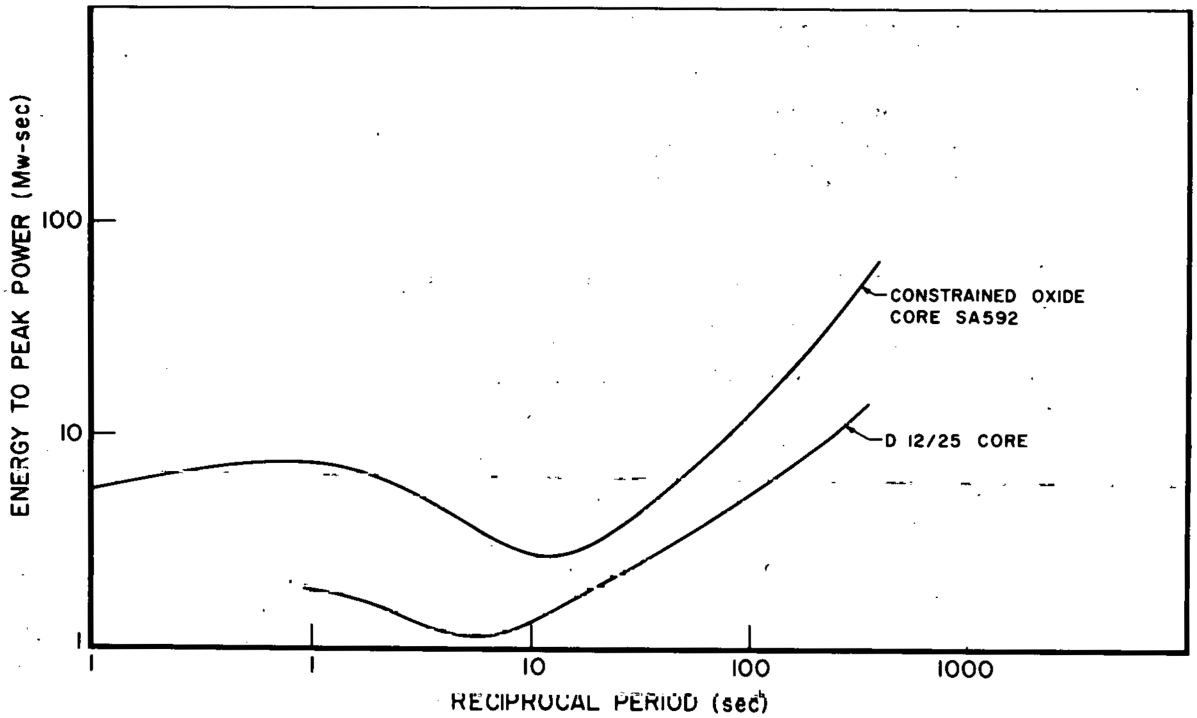

Fig. 5 Energy release to time of maximum power vs reciprocal period for the constrained oxide core and the D-12/25 core.

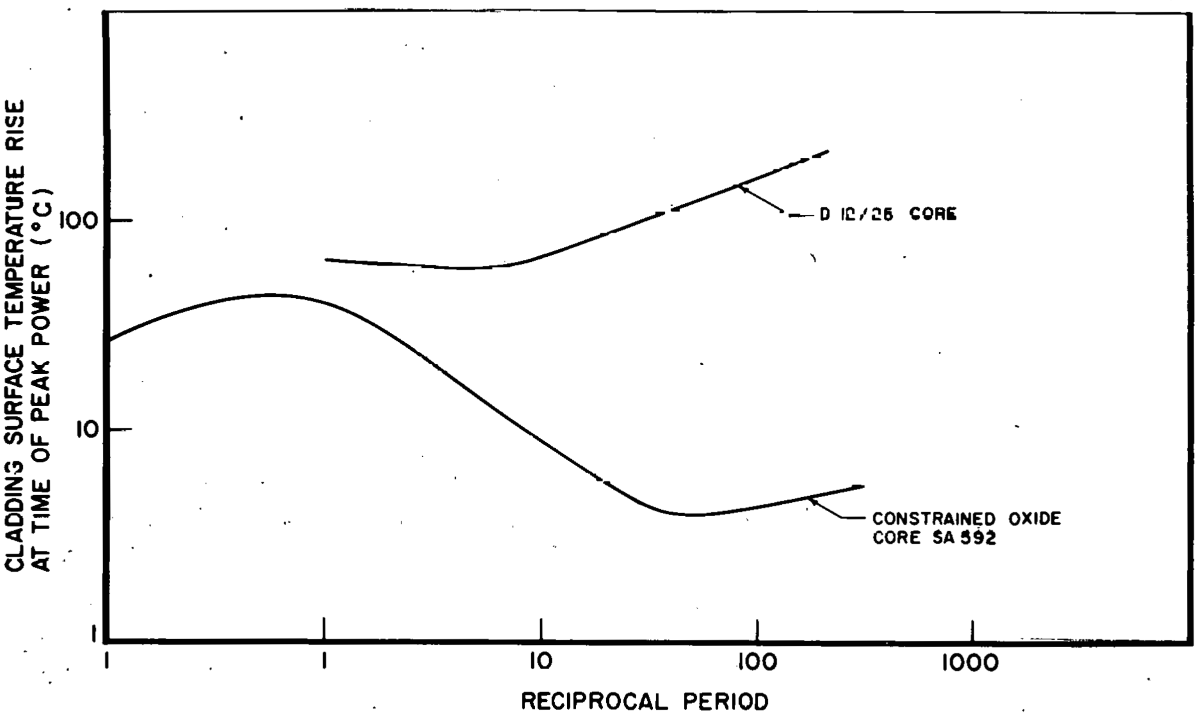

Fig. 6 Fuel surface temperature rise at time of maximum power vs reciprocal period for the constrained oxide core and the D-12/25 core.

the D core. With increasing α_o, initial shutdown of the oxide core by moderator heating decreases. This behavior is in contrast to the situation in the highly enriched plate-type D core where, because of the shorter thermal-time constant of the fuel elements, higher surface temperatures at time of peak power are obtained and the boiling mechanism becomes increasingly important as α_o increases.

3. 3.2 MSEC TEST

The shortest period transient which has been run on this core to date had an initial asymptotic period of 3.2 msec. The maximum power obtained from this self-limiting burst was about 7500 Mw, and the total energy release was about 107 Mw-sec. Motion pictures indicated the formation after the power burst of large steam voids accompanied by violent motion of the fuel rods in spite of the fact that the constraining grid was in place for this test.

Calculations indicated that the maximum UO_2 temperature in the flux peak region approached approximately 1800°C for the 3.2-msec test. Further calculations showed that the yield stress of the stainless steel cladding was approached during this transient [10]. For these reasons, it was deemed necessary to obtain post-test information on the condition of the fuel rods.

Representative fuel rods were chosen from the flux peak region for metallurgical examination. First, dimensional measurements were made on the fuel rods to determine whether swelling, bowing, or warping had taken place. None of these effects could be detected. Next, a rod was sectioned and examined for melting and/or sintering of the UO_2 and changes in microstructure of the stainless steel cladding which could be attributed to the transient. All of these results were negative. Finally, radiographs of representative fuel rods revealed some cracking of the UO_2 in the high flux regions of the core. This cracking of the compacted UO_2 was the only detectable damage from the 3.2-msec transient.

4. EXTRAPOLATION OF TRANSIENT DATA TO AN APPROXIMATE 1.8-MSEC-PERIOD TEST

The hazards analysis presented in Section 5 of this report must rely for its basic assumptions on predictions based on extrapolation of the presently available data. During the course of preliminary tests for the oxide 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 also has been possible in the case of transient pressure data.

A degree of caution is required, however, even in short extrapolations if the next test to be performed results in crossing the "threshold" for a new effect such as may arise when temperatures achieve the melting or vaporization points for the fuel, strain yield points of certain reactor components are exceeded, etc, if it is possible that the new effect can materially change the dynamics of the system as a whole. In the case of the 3.2-msec-period test, described above, wherein the fuel temperatures did not approach melting and metallurgical investigation did not reveal any damage to fuel rod or cladding, no damage was obtained in the facility. Shorter period and more energetic power excursions may, however, cause fuel rod rupture by thermal stresses, melting, or vaporization. In the event of sudden rupture of a fuel rod containing powdered UO_2 during a short period transient, there is the possibility of a rapid transfer into the water of nearly all of the burst energy stored in the rod. Since the energy in the fuel would be dispersed throughout the water, the transient pressures generated by sudden steam formation might be expected to be much higher than those obtained in the equivalent plate-type reactor accident, so that much greater mechanical damage would then ensue. In fact, if a single fuel rod were to rupture, the resulting pressures might rupture the surrounding rods, which may then cause still others to rupture, etc, giving rise to a sort of chain reaction.

Continued testing of the oxide core to shorter periods will be planned and executed with the same degree of caution and preparation used in testing of the plate-type cores. 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 4- and 1.8-msec periods. As seen from the plots of the constrained oxide core data, Figures 7 and 8, both peak power and energy to time of peak power, $E(t_m)$, appear to be well-behaved functions of the reciprocal period and straight line extrapolations have been made on the logarithmic plots. At a period of 1.8 msec (α = 556 sec^{-1}) these extrapolations yield an estimated peak power of about 28 Gw and an energy to time of peak power of about 110 Mw-sec. Because the power burst is approximately symmetrical for power excursions with this core, the total energy release for a 1.8-msec-period excursion is expected to be a little more than twice the energy to peak power, or about 250 Mw-sec.

The oxide core with its low heat transfer rate has shown no appreciable surface temperature rise on the fuel rod at time of maximum power during the shorter period excursion tests and the maximum temperature of the rod surface has not exceeded boiling by any appreciable amount except on the 3.2-msec-period test which indicated some film blanketing with higher temperatures. These data indicate that cladding failure by melting might not be expected during the shorter period tests unless total steam blanketing occurred which might totally insulate a fuel rod. Even in this circumstance of adiabatic heating of fuel rods, calculations indicate that incipient melting of the clad would not occur at the core hot-spot for a test with a period longer than 2.8 msec. Recognizing the fact that the heating of the fuel rod is not entirely an adiabatic process leads to the conclusion that melting of the cladding is a possible but not necessarily probable cause of fuel rod failure. The heating of the fuel at the centerline or interior of the rod is, however, more nearly an adiabatic process and calculations of meat temperature for one gram of fuel at the core hot-spot for adiabatic heating as a function of reciprocal period are shown in Figure 9. These data indicate that incipient melting in the hot-spot might occur at an initial period of approximately 2.3 msec. After melting has been initiated in the fuel, the maximum fuel temperature is controlled by a feedback loop that includes such factors as the latent heat of fusion of UO_2, and the effect of melting on the Doppler reactivity compensation and hence on the power history. For the purposes of this report, it has been assumed that melting has no significant feedback effect on the energy release or power for a 1.8 msec or longer period test because (a) the melting is expected to occur after the peak of the power excursion, and (b) less than 5% of the total core volume is at or above the energy density required for melting of UO_2. On these

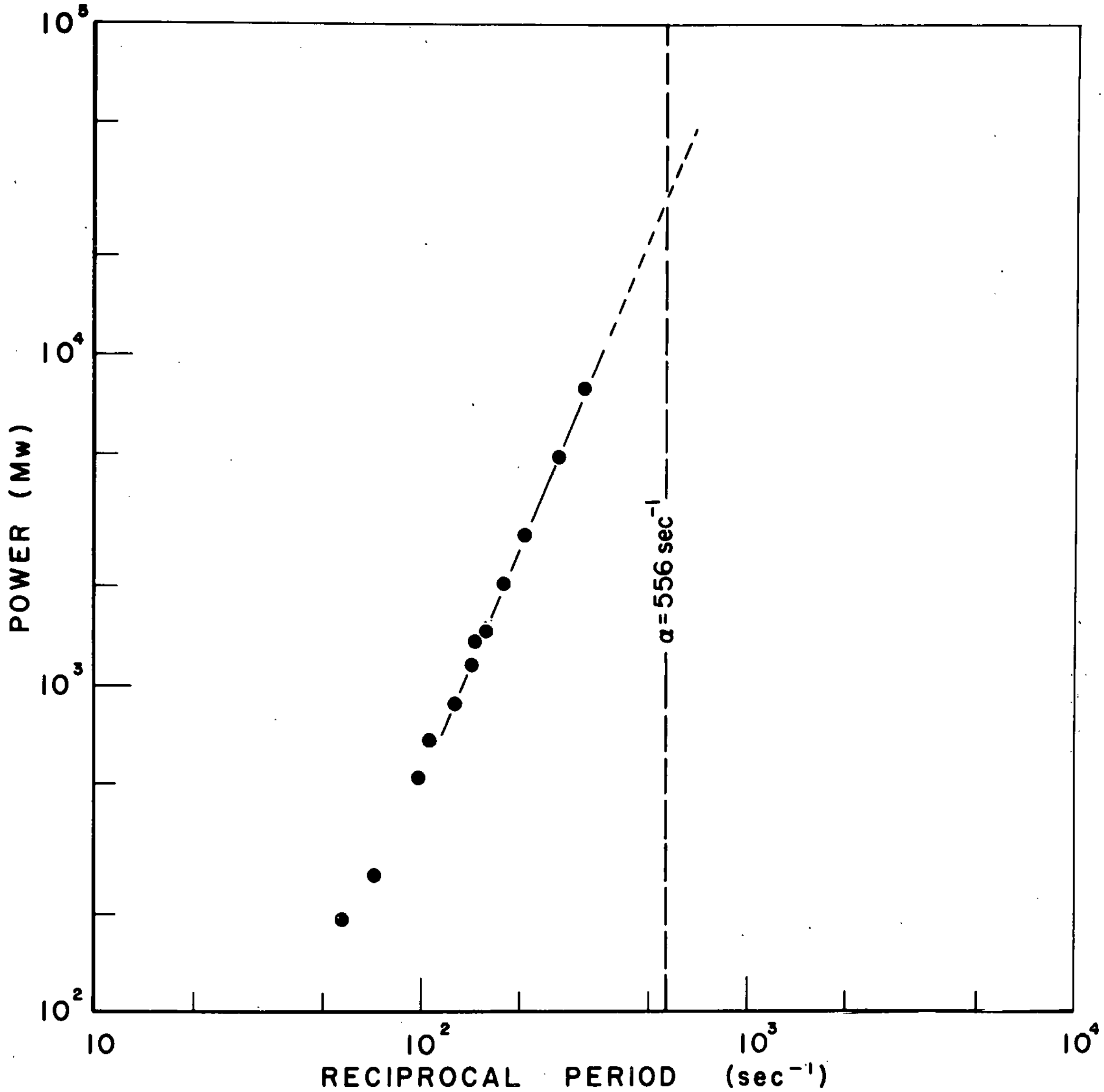

Fig. 7 Maximum power vs reciprocal period for the constrained oxide core.

basis, previous data are assumed to be extrapolable and indicate that melting in the interior of the fuel rod can be expected, following the power peak, for a 1.8-msec-period test; but no significant vaporization of the fuel would be expected.

An additional possible mechanism for failure of the fuel rods in addition to that of melting is stress rupture of the cladding. The causes of stresses in the cladding can be due to either temperature gradients which give rise to thermal stresses or internal pressures due to expansion of the fuel or entrapped gases. Calculations have been performed to obtain the temperature distribution in a fuel rod during a power excursion and consequently obtain a stress distribution due to the temperature gradient. The calculated temperature distribution in a fuel rod for the 3.2-msec-period test at the time of maximum tensile stress is shown in Figure 10. This occurred approximately 50 msec after time of maximum power and indicated a maximum tensile stress of about 90,000 psi, which was well below the yield strength of about 125,000 psi for the clad material.

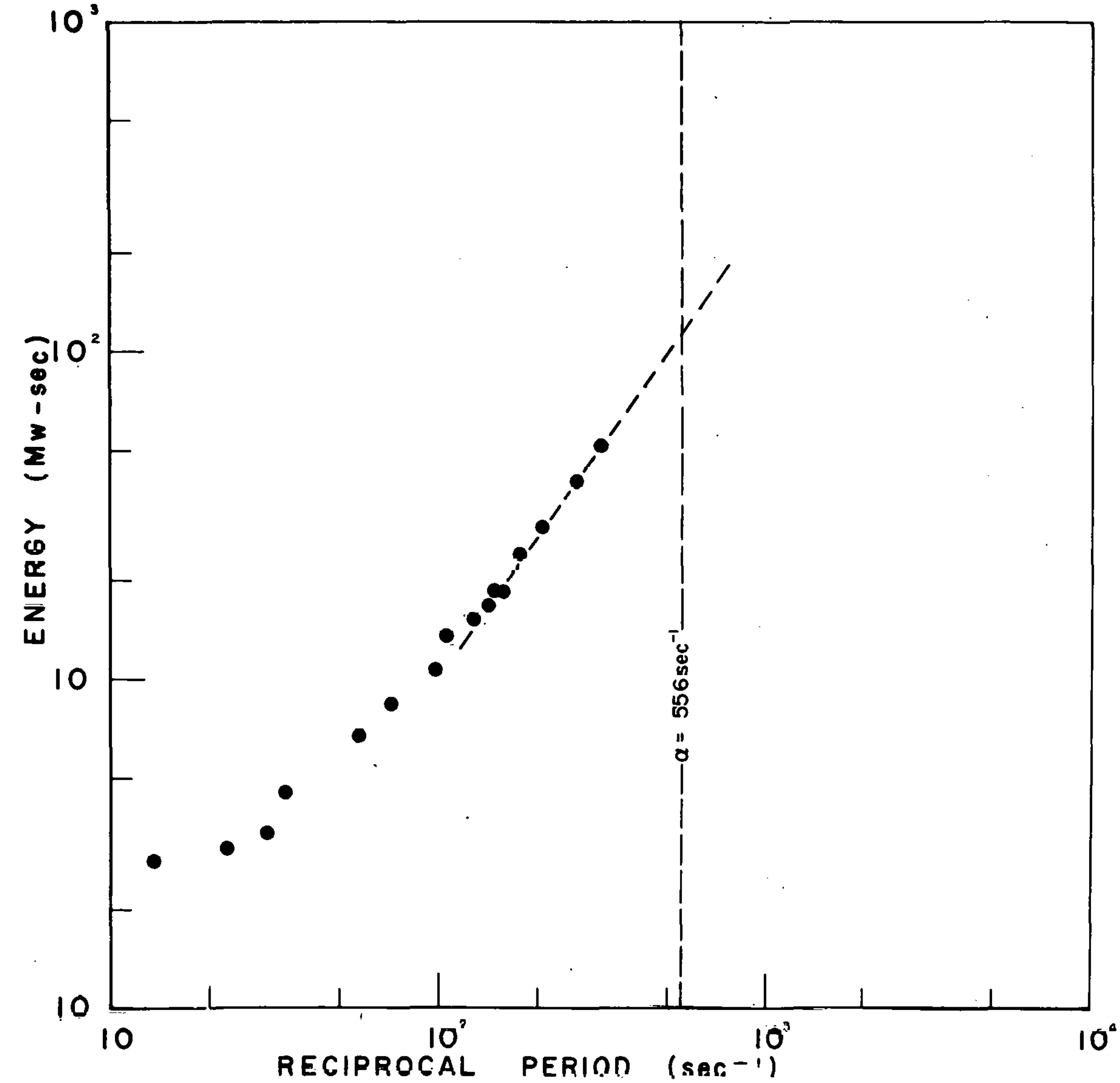

Fig. 8 Energy release to time of maximum power vs reciprocal period for the constrained oxide core.

The calculated maximum compressive stress was about 130,000 psi. As was indicated in the previous section, metallurgical examination of the fuel rods did not indicate any damage was incurred in the 3.2-msec-period test, which tends to corroborate these calculations. Extrapolation of the thermal stress calculations to shorter period tests is difficult because once the surface temperature exceeds boiling and transient boiling heat transfer takes place, the temperature distribution in the clad is not well determined. Consequently, a large uncertainty must be placed on the thermal stress extrapolation to shorter periods that are shown in Figure 11. Also, once the stresses exceed the yield point, plastic deformation will take place to relieve further stresses.

Internal pressures due to expansion of the fuel or entrapped gases also will cause additional stress on the clad which may be added to the thermal stress. Calculation of such additional stresses is very indeterminate because of the nature of the fuel (compacted powder, 87% of theoretical density) which contains appreciable void volume. This allows possible space for expansion without an appreciable pressure buildup that might cause rupture of the rod.

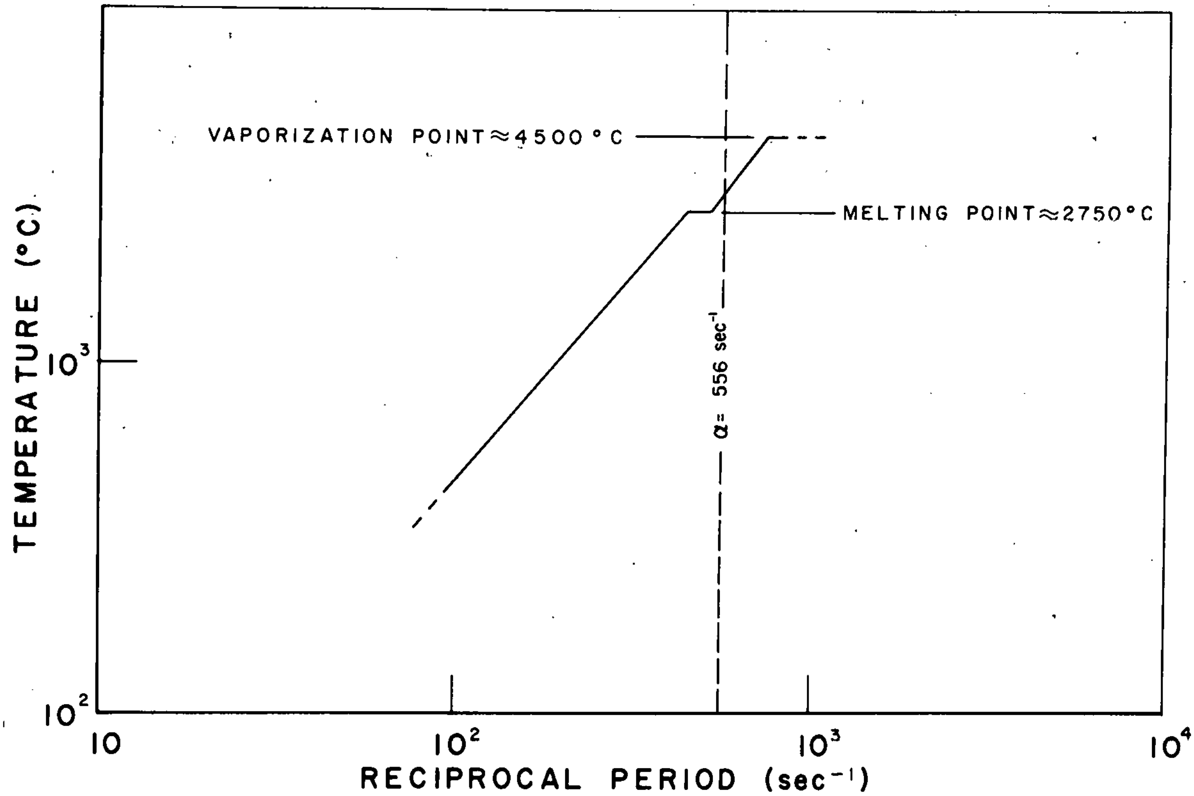

Fig. 9 Maximum fuel temperature for adiabatic heating vs reciprocal period for the constrained oxide core.

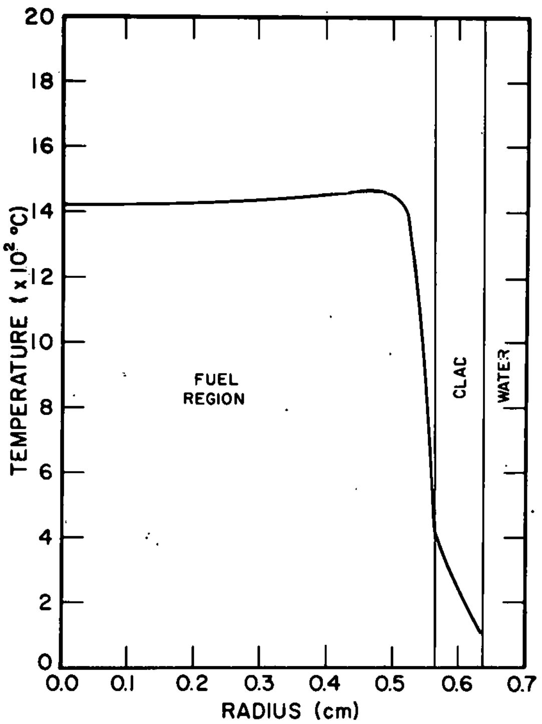

Fig. 10 Calculated temperature distribution in a fuel rod at time of maximum thermal stress for a 3.2 msec power excursion with the constrained oxide core.

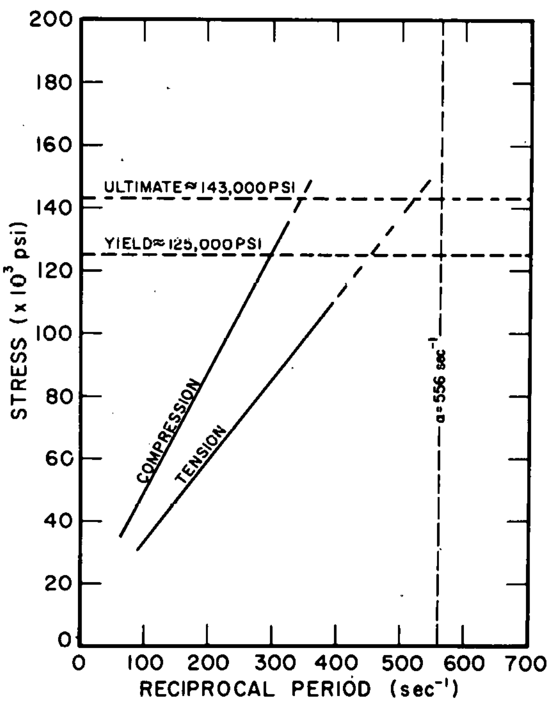

Fig. 11 Calculated maximum compressive and tensile thermal stresses in fuel rod as a function of reciprocal period for power excursions with the constrained oxide core.

Because of the short duration of the large stress values (nearly an impact loading) and because failure also is due to local conditions in the cladding (pits or cracks which can act as stress risers), it is difficult to predict that catastrophic failure of a fuel rod will occur at any given period. It is possible that it could occur at any period less than about 3 msec; however, the probable results of this mechanism would more likely be analogous to thermal fatigue which would be progressive cracking and deformation of the fuel rods from continued or repeated tests at the short periods.

The possible methods of failure of fuel rods which may release large amounts of energy into the water and cause a destructive incident are then (a) melting of the clad and (b) rupture of the rod by internal pressure and/or thermal stresses. For a single 1.8-msec test it is more probable that if destruction is to occur it will be caused by melting of the clad, intimate contact between UO_2 and H_2O with a sizeable steam-pressure pulse that may cause additional failure of other rods. If this mode of failure does not occur for the single test, then failure by thermal fatigue may be expected for repeated tests.

The power excursion tests that have been performed to date have not shown any appreciable pressure (maximum of about 8 psig). The low pressure generation is apparently a consequence of the long thermal time constant fuel which limits the rate of energy deposition into the water. These relatively low levels of pressure are expected to continue for the shorter period tests as long as the fuel rods maintain their integrity. Calculations indicate, however, that if a fuel rod or rods were to rupture with consequent mixing of UO_2 and water, the high energy transfer rates could result in pressures up to tens of thousands of psi. The magnitude of the calculated pressure is directly dependent on the assumptions made as to the nature of the cladding failure.

In evaluating the radiological hazard potential of a 1.8-msec-period test (see Section 5) the extrapolated total energy has been estimated to be 250 Mw-sec. In addition, since the severity of pressures obtained in a 1.8-msec-period test are indeterminate, the destructive pressure 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 destruction occurs due to this pressure and that dispersal of the fuel is extensive.

V. HAZARDS EVALUATION FOR A 1.8-MSEC-PERIOD CORE DESTRUCTIVE TEST

1. RADIATION HAZARDS

1.1 General

In this section an evaluation is presented of the radiation hazards to personnel as a result of a 1.8-msec-period destructive test. The evaluation is based on a conservative 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 power-excursion 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.

1.2 Meteorological Conditions

The destructive test will be conducted only when the prevailing meteorological conditions meet certain specified requirements, ie, lapse conditions and winds 10 to 30 miles per hour from the southwest. The mean wind direction limits are from 200 to 240°.

Doses will be calculated downwind from the reactor site (equivalent to assuming wind direction is toward control center) for both lapse and inversion conditions. During the period when the destructive test will be conducted, the probability of an inversion lasting all day is very small (of order 0.05%). The probability of inversions as a function of duration based on the weather history at the NRTS is given in Table III [11].

TABLE III

PROBABILITY OF INVERSION LASTING FOR GIVEN LENGTH OF TIME

Probability (%)	Length (hr)
0.05	24
10	16 - 17
50	14
75	13
90	5

1.3 Fission Product Inventory in Core

The quantity and quality of fission products which can be released to the atmosphere following the destructive test depends upon the previous operating history as well as the energy release during the transient.

A series of approximately 100 transients has been performed on the UO2 core. Approximately two years will have elapsed between the end of this series and the forthcoming destructive test series. A total energy release of 1200 Mw-sec has been estimated for the first series of transients. It is further assumed for purpose of calculation that this energy release occurred within a narrow interval of time. The activity remaining in the core after two years will be composed of the long-lived fission products and their radioactive daughter products. Any short-lived fission products with a half-life less than 50 days will have decayed to very low values. The activity from the long-lived fission products can be found from:

$$A = (F_o/3.7 \times 10^{10}) \, \Sigma_i \lambda_i y_i e^{-\lambda_i t} \tag{1}$$

where:

A = activity in curies at time t

F_o = number of fissions in 1200 Mw-sec

3.7×10^{10} = disintegrations/sec/curie

λ_i = decay of i^{th} isotope

y_i = yield of i^{th} mass number

t = time in sec

The activity from the short-lived daughter products in equilibrium with the long-lived fission products will be the same as the parent isotope.

That is;

$$\lambda_1 N_1 = \lambda_2 N_2 \tag{2}$$

where λN is the activity in disintegrations per sec. Table IV lists the isotopes which make a significant contribution to the fission product activity after a two-year cooling period.

It is estimated there is approximately 1.3 curies total β and γ activity in the core from first series of tests.

Before the final destructive test is conducted, a series of 12 to 18 transients is expected to have been run. Assume that 500 Mw-sec of energy are released during this series of transients. Assume further that all the energy was produced 30 days before the final destructive test.

The activity from the fission products produced during the initial tests on the destructive series will be composed mostly from short-lived isotopes and can be estimated to within a factor of 2 by the Way-Wigner equation [12]

$$A(t) = A(o) \, t^{-n} \tag{3}$$

TABLE IV

ACTIVITY IN UO_2 CORE AT BEGINNING OF DESTRUCTIVE TEST SERIES

Isotope	T 1/2	Yield	γ_i (sec^{-1})	A_i (Curies)	Type of Radiation
Kr-85	10.57 yr	0.003	2.8×10^{-9}	0.0060	β
Sr-89	53 da	0.048	1.515×10^{-7}	0.0005	β
Sr-90	28 yr	0.059	7.86×10^{-10}	0.0443	β
Y-90	64.8 hr	0.059	2.97×10^{-6}	0.0443	β
Y-91	58.3 da	0.059	1.377×10^{-7}	0.0014	β
Zr-95	65 da	0.064	1.235×10^{-7}	0.0033	β γ
Nb-95	35 da	0.064	2.29×10^{-7}	0.0073	β γ
Ru-106	1 yr	0.0038	2.2×10^{-8}	0.0210	β
Rh-106	30 sec	0.0038	2.31×10^{-2}	0.0210	β γ
Sn-119	230 da	0.001	3.21×10^{-8}	0.0004	γ
Sb-125	2 yr	0.00023	1.1×10^{-8}	0.0013	β γ
Te-125	58 da	0.00004	1.38×10^{-7}	0.0003	γ
Cs-137	27 yr	0.059	8.15×10^{-10}	0.0461	β
Ba-137	2.63 min	0.055	4.39×10^{-3}	0.0438	γ
Ce-144	290 da	0.061	2.77×10^{-8}	0.2920	β γ
Pr-144	17.5 min	0.061	6.6×10^{-4}	0.2920	β
Pm-149	2.6 yr	0.026	8.46×10^{-9}	0.1300	β
Sm-151	73 yr	0.005	3.02×10^{-10}	0.0015	β γ

$\Sigma_i A_i$ = 0.957 Curies

A_γ = 0.37 Curies

A_β = 0.91 Curies

where:

A(t) = activity at any time t sec later (curies)

A(o) = initial activity (curies)

$n = 1.2$ for 10 sec $< t <$ 100 days ($\approx 10^7$ sec)

In order to find A(o), Way has given the following formulas:

$$\beta \text{ activity } A_\beta = 1.40\, t^{-1.2} \text{ mev/sec/fission} \tag{4}$$

$$\gamma \text{ activity } A_\gamma = 1.26\ t^{-1.2} \text{ Mev/sec/fission} \tag{5}$$

The average β and γ energies per disintegration are a function of the irradiation time and also the decay time. Assume an average value of 0.7 Mev/photon for γ disintegrations and 0.40 Mev/disintegration for β decay. Using these two values, the activities of β and γ emitters are

$$A_\gamma = \frac{(1.26\ t^{-1.2})\ (3.1 \times 10^{10})}{(0.7)\ (3.7 \times 10^{10})} = 1.51\ t^{-1.2} \text{ curies/watt-sec}$$

and

$$A_\beta = 2.94\ t^{-1.2} \text{ curies/watt-sec}$$

The total $\beta + \gamma$ activity is then

$$A = 4.45\ t^{-1.2} \text{ curies/watt-sec}$$

The fission product activity from the final destructive test also will follow the Wigner-Way equation. Assuming the destructive test produces 250 Mw-sec of energy, the β, γ, and total activity will be

$$A_\gamma = 3.78 \times 10^8\ t^{-1.2} \text{ curies, } 10 \text{ sec} < t < 10^7 \text{ sec}$$

$$A_\beta = 7.35 \times 10^8\ t^{-1.2} \text{ curies}$$

$$A = 11.1 \times 10^8\ t^{-1.2} \text{ curies}$$

The total activity in the core is shown in Figure 12 as a function of time following the destructive test. Also shown is the contribution from each of the sources.

1.4 Direct Gamma Dose

The dose to personnel at the Spert control center from direct gamma radiation has been calculated assuming the entire core is lifted out of the reactor vessel and placed above ground level. Also assume there is no self-shielding in the core itself. The above assumptions tend to overestimate the dose since the probability that the core leaves the reactor vessel is small and an experimental measurement of the fission product release from melted stainless steel clad UO_2 fuel indicated approximately a 5% gross gamma release [13].

The dose rate in roentgen per hour was calculated from

$$\text{Dose rate (r/hr)} = \frac{A_\gamma E_\gamma e^{-\mu x}}{4\pi x^2 F_\gamma} \tag{6}$$

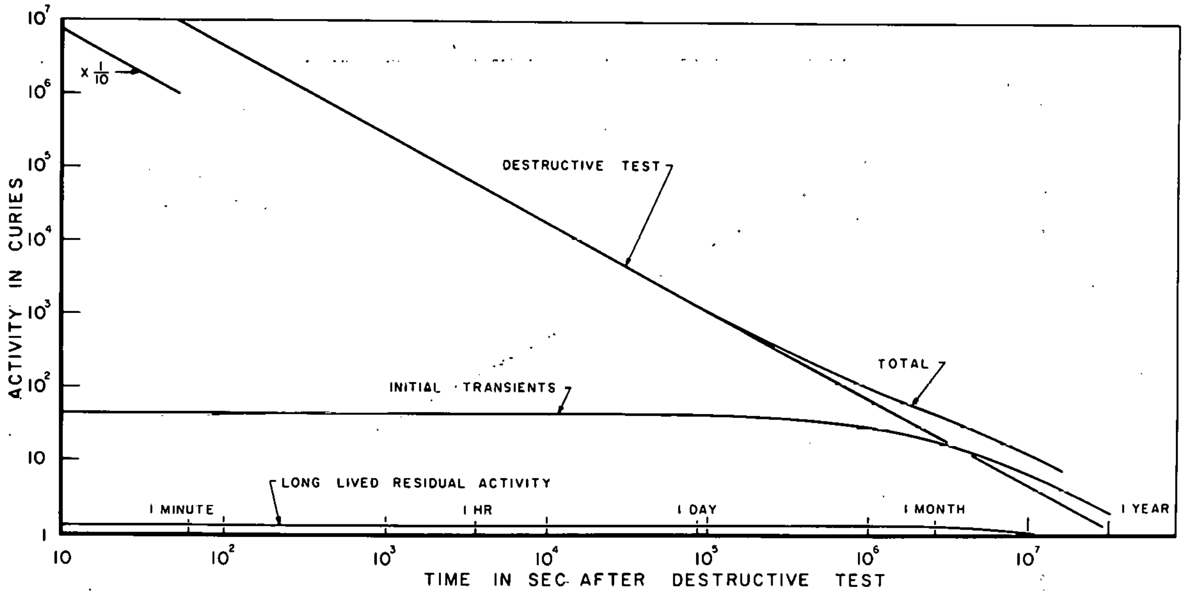

Fig. 12 Total activity in core following the destructive test.

where:

A_γ = γ activity in disintegrations/sec

E_γ = gamma energy in Mev/photon

μ = linear absorption coefficient in air

x = distance from source

F_γ = energy flux to produce 1 r/hr

The values assumed for each of the factors are as follows:

$A_\gamma = 14.0 \times 10^{18}\, t^{-1.2}$

$E_\gamma = 0.7$ Mev

$\mu = 3.5 \times 10^{-5}\ \mathrm{cm}^{-1}$

$x = 8 \times 10^4$ cm (1/2 mile)

$F_\gamma = 5.2 \times 10^5$ Mev $\mathrm{cm}^{-2}\ \mathrm{sec}^{-1}$ (ref 14)

Substituting the above values into Equation (6) gives

$$r/hr = 14.1\, t^{-1.2} \qquad 10\ \mathrm{sec} < t < 10^7\ \mathrm{sec} \qquad (7)$$

A plot of Equation (7) is shown in Figure 13.

The total gamma dose received in a given period of time can be found by numerically integrating the area under the curve in Figure 13. The accumulated

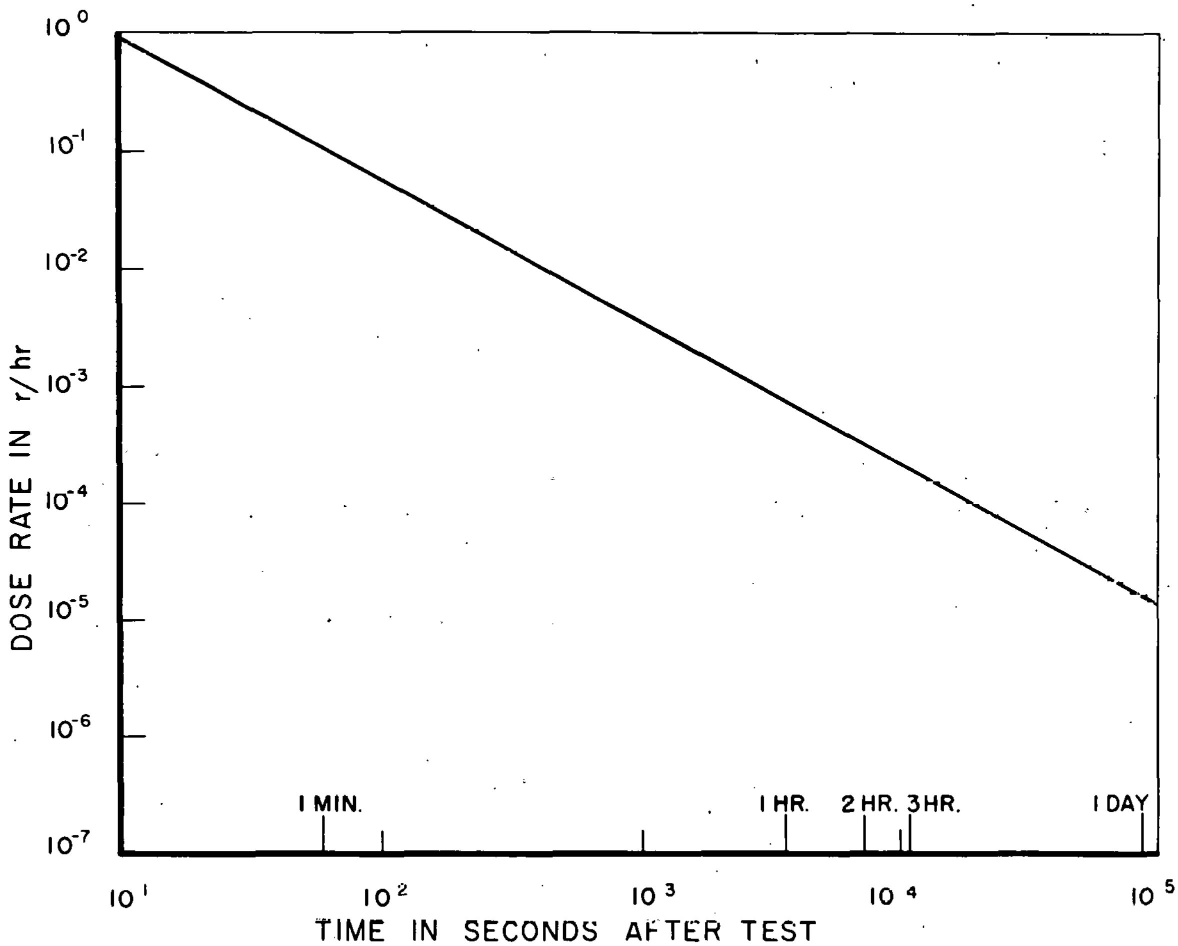

Fig. 13 Dose rate at control center after destructive test.

dose for a three-hour period immediately following the destructive test was found to be 9.34 mr, essentially the infinite time dose. Personnel normally will remain inside the control center during the test, thereby reducing possible exposures.

1.5 Fission Product Cloud Dose

The gamma dose received by an observer at a distance x meters downwind from the release point of a cloud of fission products has been calculated [12]. Pessimistic assumptions have been used which tend to maximize the dose.

The assumptions used in the calculations are as follows:

(1) finite cloud size

(2) cloud does not rise

(3) instantaneous release

(4) center of cloud passes through observers position

(5) an average gamma energy of 0.7 Mev/photon

(6) no fallout

(7) no air scatter

(8) the quantity of fission products released to the atmosphere is that given in Reference 15, ie, 100% of the noble gases, 50% of the halides, and 1% of the solid products. This amount corresponds to approximately 16% of the total fission products produced.

The dose in roentgens from a fission product cloud is given in Figure 14 as a function of distance from the point of release and wind speed. Doses at selected distances are listed in Table V for a wind speed of 5 m/sec.

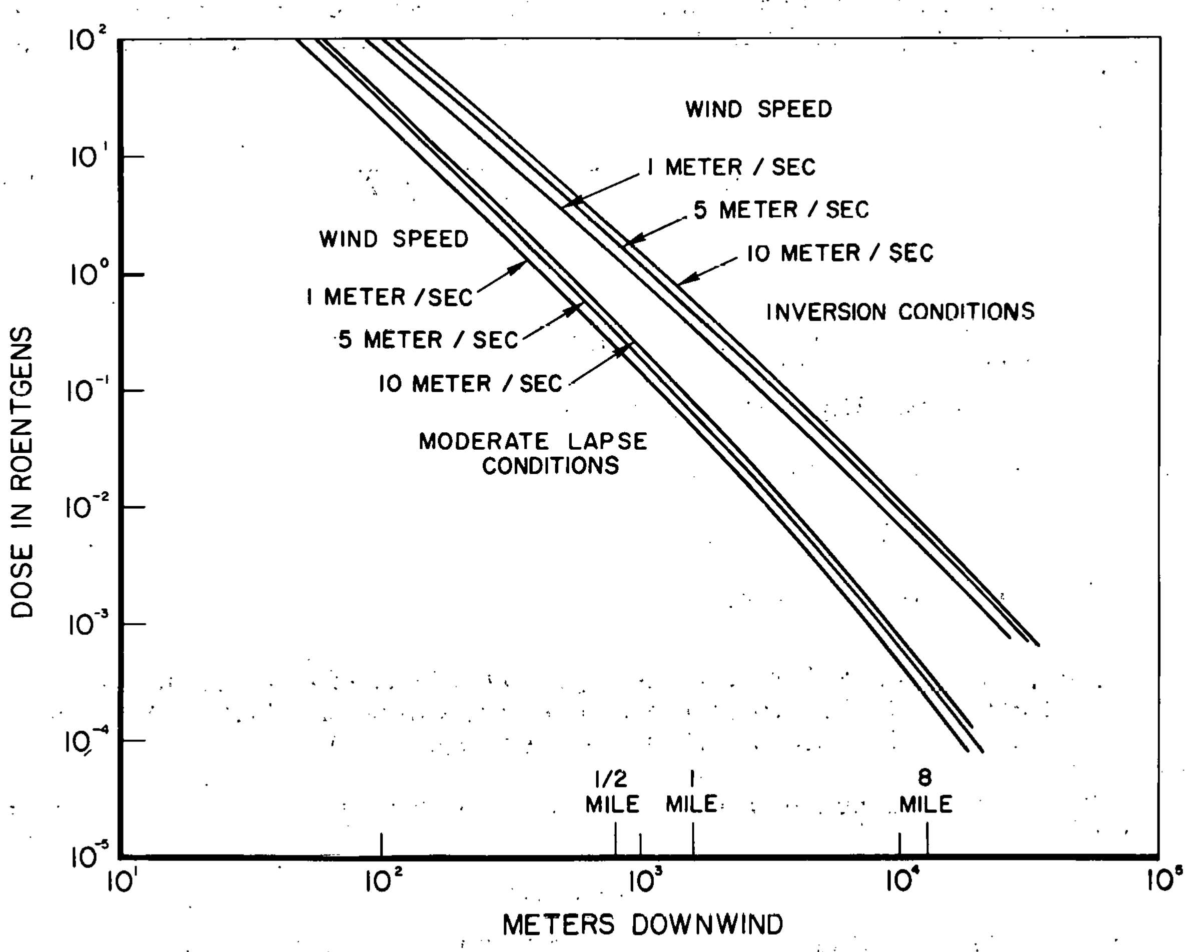

Fig. 14 Dose from fission product cloud after destructive test.

The three hour integrated direct gamma dose for directions other than downwind from part 1.4 was 9.34 mr at 1/2 mile.

TABLE V

DOSE DOWNWIND FROM RELEASE POINT FOR WIND SPEED OF 5 m/sec

Distance (mi)	Dose (mr)	
	Lapse Conditions	Inversion Conditions
1/2	320	1850
1	64	470
8	0.33	5.3

1.6 Fission Product Fallout

The concentration in micromicrocuries per square centimeter of fission products on the ground downwind from the reactor due to fallout was calculated from [16]

$$w_{max} = \frac{n\,Q}{2\,e\,\sqrt{\pi}\;Cy\;x^{2-(n/2)}} \tag{8}$$

where:

w_{max} = maximum concentration in curies/meter^2

n = stability parameter = 0.25

Cy = horizontal coefficient = 0.44

x = distance in meters

Q = source strength in curies

The source strength of each isotope was calculated by assuming 1% of the total accumulated in the core from previous tests escaped and that 100% produced in the destructive test escaped if its precursor was a halide.

The calculated concentrations are given in Table VI for various distances downwind.

TABLE VI

CONCENTRATIONS IN $\mu\mu c/cm^2$ OF FISSION PRODUCTS ON GROUND FROM FALLOUT

Distance (mi)	Sr-89	Sr-90	Y-91	Cs-137	Ba-140	Ce-141	Ce-144
1/2	17	0.034	37	0.12	176	66	7.66
1	4.6	0.01	10	0.034	49.1	18	2.14
8	0.09	0.0002	0.21	0.0007	1.0	0.37	0.043

The concentrations listed in Table VI represent no significant hazards.

1.7 Conclusions

The results of this evaluation of the radiation hazards associated with the performance of core destructive tests indicate no hazard under the planned conditions and ample time would be available for orderly evacuation of all personnel in the vicinity of the test areas under the worst conditions without significant exposure.

2. MISSILE DAMAGE POTENTIAL

An evaluation of the possibility of damage to the Spert II and III facilities or to the control center by missilcs from Spert I destructive tests must draw heavily on the experience of Borax, SL-1, and previous Spert tests. 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. The previous Spert destructive test with the plate-type core also has shown that the missiles are a localized problem as no sizeable debris was found over approximately 75 ft from the reactor.

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

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.

3. 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 may 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 or evacuated from the Spert area under the supervision of a designated evacuation warden. Non-Spert personnel may 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 test 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 may be required to operate cameras in the control center area outside of the building. One photographic liaison man also 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

ID 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 (2)

Health Physicists (2)

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.

VI. REENTRY AND CLEANUP OPERATIONS

1. GENERAL

All reentry 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 has been bulldozed clean of sagebrush and leveled to permit easier identification and recovery of debris and easier access with mobile equipment. If radiation levels require its use, a shielded, self-propelled hydraulic crane ("cherry-picker") will be available for reentry, to permit survey photography of the area and debris, and determination of high radiation sources by means of gamma detectors.

2. REENTRY PROCEDURES

As soon after the test as possible, consistent with knowledge that the reactor is in a safe shutdown condition (Appendix B) and with the monitored progress of any fission product cloud, a special mobile health physics trailer will be set up on the Spert I access road at the closest, convenient point to the reactor area, commensurate with radiation levels. This trailer will contain reentry 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 reentry operations after the destructive test will be carried out by two health physicists, a reactor engineer, and a reactor physicist.

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. The first monitoring and inspection team will enter the area when the radiation field drops to a safe point to permit reentry without excessive exposure to personnel. (Radiation exposure limits are to be followed in accordance with Phillips Petroleum Company AIB-6.15.) 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 to survey the contaminated area, while the first team proceeds to the mobile "hot change" 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 ID Health and Safety Branch; Phillips Petroleum Co. health physics personnel also will obtain dosage measurements, using radiation accident dosimeters, constant air monitors, and remote area monitors located in the immediate area surrounding the reactor building. The area monitors will be connected to remote meters and a strip chart for readout 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.

3. CLEANUP OPERATIONS

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

Prior to actual cleanup and removal operations, reasonably complete written descriptions of the conditions of the reactor, 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 personnel, 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 provided a guide for further operation. This survey will be accomplished by movie cameras mounted on the self-propelled crane and/or by still pictures made upon physical reentry of personnel. A detailed radiation survey will also be made, using either (a) 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; (b) pinhole-camera photography to define localized gamma radiation sources, or (c) where radiation levels permit, health physics instruments upon physical reentry of personnel.

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 may 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/or 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 use 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 Idaho 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 previous Spert destructive test that the facility can be placed in operation in at least six months after such a test.

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 personnel withdrawn about 1/2 mile from the test site. An analysis of the radiation hazards involved in a 1.8-msec-period violent destructive test indicates no hazard, and that under the worst conditions ample time would be available for orderly evacuation of all personnel in the vicinity of the test to further minimize any possible radiation exposure. 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. 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 and previous Spert I destructive test, which demonstrated the feasibility of core destructive testing under similar conditions, it is concluded that the Spert I Oxide Core Destructive Test Program can be conducted in a manner 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

1. PLANT SITE AND BUILDINGS

1.1 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 Figure A-1.

A general plan of the Spert site is shown in Figure 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.

1.2 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.

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.

1.3 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.

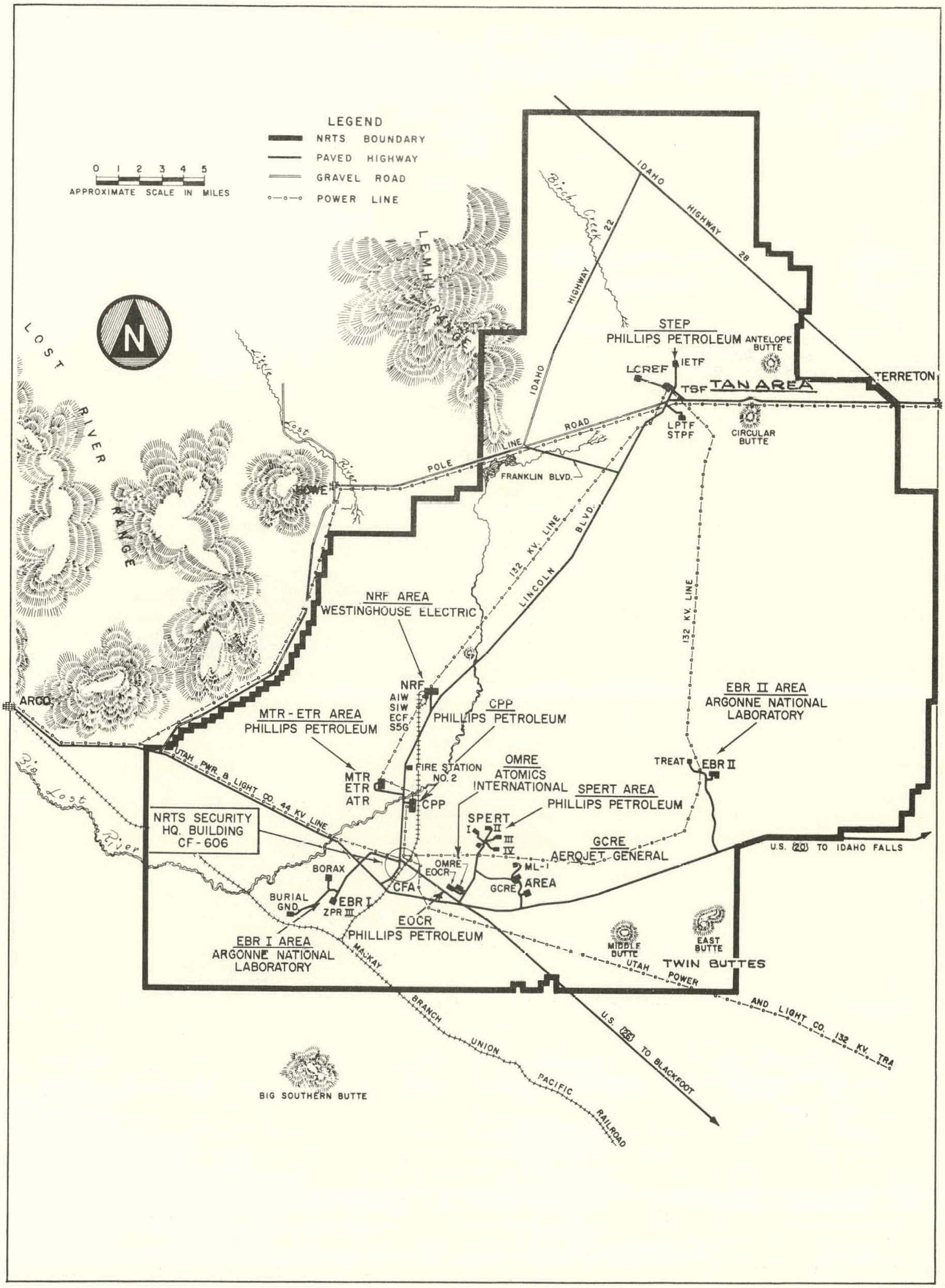

Fig. A-1 Location of Spert site with respect to other NRTS installations.

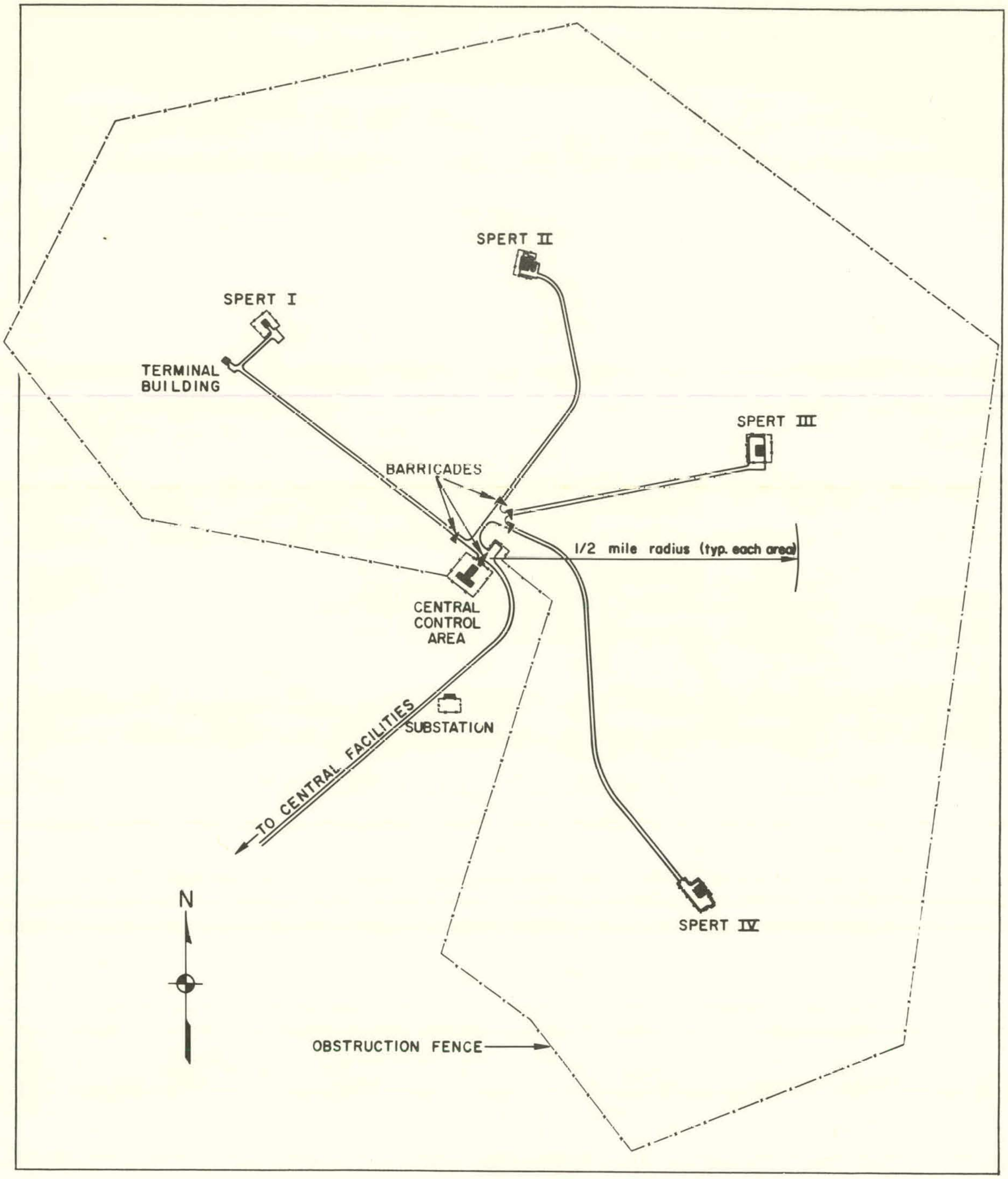

Fig. A-2 Layout of Spert site.

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 Figure A-3 and the floor plan is shown in Figure 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 12-gpm reactor fill

Fig. A-3 Spert I terminal building.

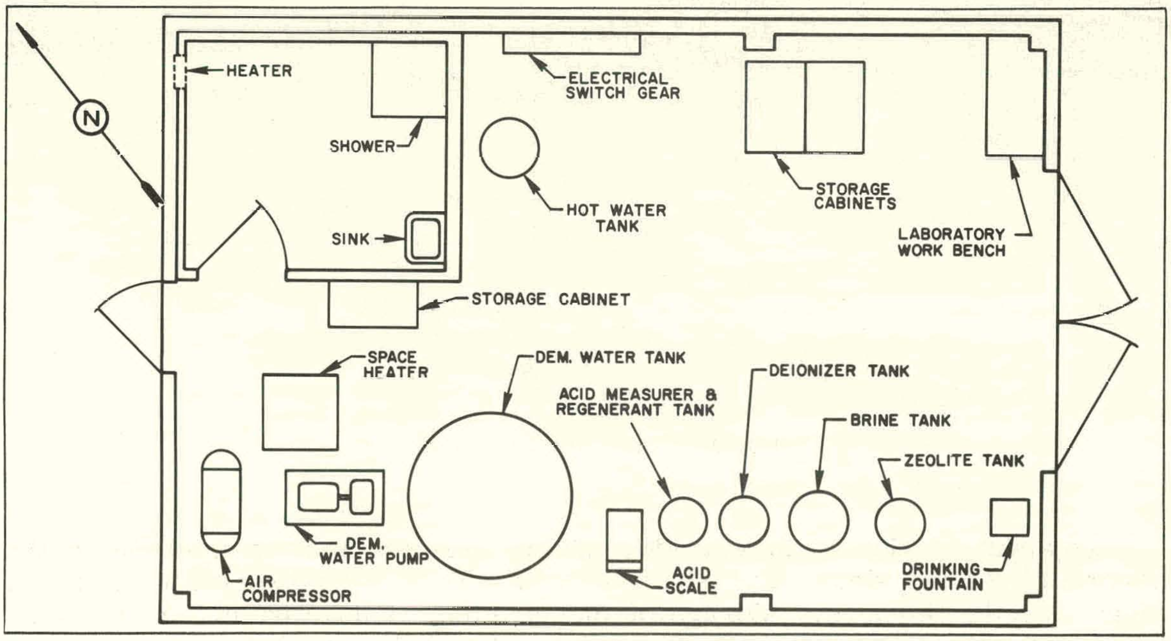

Fig. A-4 Plan view of terminal building.

pump, a 110-psig compressed-air system, and a personnel decontamination and change room.

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

1.4 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. The external view, plan and elevation are shown in Figures 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 are provided at each end of the building. The building is heated with an oil-fired, forced-air circulation furnace.

Fig. A-5 Spert I reactor building.

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

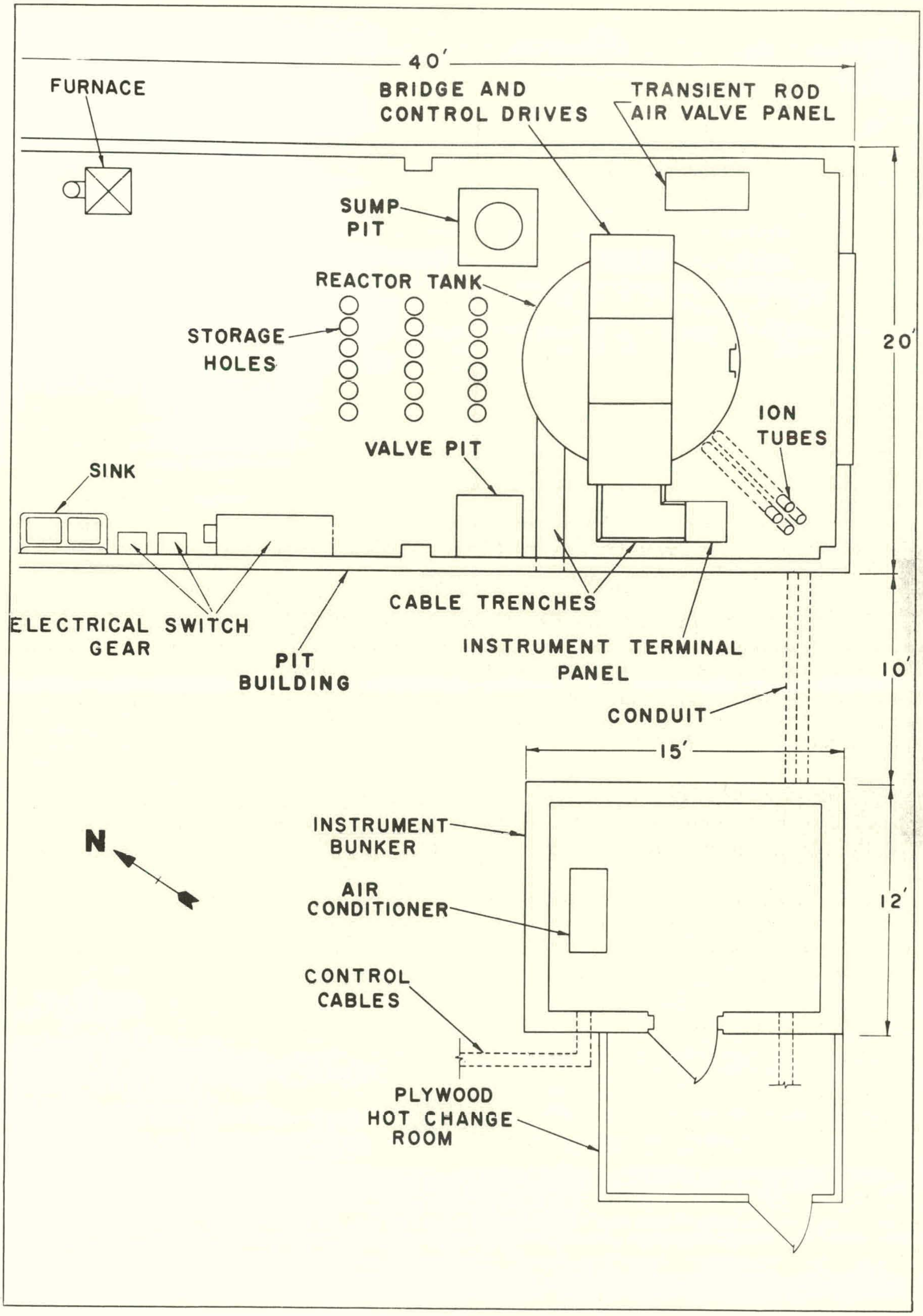

Fig. A-6 Plan view of reactor building.

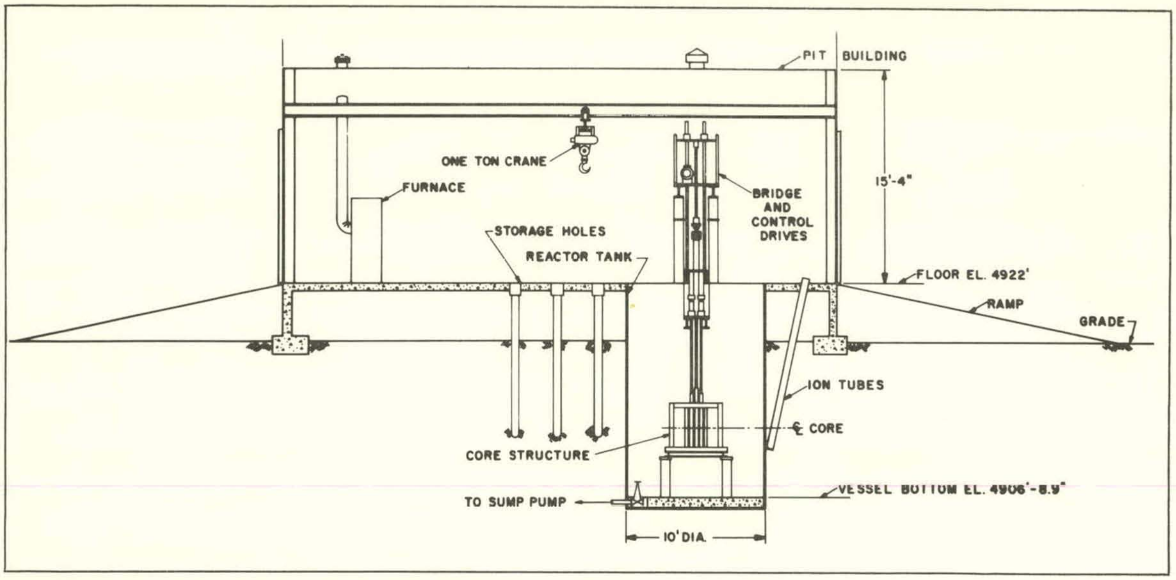

Fig. A-7 Elevation view of reactor building.

building floor to the reactor tank wall to a position approximately 2 ft above the concrete floor of the reactor tank.

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 and an air-operated valve on the deionized water reactor fill line. The other pit extend 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 ft^3 and is surrounded by an earth dike.

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 the health physics instrumentation load to the generator. The standby plant is rated at 2 kva, 2 kw, 1.0 power factor, 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.

1.5 Instrument Bunker

The instrument bunker, shown in the foreground in Figure 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 also is 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. The front of the block instrument bunker is enclosed by a plywood, hot change room. This room is 10-1/2 x 8 x 8 ft and houses the batteries for the intercom system, the air system for the poison injection system and the electrical interlocks for the poison injection system. This room is shown in Figure A-6.

1.6 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, also are controlled from this control room. The reactor control console is discussed in detail in Section 3.6 of Appendix A. An air conditioner supplying both the Spert I and Spert II control rooms protects the electronic instruments from overheating.

2. REACTOR COMPONENTS

2.1 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 pins, 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, core structure, upper and lower support bridges and the fuel pins. A detailed description of the control system including the control- and transient-rods and their associated drives, electrical circuits and nuclear instrumentation is given in Section 3 of Appendix A.

2.2 Reactor Vessel

The reactor vessel was fabricated in 1955 by the Western Steel Co. 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- x 4- 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, 48- x 15- x 1-in., 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 1-8NC-2 made of 18-8 stainless steel. Twenty additional bolts are welded to the base structure for use as instrument bracket supports. Figure A-8 is a photograph of the bottom of the reactor tank just prior to pouring the concrete.

The condition and physical dimensions of the reactor vessel were somewhat changed due to previous destructive testing. The concrete pad in the bottom of the vessel was separated from the vessel walls and required sealing. The vessel walls were bulged outward approximately 3 in. at a maximum between 3 and 6 ft from the bottom of the vessel. Small cracks in the vessel walls required welding to produce a water-tight seal.

Also pictured in Figure 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 screen 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 test.

Two 4-in. mild-steel channels, rolled to a 10-ft diam, 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 carbon steel tank was originally sandblasted and sprayed with two coats of Phenoline primer and three coats of Phenoline paint. The concrete vessel floor was originally hand brushed with two coats of Phenoline primer

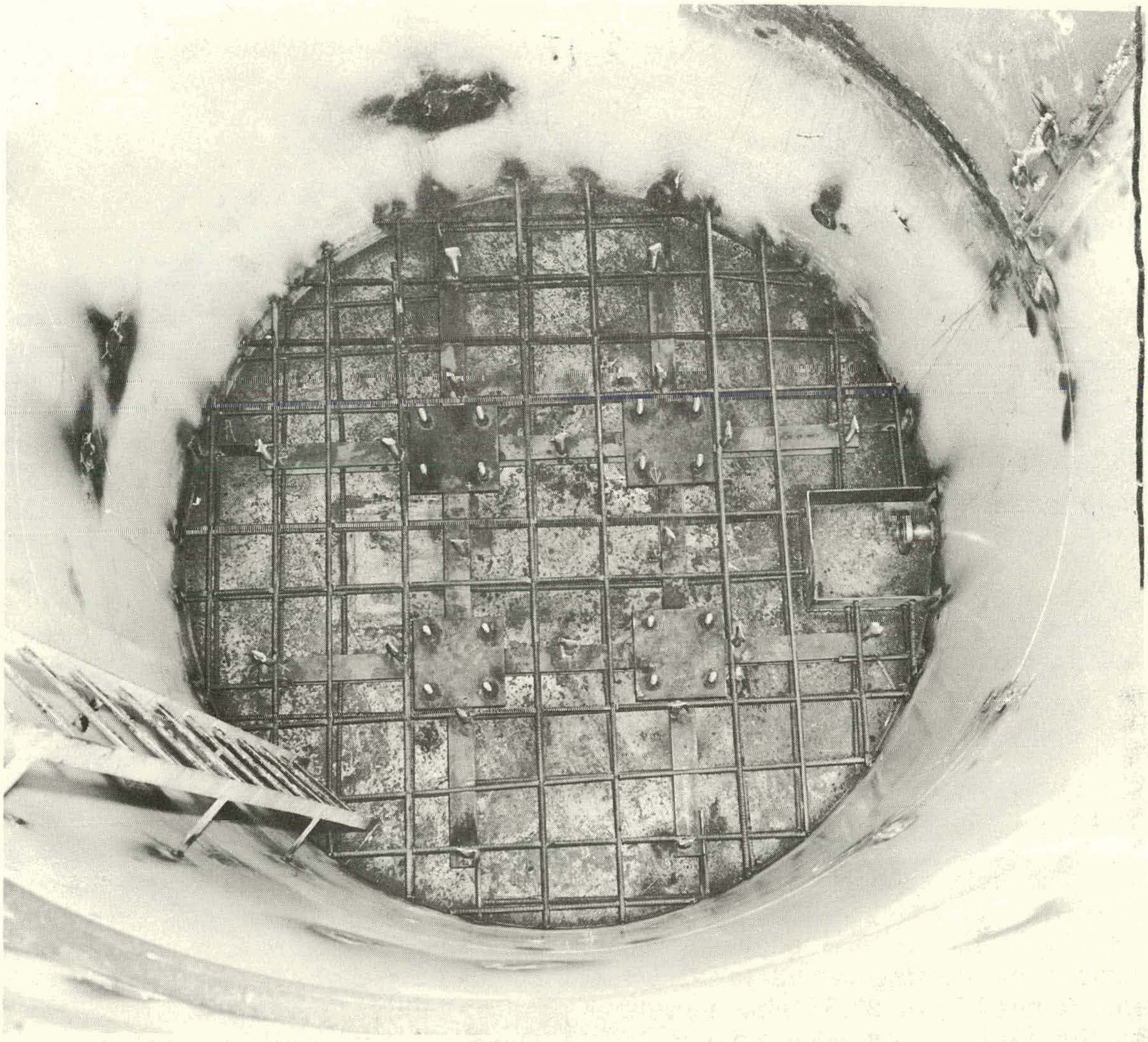

Fig. A-8 Bottom of reactor tank prior to pouring concrete.

and three coats of white Phenoline paint. After the destructive test on the D-type core, the loose paint on the vessel walls was chipped and wire brushed to prepare the surface. The vessel walls, floor, and core support were then brushed with one coat of orange Phenoline primer and two coats of white Phenoline paint.

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, two stirrer mounts, a fill pipe mount, mounting brackets for a poison injection pipe and mounting brackets for thermopile temperature measuring device.

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

2.3 Core Structure

The destructive oxide core structure consists of the core support assembly and core assembly.

The core support assembly is constructed of mild-steel main members and stainless steel hardware. The basic section of this assembly consists of four upright 6 WF 15.5 "H" beams 52-5/8 in. long placed in the shape of a square with 34-1/4-in. sides.

The upright members are connected by means of two 6 U 8.2 channel braces. The braces are bolted to the upright members by means of two 1/2-13 NC stainless steel bolts. The upright members are welded to 14- x 14- x 1-in. mild-steel plates. These plates are bolted into the vessel floor by means of the sixteen 1-8NC stainless steel core support column anchor bolts. One inch mild-steel plate 6-1/2 x 8-1/8 in. are welded to the upright members. These plates are located horizontally on the upper end of the upright members and are used to bolt the core assembly in place.

The core assembly consists of three basic parts: the support weldments, the grid structure, and the guide structure. The major components of this assembly are fabricated from 6061-T6 aluminum and the incorporated hardware is 18-8 stainless steel.

The grid support weldment is fabricated from 3 U 1.46 aluminum channel. The structure is in the form of a parallelopiped of dimensions of 28-1/8 x 28-1/8 x 13 in. The short dimension of 13 in. consists of four square posts formed by welding two 13-in. sections of channel together. The long dimensions are formed by a 23.971-in. section of channel with 45° beveled corners welded to the 13-in. posts. The support leg weldment is fabricated from two 3 U 1.46 channels 30-5/8 in. long welded into a square. The support leg weldment has a 3/4-in. plate offset from the square post located 24-1/4 in. from the bottom of the post used to link the core assembly to the core support assembly by means of a 1-1/16-in. bolt. The support leg weldment is bolted to the grid support weldment by means of a 5/8-in. plate weldment to the support leg weldment. Twelve 3/8-16NC bolts are used for this connection at each of the four corners.

The core structure consists of the upper grid plate, the intermediate grid assembly, the lower grid plate, and the transient and control rod blade guides.

The lower grid plate is machined from a solid plate into an octagon with 29-3/4-in.-wide flats and 1-1/2 in. thick. One-half-inch slots are milled into the plate to facilitate the control rods and the transient rod. Four holes are drilled on the outside tip of each control rod and 269 holes are drilled in each quadrant of the plate. There are 1076 of these holes drilled 1/4 in. through 0.516 c'bore one in. deep with 1/16 x 45° chamfer and the 16 holes are drilled 1/4 in. through 0.516 c'bore 3/4 in. deep with 1/16 x 45° chamfer. These holes are to hold the lower end of the fuel pins. The lower grid plate is bolted to the grid support weldment by sixteen 3/8-in.-16NC bolts.

The upper grid plate is machined out of a solid section of 6061 T6 aluminum to the same dimensions as the lower grid plate. It is, however, only 5/8 in. thick. The fuel pin hole pattern is identical with that of the lower grid plate with the exception of the hole dimensions. All holes for fuel pins in the upper grid plate are drilled 0.516 in. through with a 1/16 x 45° chamfer. Figure A-9 shows the pin hole pattern for one quadrant.

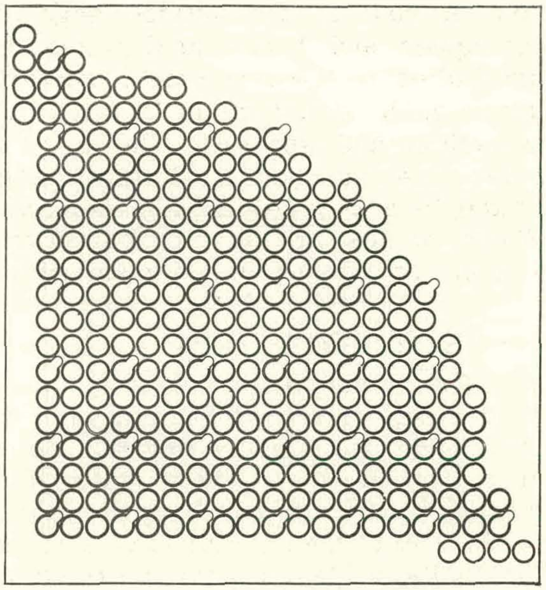

Fig. A-9 Pin hole pattern for one quadrant.

In previous operation of this oxide core, it was found that on short period transients there was considerable bowing of the fuel pins. This bowing occurred in the area of peak neutron flux or approximately 17 in. above core bottom. To counteract this bowing effect, an intermediate grid assembly will be utilized. This intermediate grid is divided into four independent sub-assemblies, one located in each quadrant of the core and separated by the control rod guide housings. Each subassembly consists of a pair of interlocking combs and a comb support structure. All major components of the intermediate grid are constructed of 6061 aluminum.

Each comb is made up of thirteen teeth 9.288 x 1/2 x 0.148 in. inset 1/4 in. into a 8.375 x 1 x 1.357 in. base. Each comb is fastened to the comb support structure by a single 3/8-16 NC anchor bolt. The teeth of each comb are spaced 0.663 in. center to center.

The combs may be set in place and removed while the core is fully loaded. A pair of matching combs are inserted into the fuel pins with the combs at 90° relative to each other. This forms a 0.633 in. square about each fuel pin allowing a maximum movement of 0.064 in. including tolerances.

Each pair of combs are held in position by a comb support plate. This plate is 2 in. wide x 1/2 in. thick encompassing the outer row of fuel pins in each quadrant. Each plate contains fourteen through holes 17/32 in. diam extending perpendicular to the control rod housings, seven extending from each housing in a quadrant. Six milled openings of 17/64-in. radius, face the vertical core center line and are located at 45° from each control rod guide housing. The comb support plates are held rigid by means of two 3/8-16 NC bolts and nuts through each control rod housing and a single support rod 19-1/2 in. long x 5/8 in. diam between each plate and the lower grid. Each comb support plate is fixed and cannot be placed or removed with water in the vessel.

The transient and control rod guides are constructed in two sections, the upper guide section and the lower guide section. Both sections are constructed of 6061 T6 aluminum with 18-8 stainless steel hardware. The upper guide section is formed by bending a sheet of aluminum 69-3/16 x 29-1/2 x 0.10 in. into an angle 69-3/16 in. long x 0.10 in. thick with 14-3/4-in. legs. Four of these angles are then placed into the shape of a cross with leg separation of 0.525 in. This spacing is obtained through spacer bars placed between the transient rod and control rod, at the outside edge of the control rod and at the outside edge of the upper and lower grid plates, in each leg of the structure. These spacers are bolted to the upper and lower grid plates. The lower guide is similar to the upper guide structure in that it is formed in the same manner using the same materials and spacers. The total length, however, is 20-3/4-in.-diam vent holes are punched on 1 in. alternating centers. The lower guide section is bolted to a 2- x 1-1/2- x 3/16-in. angle 9-1/2 in. long on each side of each leg. These angles are in turn bolted to the lower grid plate. The guide plates and spacers of both upper and lower guides are bolted together.

2.4 Lower Support Bridge

The lower bridge support is 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 8 in. above 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 x 19-7/8 x 1-3/8 in. thick and has four 2-13/16-in.-diam holes to accommodate the control rods and one 4-1/16-in.-diam 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.

2.5 Upper Support Bridge

The carbon steel upper bridge structure, which supports the control rods and transient rod drives, is pictured in place in Figure A-10.

The upper bridge portion fastened 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.-diam 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 x 2-1/2 ft high x 3-1/3 ft wide and is equipped with guard rails and gratings. A detailed description of the rod drives is contained in Section 3.2 of Appendix A.

2.6 Fuel Rods

The fuel rods comprising the Spert I experimental core were those previously used in the Babcock and Wilcox N. S. Savannah critical assembly. The fuel rod is a 6-ft long, welded-seam, 304 stainless steel tube, containing low-enrichment UO_2 powder, swaged-compressed to an effective density of approximately 87% of the theoretical density of UO_2. The fuel element design parameters are given in Table A-I.

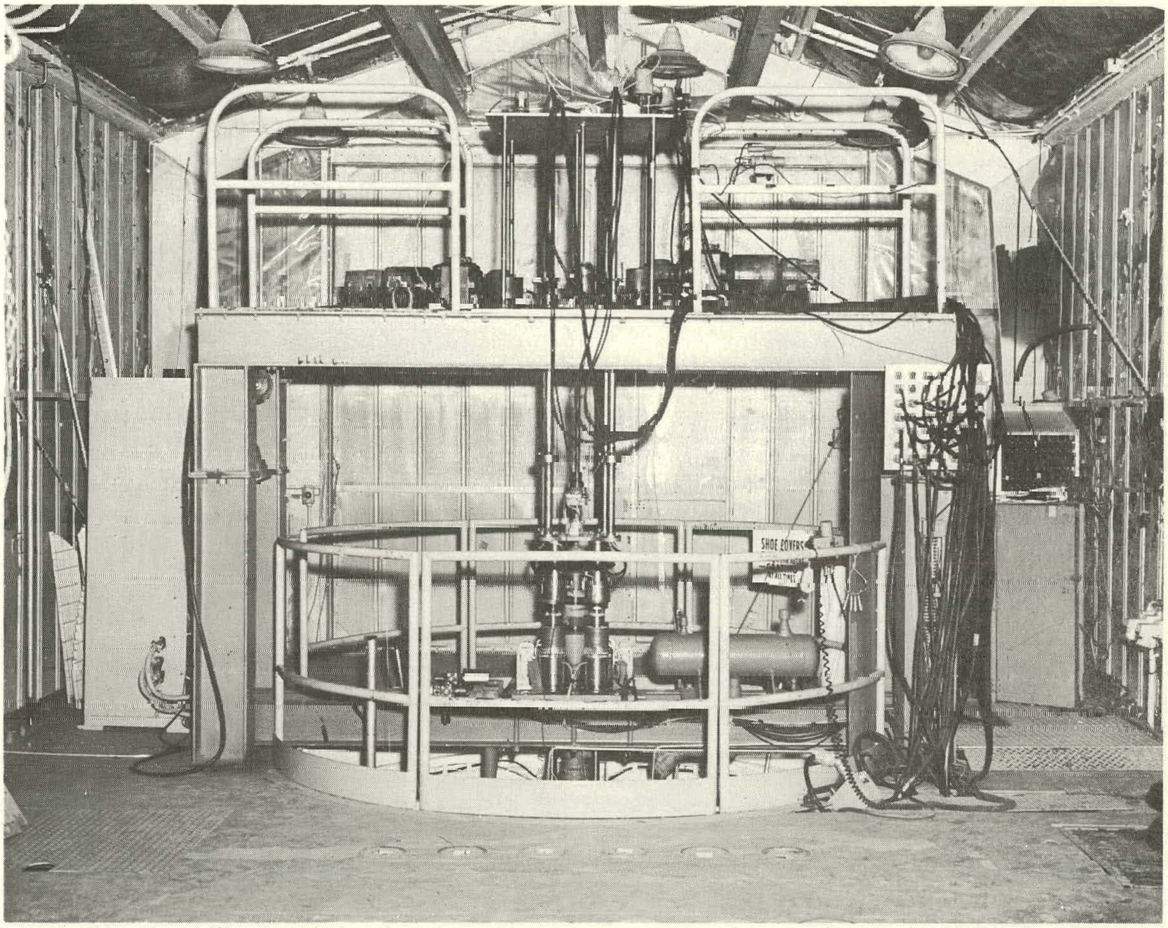

Fig. A-10 View of interior of reactor building.

TABLE A-I

N. S. SAVANNAH - TYPE FUEL ROD PARAMETERS

Total length of fuel rod	71.5 in.
Active length	66.9 in.
Outside diameter of fuel rod	0.500 in.
Cladding thickness	0.028 in.
Fuel	Compressed UO_2 powder
Fuel enrichment	4.02 Wt% U-235
Effective UO_2 density	9.45 g/cm^3
Mass UO_2 per fuel rod	1600 g
Mass U-235 per fuel rod	56.7 g
Mass U-238 per fuel rod	1353 g

3. REACTOR CONTROL

3.1 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 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 withdrawals of rods removes neutron-absorbing material. In some core designs the rods also include a fuel follower so that control rod withdrawals also add 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 the 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 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.

3.2 Control Rod and Transient Rod Drive Systems

3.21 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 coupled 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 Figure A-11.

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 absorber contact, is about 300 msec.

The transient rod drive is connected to the transient rod through an air operated latch. As a transient is initiated the transient rod and shafting below the transient rod latch are accelerated by an air piston arrangement. This accelerated drop reduces the travel time between upper rod limit and the transient rod shock absorber contact to 80 msec.

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.

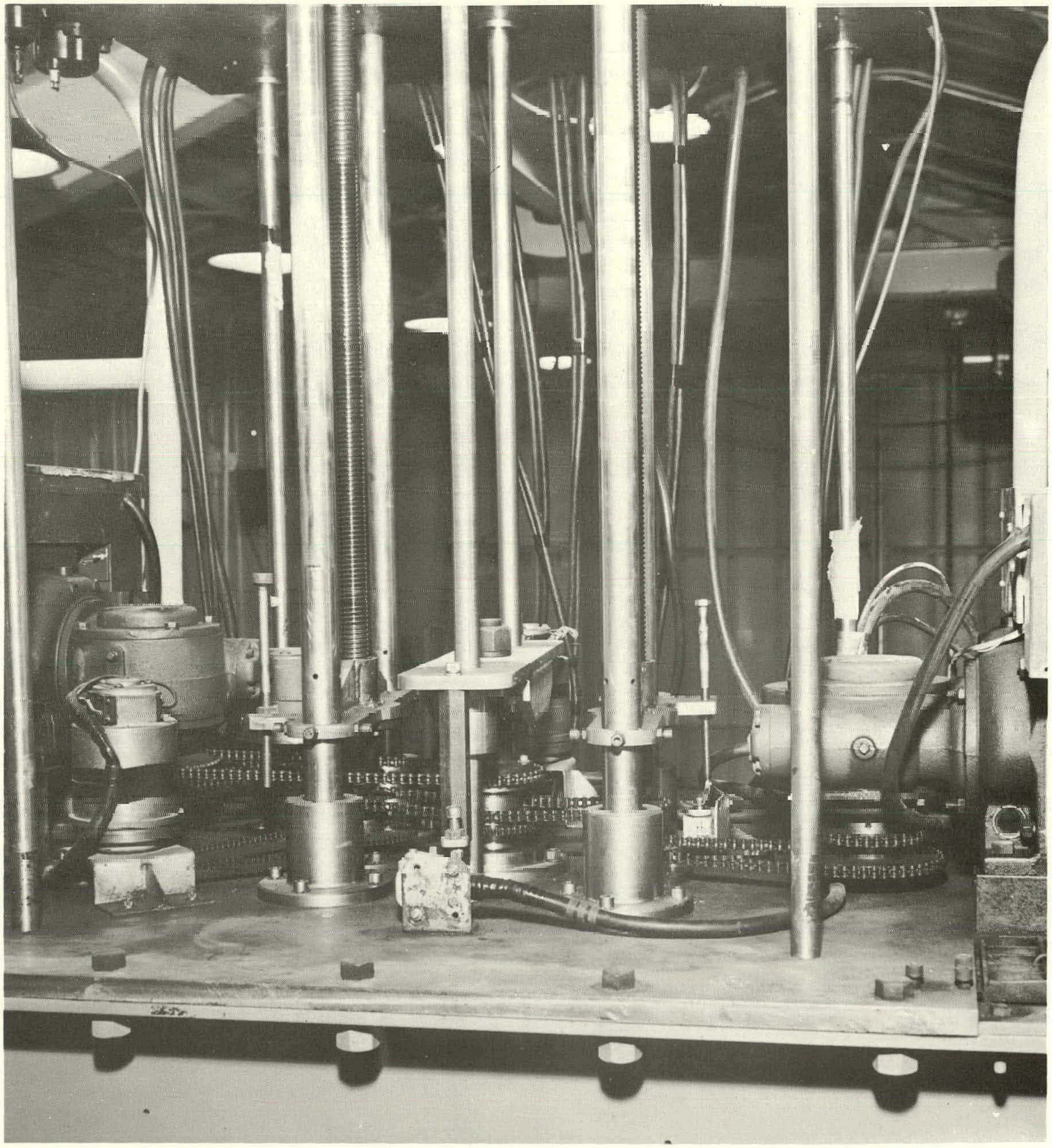

Fig. A-11 Control and transient rod drive system.

3.22 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.-diam 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 or 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 Figure A-12. 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 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.

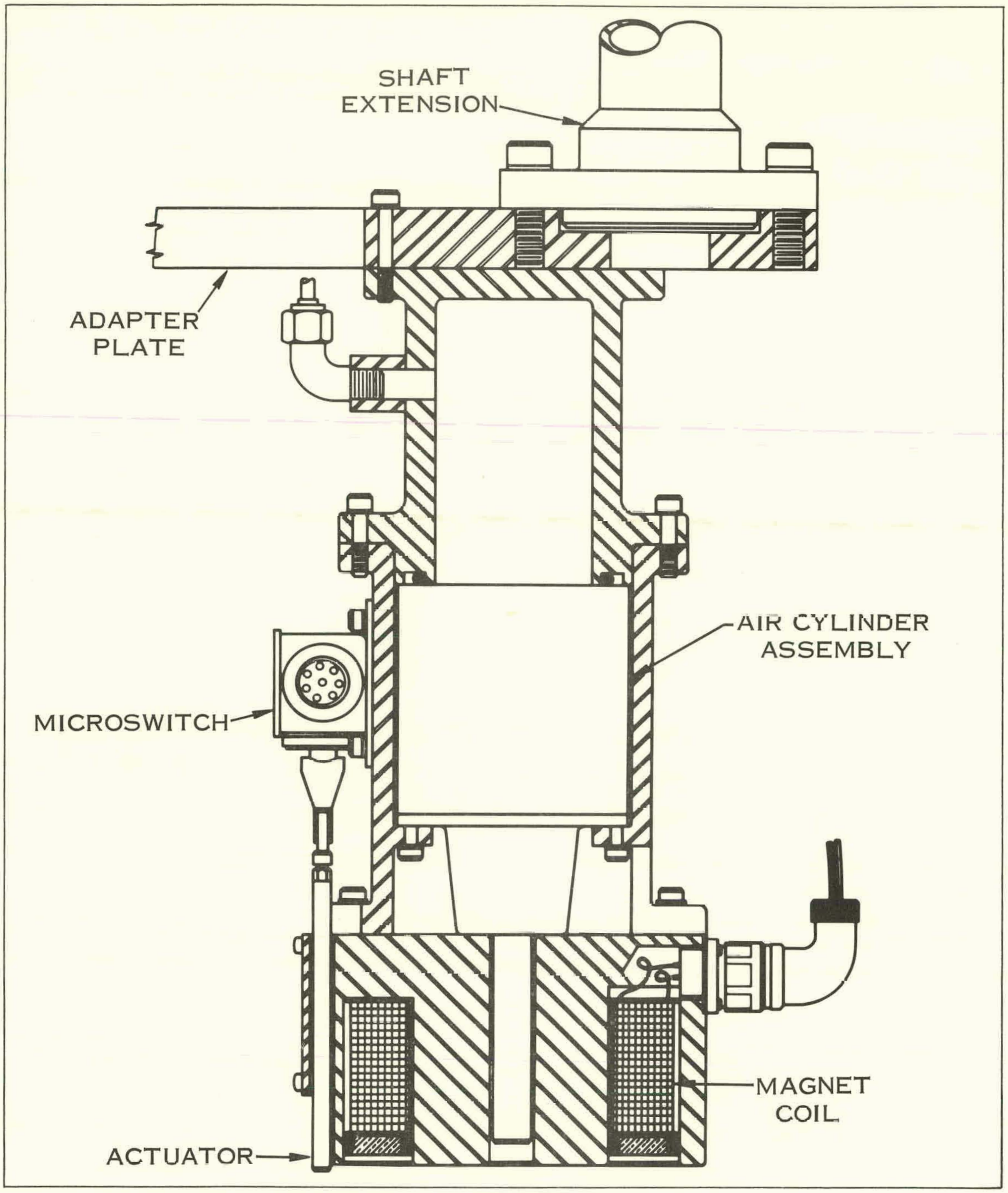

Fig. A-12 Control rod magnet.

When a magnet is in contact with the armature, a microswitch 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.-diam 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 ejection system is comprised of four basic parts. The spring shock, shock absorber, and air cylinder and piston sections are all included in one assembly while the air supply system is a separate assembly. Figure A-13 shows the transient rod spring shock, shock absorber, and air cylinder and piston sections.

In operation of the transient rod, the shafting linkage between the transient rod and the transient rod drive system is accomplished through the air operated latch (See Figure A-14). This latch is in the locked position until just prior to firing time. The air piston is a two-way piston with hold air on the lower side of the piston and fire air on the upper side of the piston. In operation, the hold air is set 20 psi higher than the fire air. This assures correct positioning of the transient rod prior to initiation of a transient. Initiation of a transient is accomplished by opening the latch connecting the transient rod and transient rod drive system. After opening the latch the hold air is quickly expelled from the system and the transient rod is accelerated downward by the fire air. This downward acceleration is active over the first 18 in. of travel of the transient rod. After the 18 in. of downward acceleration, the transient rod is decelerated for an additional 8 in. providing 26 in. of total rod travel.

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

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

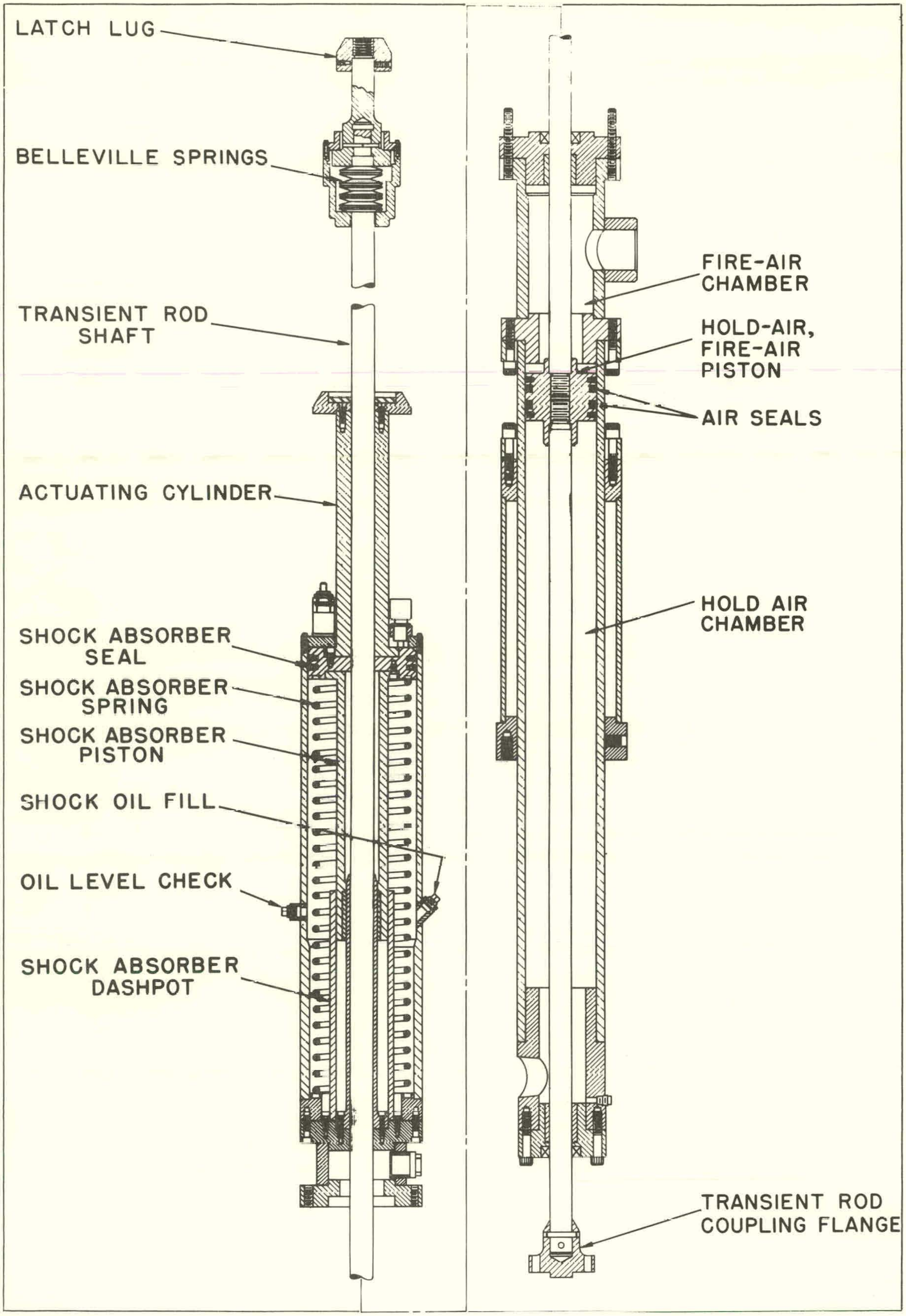

Fig. A-13 Transient rod piston and shock absorber mechanism.

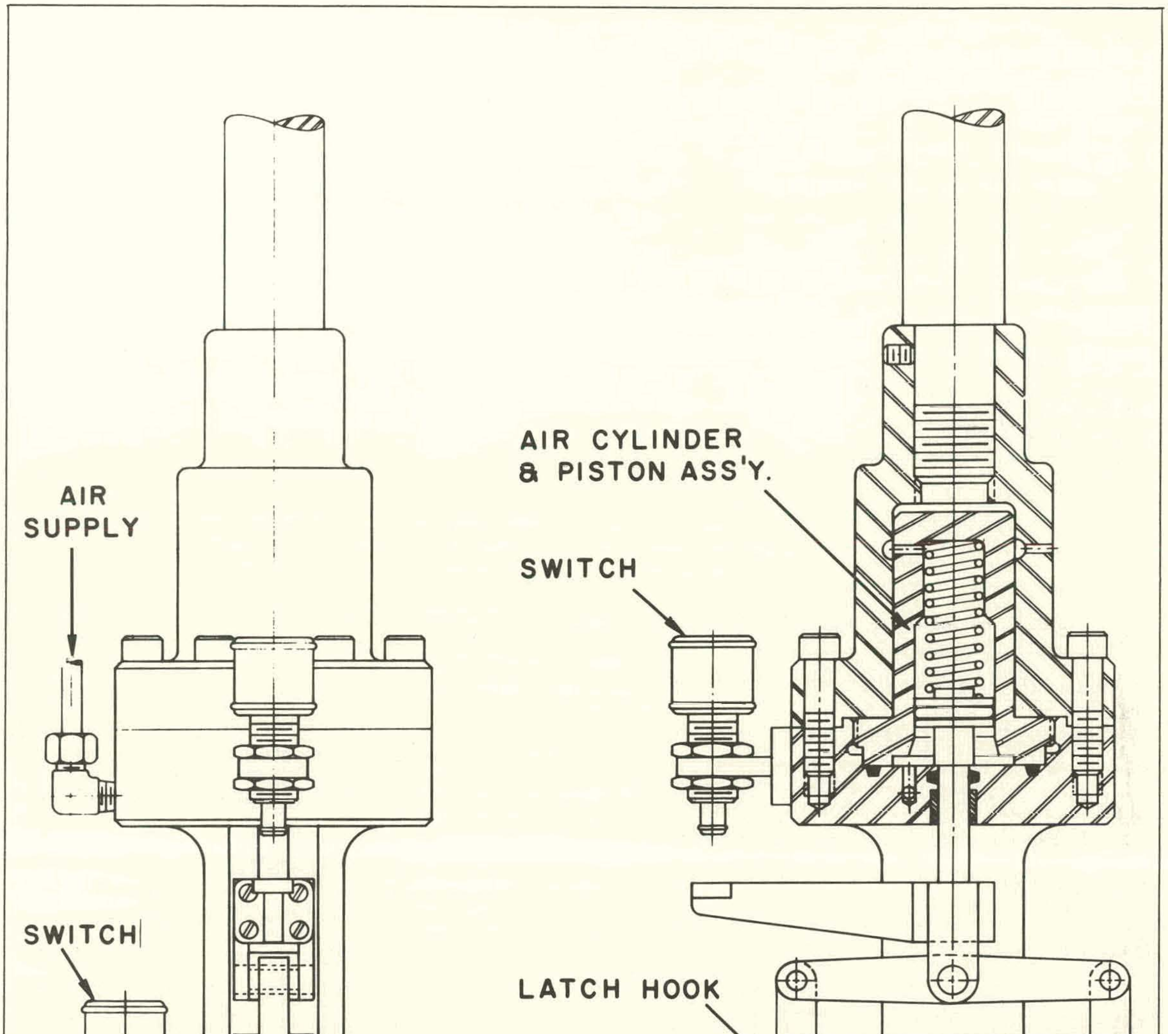

Fig. A-14 Transient rod latch.

The transient rod scram system is designed to operate with a maximum of 500 psi pressure. This pressure produces a drop time of approximately 80 msec for 18 in. of accelerated travel.

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

(7) Control Rod Shock Absorbers - Deceleration of the control rods 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 in up. Air is allowed to escape through 3/16-in. grease fittings during the compression of the shock absorber. Figure A-15 shows the shock absorber assemblies.

3.3 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, one poison (neutron-absorbing) blade and one aluminum-follower blade.

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.-diam, 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.

A 3/4-in.-diam, 63-7/8-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 also is pinned.

The poison and follower section of the control rod consists of one flat blade securred to a yoke by six 1/4-in.-diam x 1-1/4-in.-long CSK Hd rivets of 304 stainless steel. These rivets are cold-headed.

Each control rod blade is made up of two sections: the follower section is constructed of 6061-T6 aluminum and the poison section is of Binal [*] containing

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(*) Trade name for the Sintercast Corporation Aluminum-Boron powder-metallurgy processed materials.

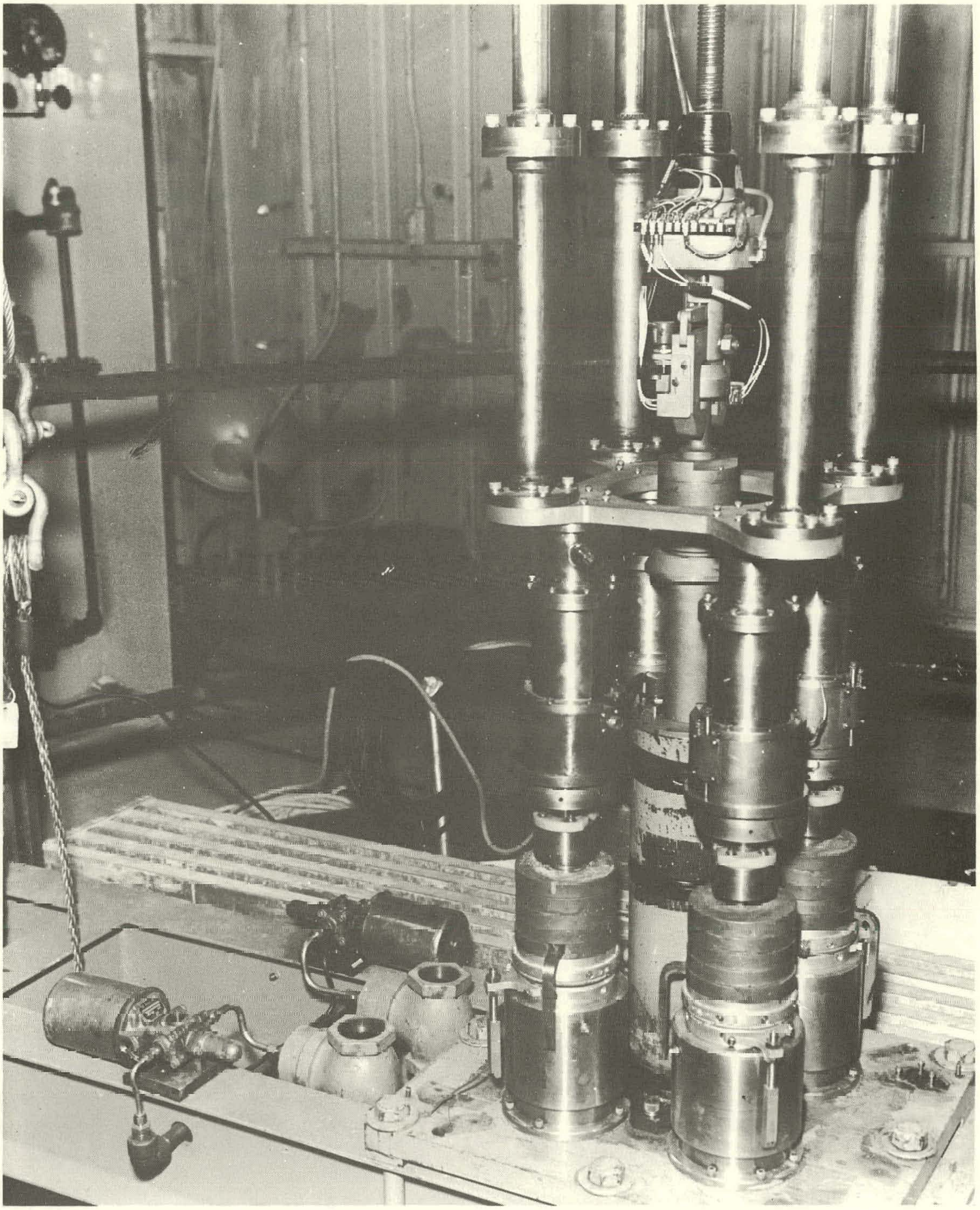

Fig. A-15 Control rod magnets and shock absorbers.

$\approx$ 7 wt% boron. The aluminum follower section is 43.7 x 7-1/4 x 3/8 in. in a solid section. The poison section is constructed of three strips of Binal formed into a solid section 55-7/8 x 7-1/4 x 3/8 in. These Binal strips are

2-3/4-in. outside strips and 1-3/4-in. inside strip. The follower section and the poison section are joined by a riveted butt joint utilizing two connecting plates 6-7/8 x 1-3/4 x 0.094 in. of 304 stainless steel and sixteen 1/8-in. diam x 3/4in.-long CSK Hd rivets of stainless steel.

Each control rod operated in a guide slot. These slots are included in the description of the core structure. Figure A-16 shows the control rod.

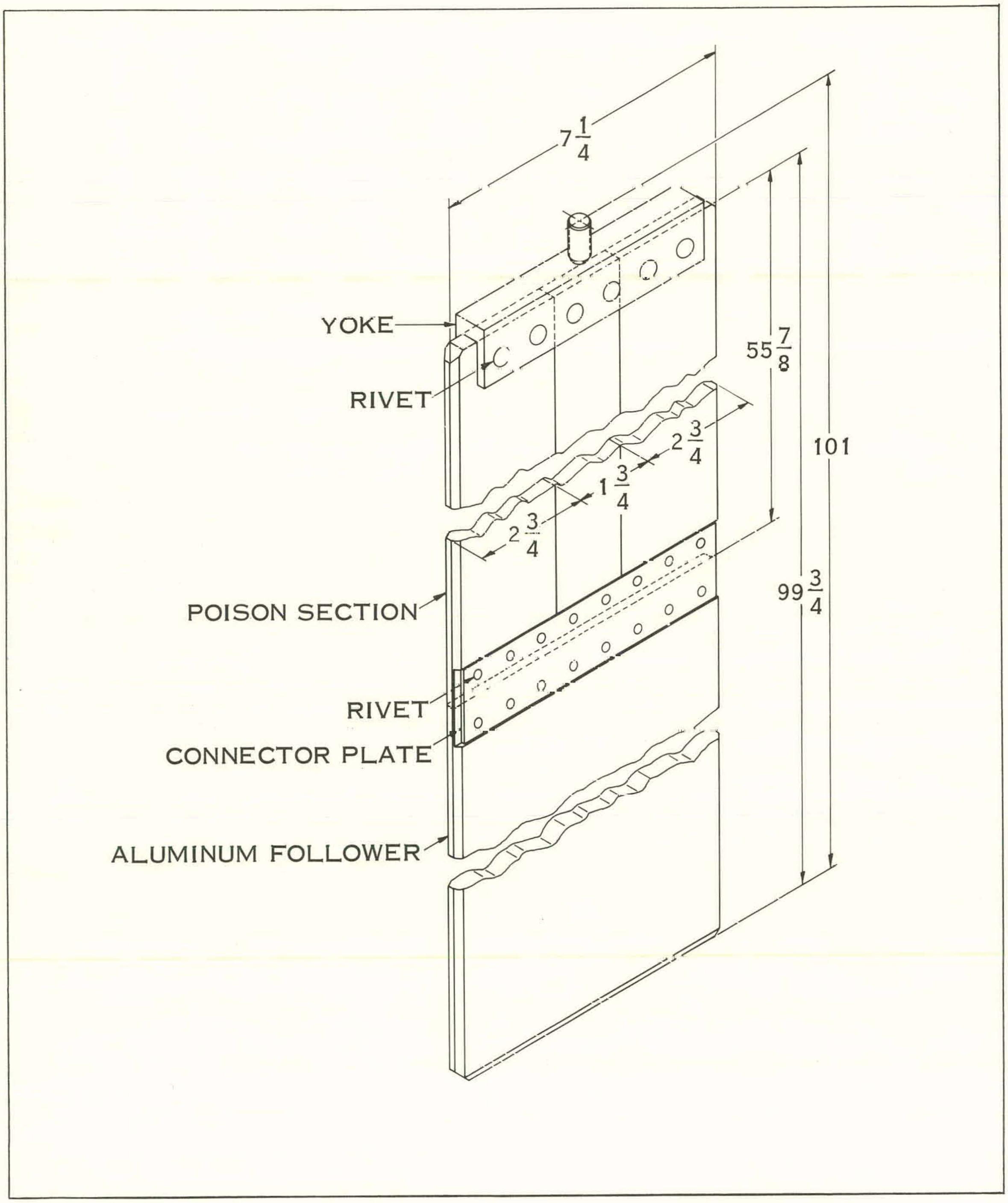

Fig. A-16 Control rod blade.

3.4 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 drive system, the scram system, a connector, a yoke, an aluminum section of blade, and a poison section of blade. The drive system and scram system have been previously described.

The connector couples the yoke of the transient rod to the shaft of the transient rod scram system. This section is machined from a section of 304 stainless steel. The body of the connector is 1-1/4 in. diam x 4-3/32 in. including a 3-in.-diam x 1-in.-thick flange coupling. This flange coupling is joined to the transient rod scram shaft by a mating flange and six 3/8-16 NC-2 socket head stainless steel cap screws. The end of the connector opposite the flange contains a single 3/4-10 NC-2 female thread 1-5/16 in. deep to enable coupling with the transient rod yoke.

The transient rod yoke is machined from 304 stainless steel stock. The entire yoke forms one integral part with the body or lower section 2-3/4 x 1-3/8 x 3/4 in. The upper section of the yoke consists of a 3/4-10 NC-2 male thread 1-1/8 in. long to connect the yoke to the connector. The body of the yoke has a milled slot 2-3/4 x 1 x 0.375 in. into which the upper section of the transient rod may be inserted. The yoke and transient rod are jointed by three 1/4-in.-diam x 1-1/4-in.-long countersunk Hd rivets of 304 stainless steel.

The aluminum section or upper section of the transient rod is milled from a solid section of 6061-T6 aluminum bar into a cruciform shape 71-7/8 in. long. The legs of the cruciform are 1-3/8 in. long x 3/8 in. thick. The upper portion of the aluminum section is a single flat 1-1/4 in. long x 3/8 in. thick which is inserted into the yoke. The bottom 3/4 in. of the cruciform section is recessed 0.078 in. on each surface to facilitate a 13-gage stainless steel plate bent into an angle. This angle or connecting plate joins the aluminum upper section to the poison lower section of the transient rod by means of 1/8-in.-diam x 3/4-in.-long countersunk Hd 304 stainless steel rivets. Four of these plates and sixteen rivets are utilized to form this butt joint.

The poison or lower section of the transient rod blade is fabricated from two pieces of Binal 26-5/8 x 2-3/4 x 3/8 in. The lower section of one of these pieces has a 0.375-in.-milled slot 13-5/16 in. long up from the bottom of the piece, and the other piece has an identical slot milled down from the top. These two pieces are then slid together to form the required cruciform shape. The upper 3/4 in. of the poison cruciform shape is recessed 0.078 in. to mate with the aluminum or upper section of the blade. The lower 3/4 in. of the poison section also is recessed 0.078 in. to facilitate another connector plate which holds the lower ends of the poison section rigid. Two of these connecting plates are utilized. These plates are riveted to the blades by eight 1/8-in.-diam x 3/4-in.-long countersunk Hd 304 stainless steel rivets (See Figure A-17).

3.5 Control System Electrical Circuits

3.51 Power Supply. The NRTS electrical power standard for applications up to 100 hp is 480 volts, 60 cycles, 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-amp, air-circuit breaker. This bus is fed directly from the Spert I substation through a 400-amp air-circuit breaker.

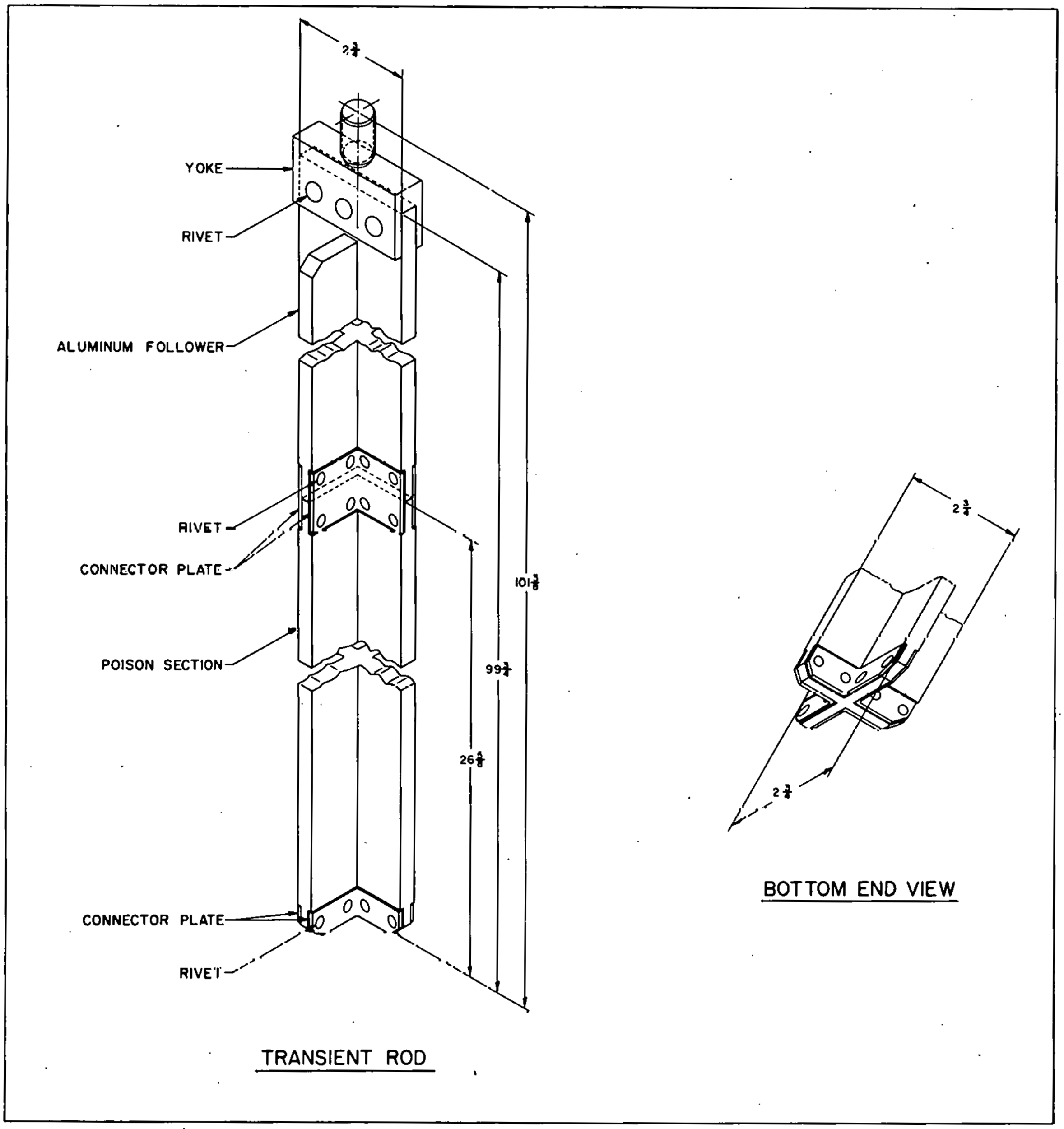

Fig. A-17 Transient rod blade.

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-amp 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 Figure A-18, the main power key switch on the control console controls the main power relay.

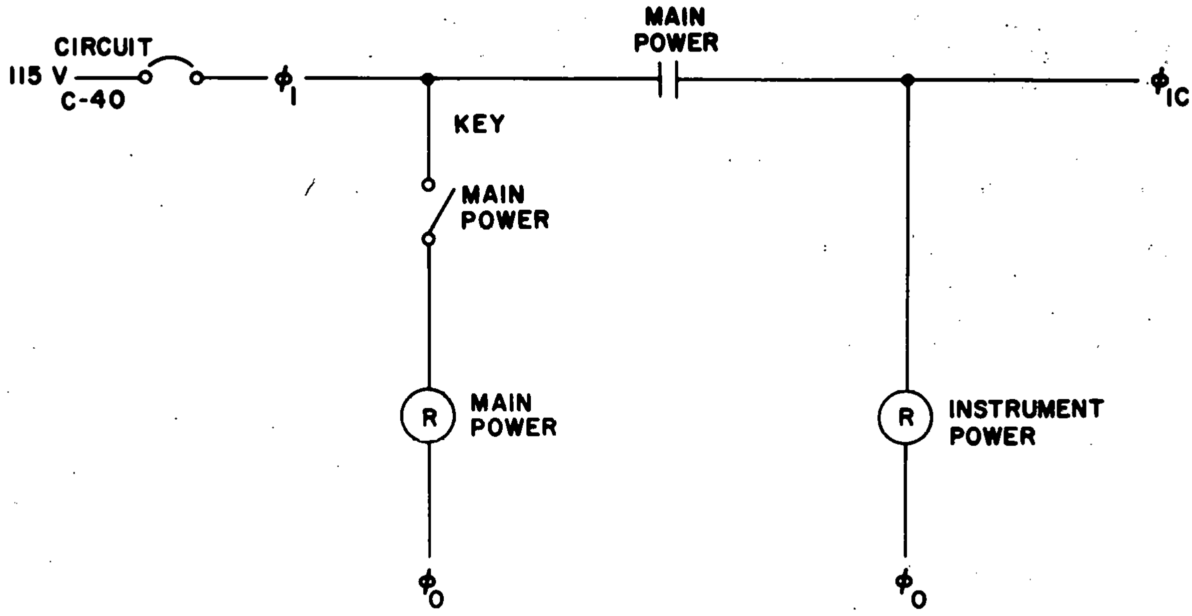

Fig. A-18 Main power circuit.

3.52 Rod Drive Insert-Withdraw Circuits. Standard NEMA size-00 reversing motor starters are used to control the 1/2-hp, 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 (Figure A-19). 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

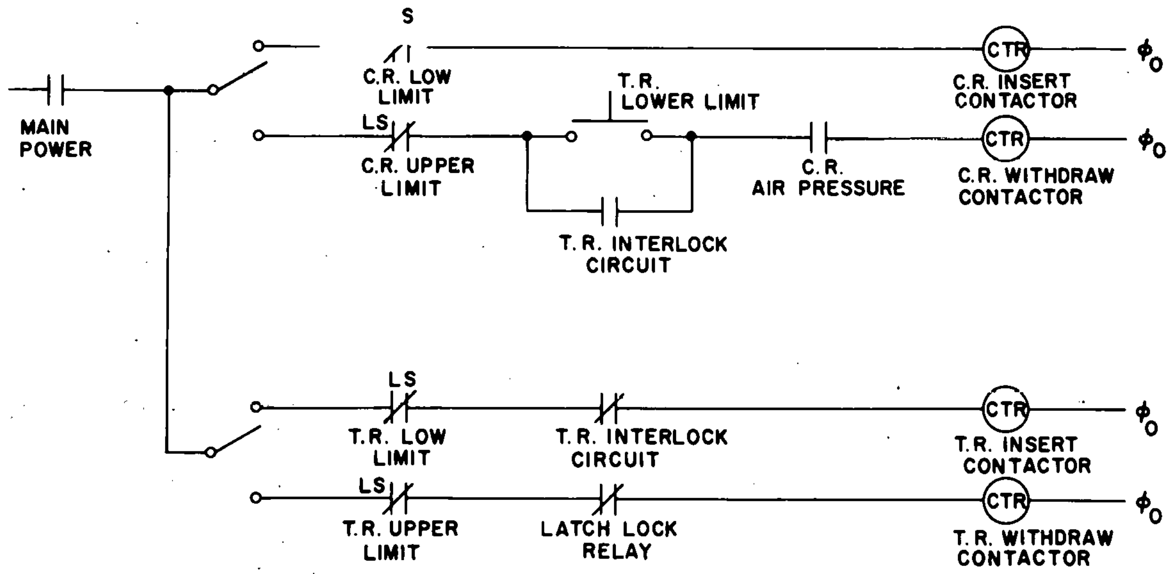

Fig. A-19 Control and transient rod insert-withdraw circuits.

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.

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 in Figure A-19 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.

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 (Figure A-19). 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 equipped with electromagnets to couple the control rods to their drives. The magnets are individually controlled so that selective withdrawal of individual control rods is possible.

3.53 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 an ammeter 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 a timer relay operated by the sequence timer provide for programmed scrams.

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.

Figure A-20 shows the control system magnet control circuits.

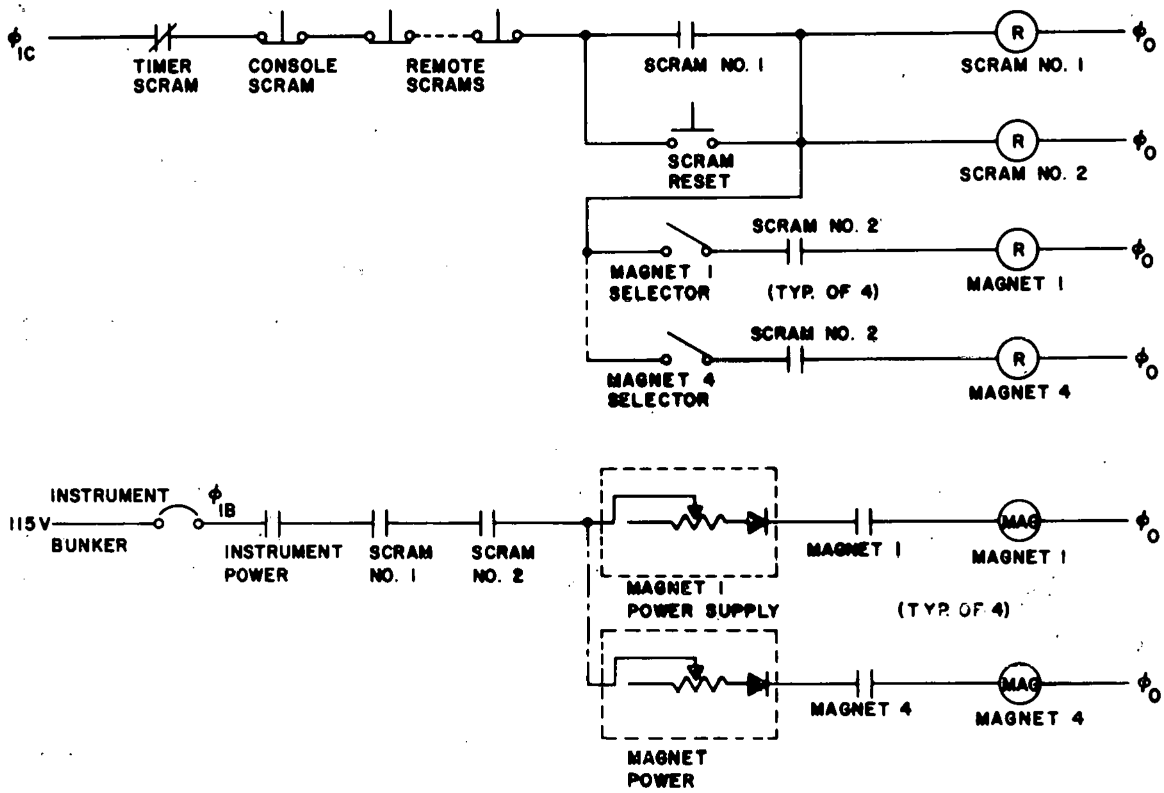

Fig. A-20 Control rod magnet circuits.

3.54 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. Figure A-21 shows the control rod piston air control circuits.

3.55 Transient Rod Air Control Circuits. The transient rod air control circuits are shown in Figure A-22 and A-23. Air to the system is controlled by a key operated switch which actuates the transient rod air master control solenoid. Transient rod fire air is applied to the top side of the piston by a push button on the console actuating a relay and the transient rod fire air control solenoid valve. Hold-air is applied in the same manner, except that certain interlocks exist to prevent damage to the system. As shown, hold-air to the piston cannot be obtained if the transient rod fire relay or the transient rod vent relay is energized. Also, the fire-air set relay, actuated by a pressure switch, and the fire-air relay both must be energized to obtain hold-air. If it were possible to apply hold-air without fire-air on the transient rod there exists the possibility of damaging the drive system by the large upward force.

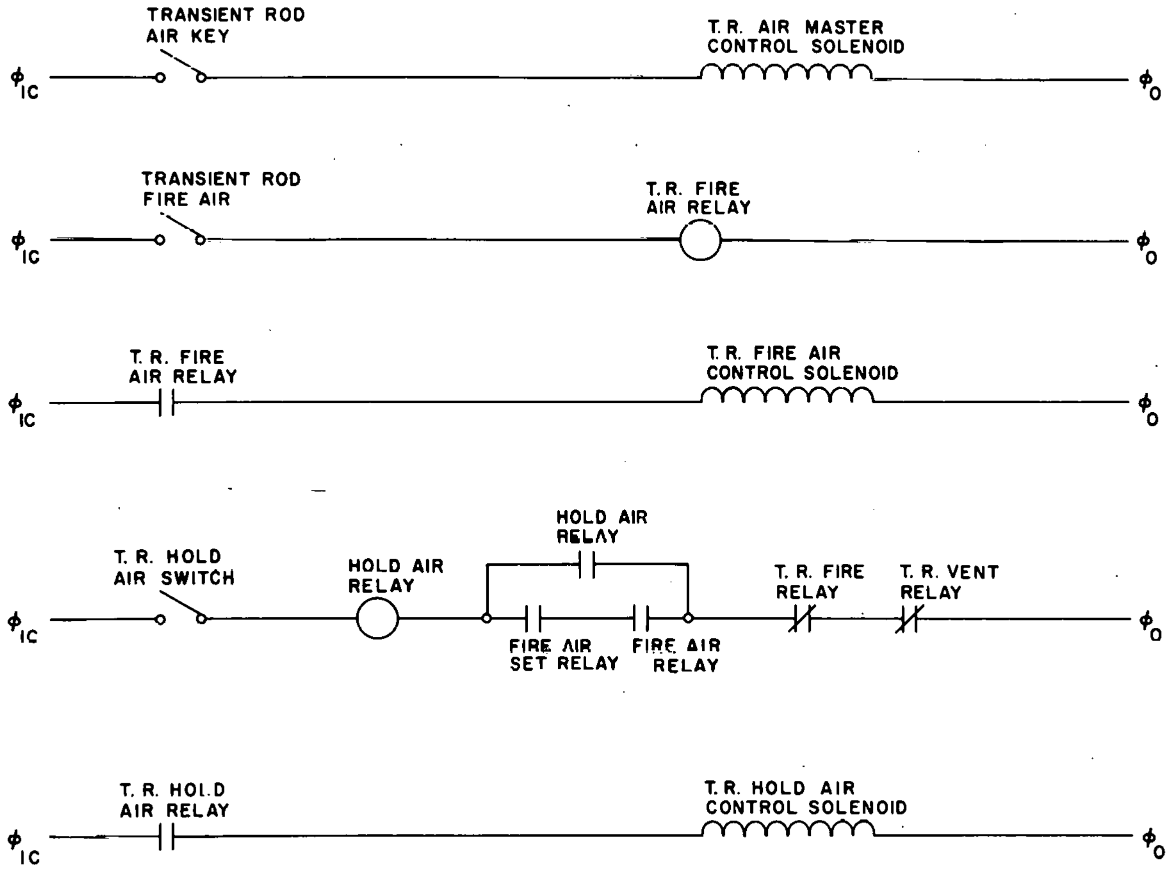

Fig. A-21 Transient rod piston air control circuits.

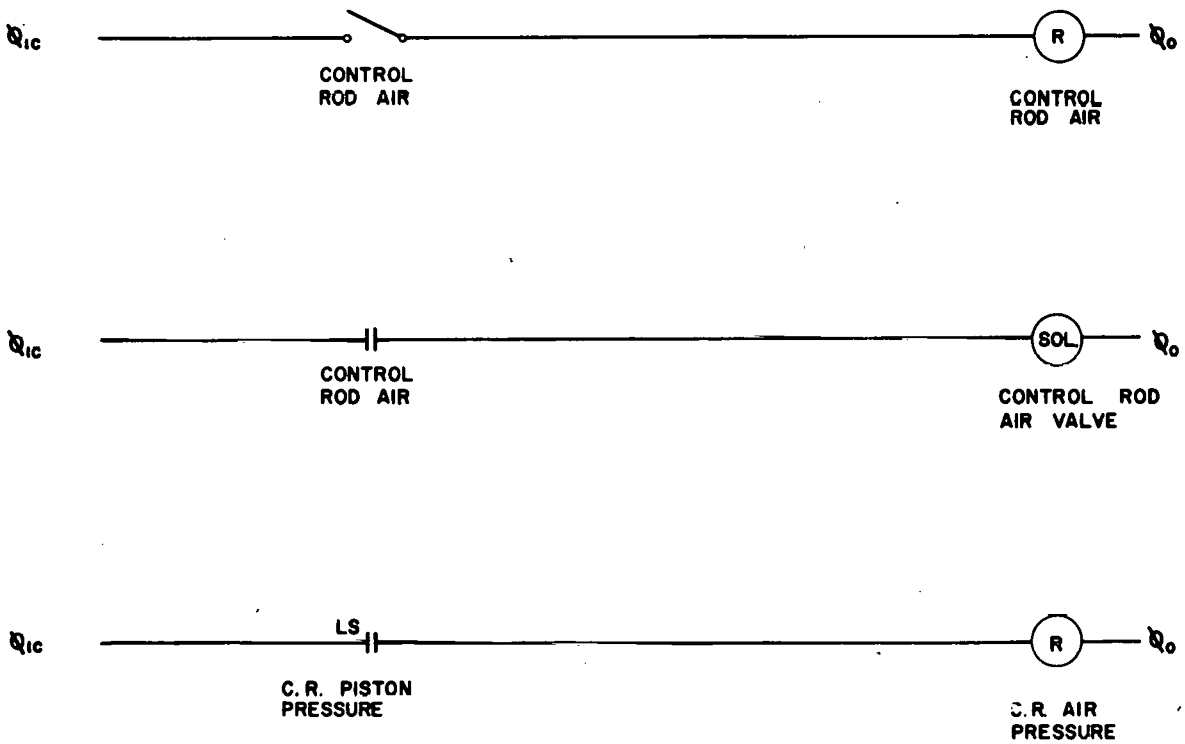

Fig. A-22 Control rod piston air control circuits.

The fire-air vent solenoid normally is open so that in the event of a power failure the fire-air is vented to prevent an inadvertent firing of the transient rod.

The transient rod fire circuit, shown in Figure A-24, incorporates an interlock to prevent firing the rod unless the latch is open. Firing occurs by opening two solenoid dump valves in parallel which rapidly vent the hold air. In case of malfunction, either of these valves is capable of dumping the hold air.

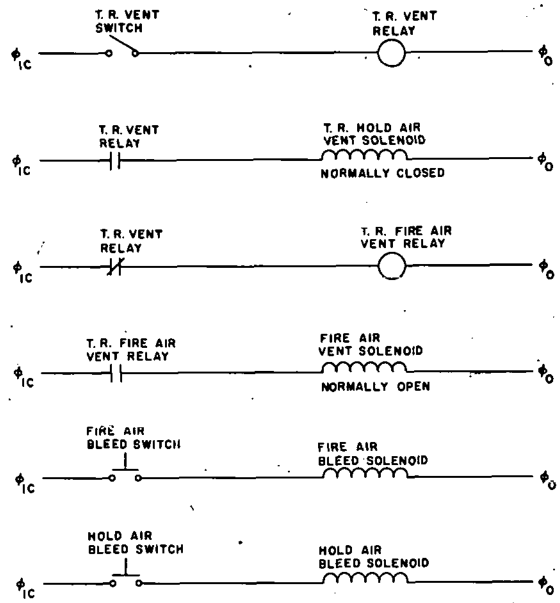

Fig. A-23 Transient rod vent and bleed circuit.

3.56 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 Figure A-25.

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,

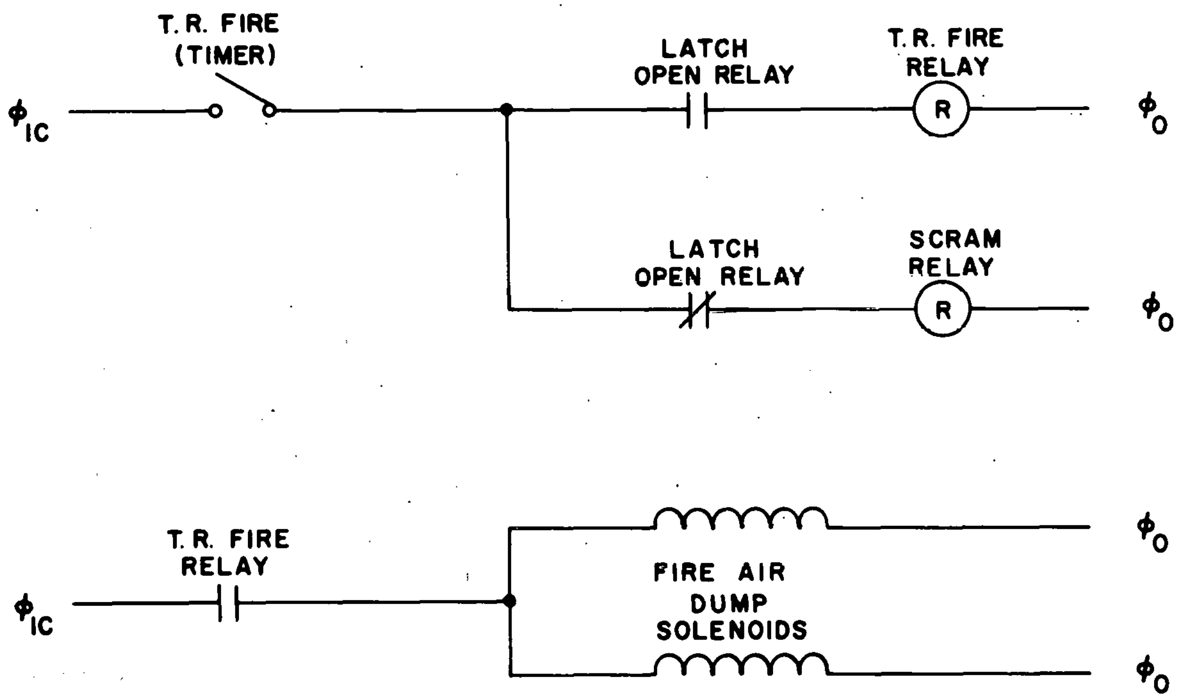

Fig. A-24 Transient rod fire circuit.

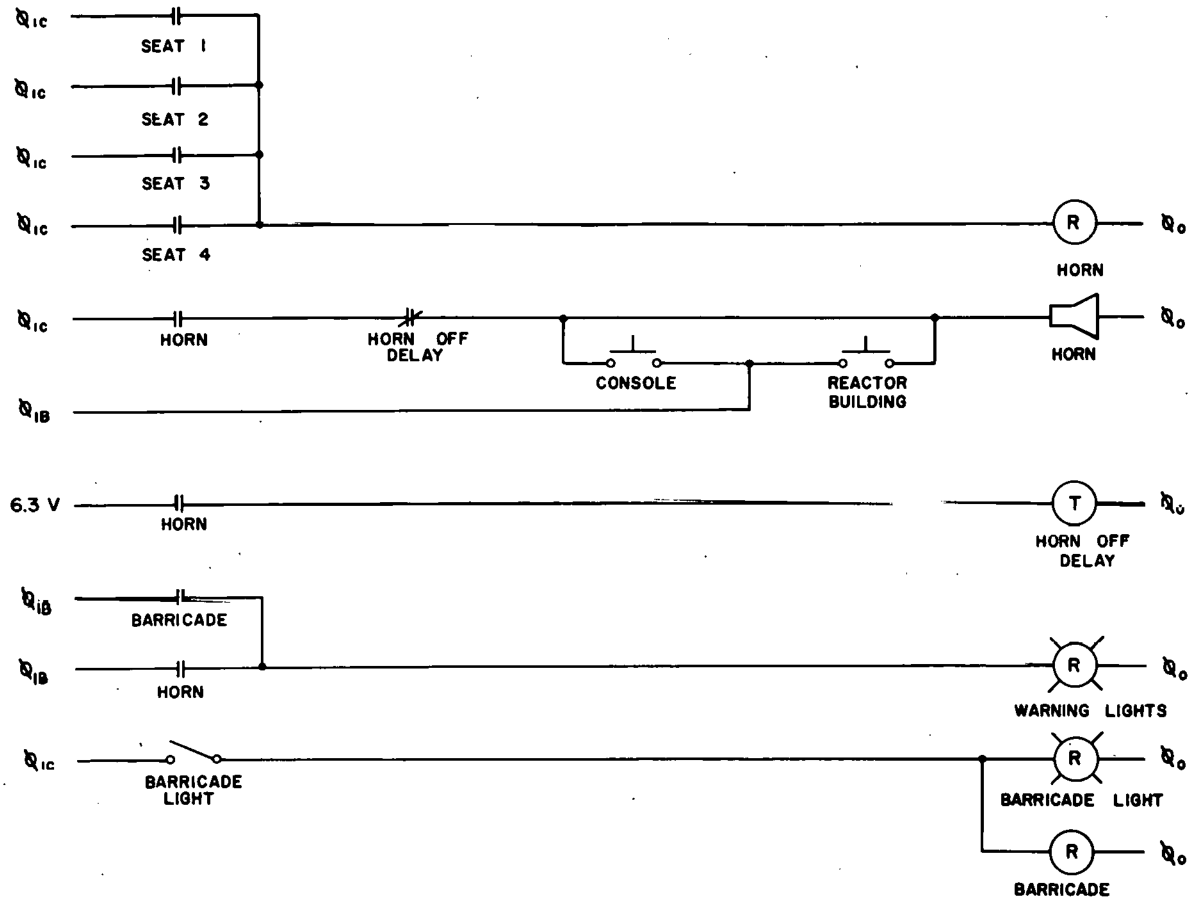

Fig. A-25 Warning horn and lights circuit.

one on the control console, the other in the reactor building. When actuated by the horn relay, the horn is operated for 15 sec 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 not energized by the manual horn buttons, and, as already mentioned, the horn operates from the horn relay for only 15 sec. 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.

3.57 Sequence Timer. An electronic timer using binary logic is used to program the time at which some or all of the following functions occur during a transient test: Cameras, tape recorder, recording oscillographs, latch unlock, transient rod fire, and control rod scram. There are two spare timing channels provided for programming miscellaneous functions. The timer is capable of counting from zero to 999.99 sec and the functions may be programmed to occur at any desired time within this range with an accuracy of ± 10 msec.

3.58 Rod-Drive Sensing-Switch Circuits. Circuits for magnet contact switches, rod seat switches and shock absorber extension lights are shown in Figure A-26. 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 subsequent dropping of the rods may be properly dissipated. Control console lights indicate the position of the shock absorber extensions.

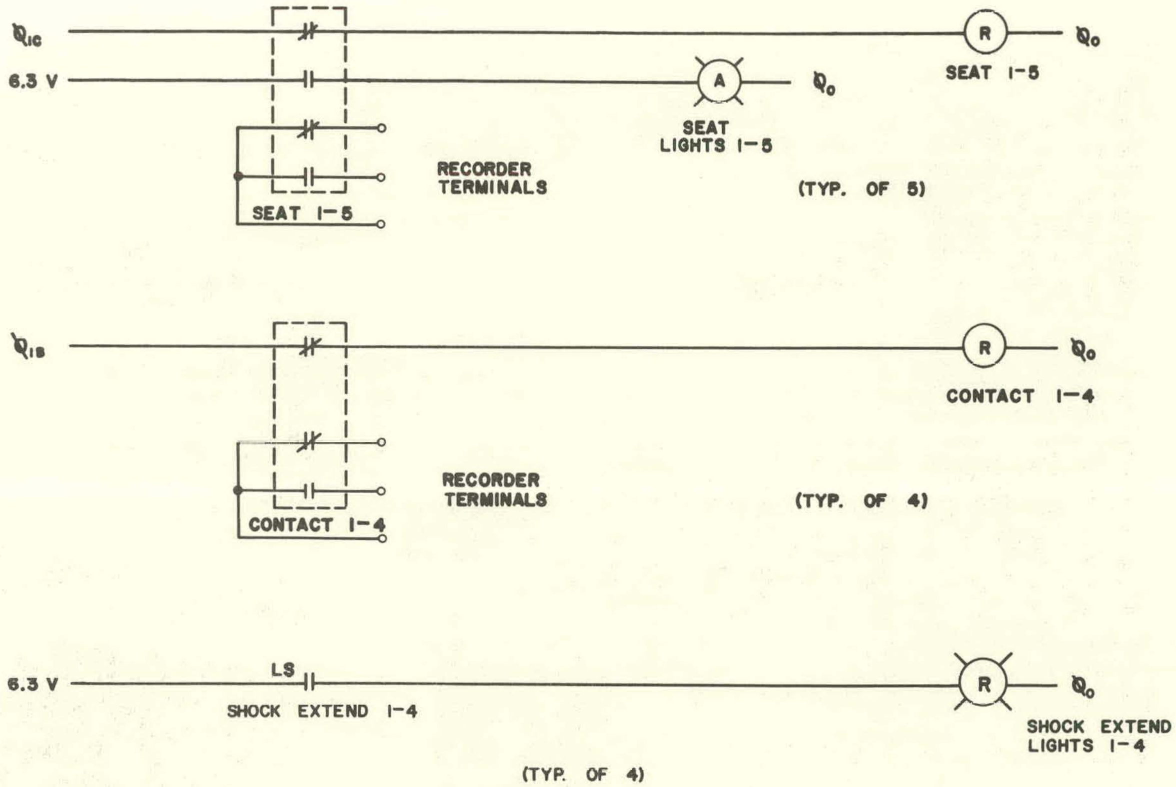

Fig. A-26 Rod-drive sensing-switch circuits.

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 is operated by relay, therefore, no continuity testing feature is provided. Rod drive limit switches (not shown) operate the control console indicating lights, no provision being necessary for the recording of limit switch signals.

3.6 Reactor Control Console

Figure A-27 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 (Figure A-28), a log-rate meter, and three 26-channel recording oscillographs (Figure A-29). The fourth panel, Figure A-30, 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.

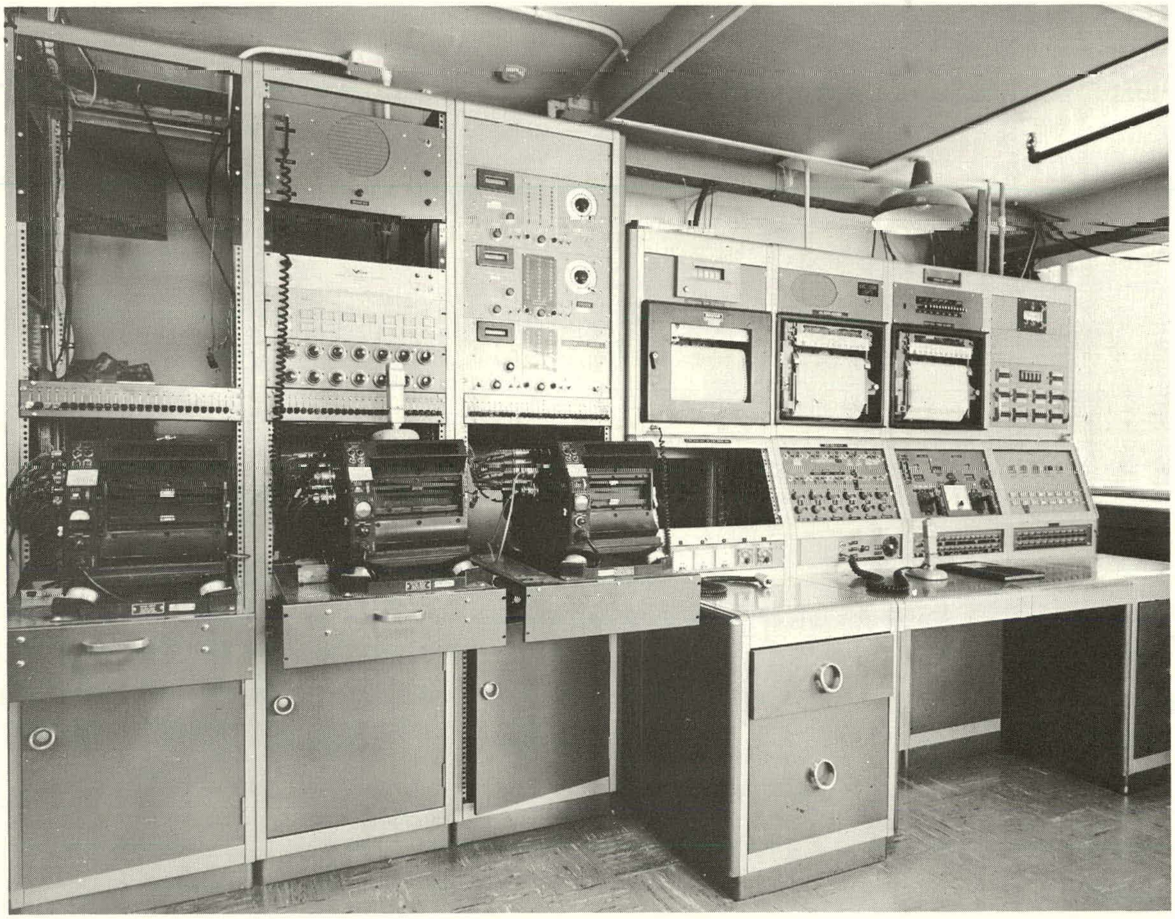

Fig. A-27 Reactor control console.

The fifth panel, in addition to the logarithmic power recorder (Figures A-30 and A-31), 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

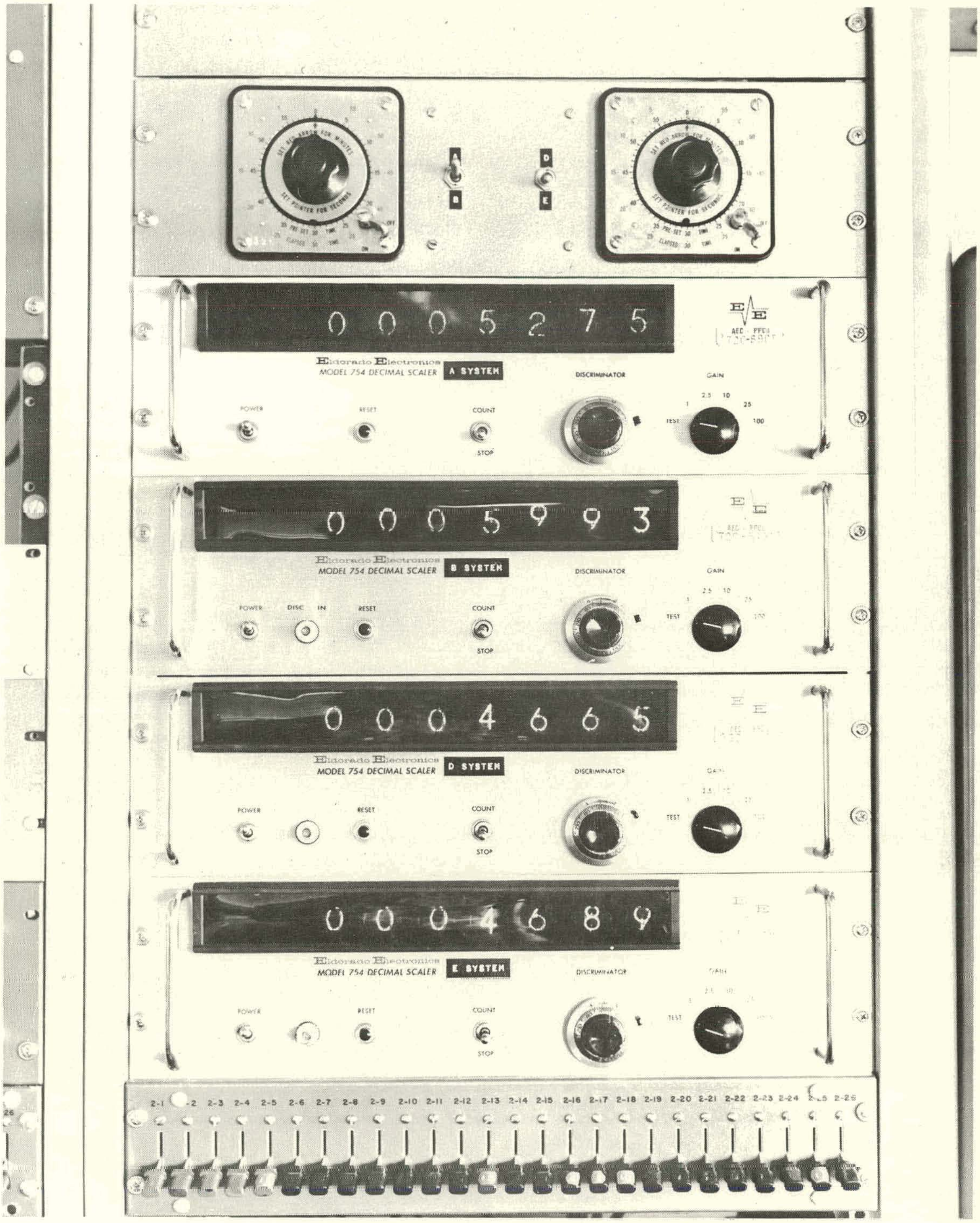

Fig. A-28 Neutron pulse counting system

reactor tank liquid-level indicator is located below the auxiliary equipment controls.

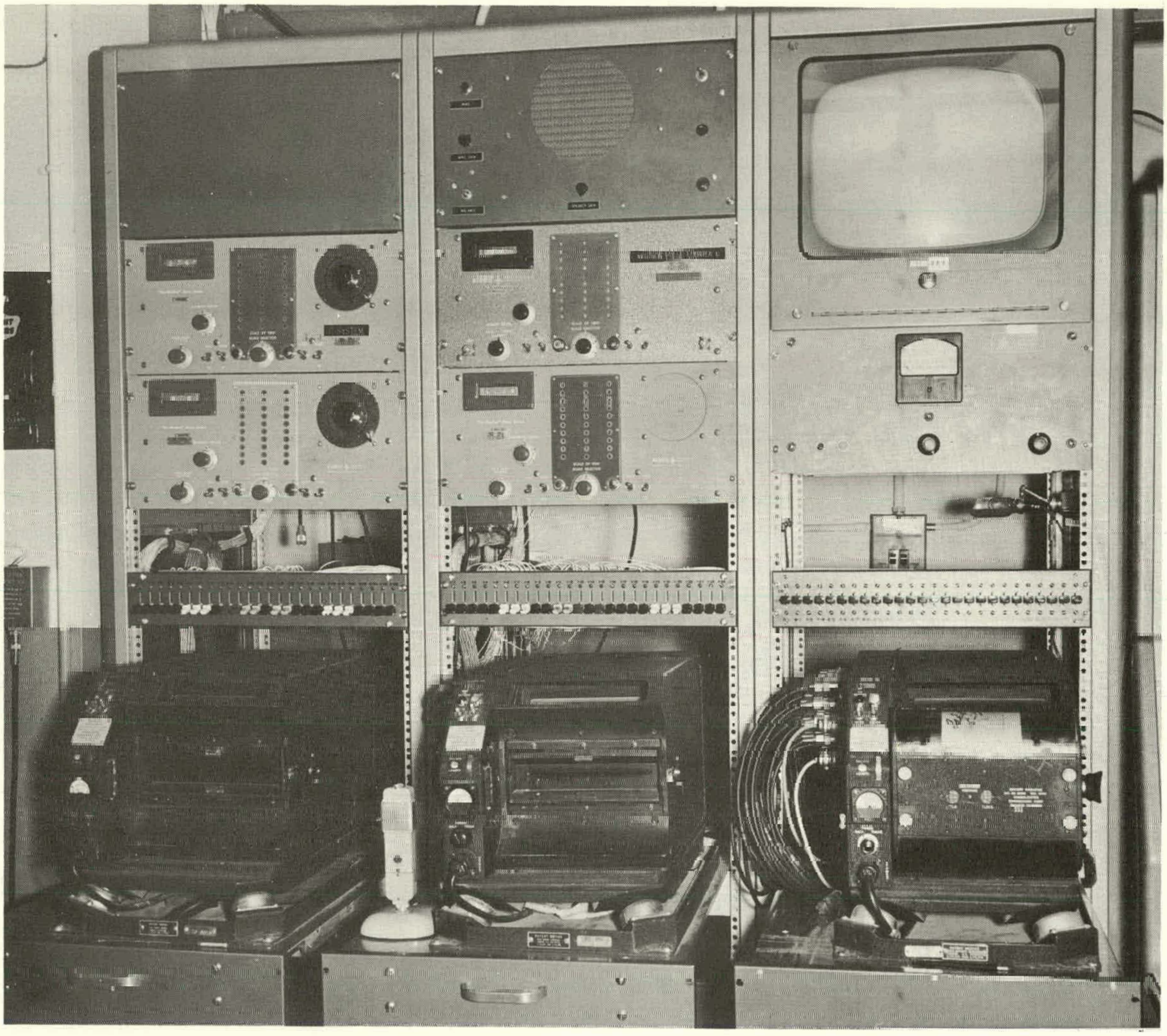

Fig. A-29 Recording oscillographs

Immediately above the auxiliary equipment controls are two rows of toggle switches (Figure A-31), which control 12 electrical outlets in the reactor building and the high voltage supply 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 (Figure A-32), contains the control switches, indicators, etc, associated with operation of the reactor, and the upper section (Figure A-33), contains a linear power recorder.

Two three-position pistol-grip switches, located on the left and right of center of the panel (Figure A-32) 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.

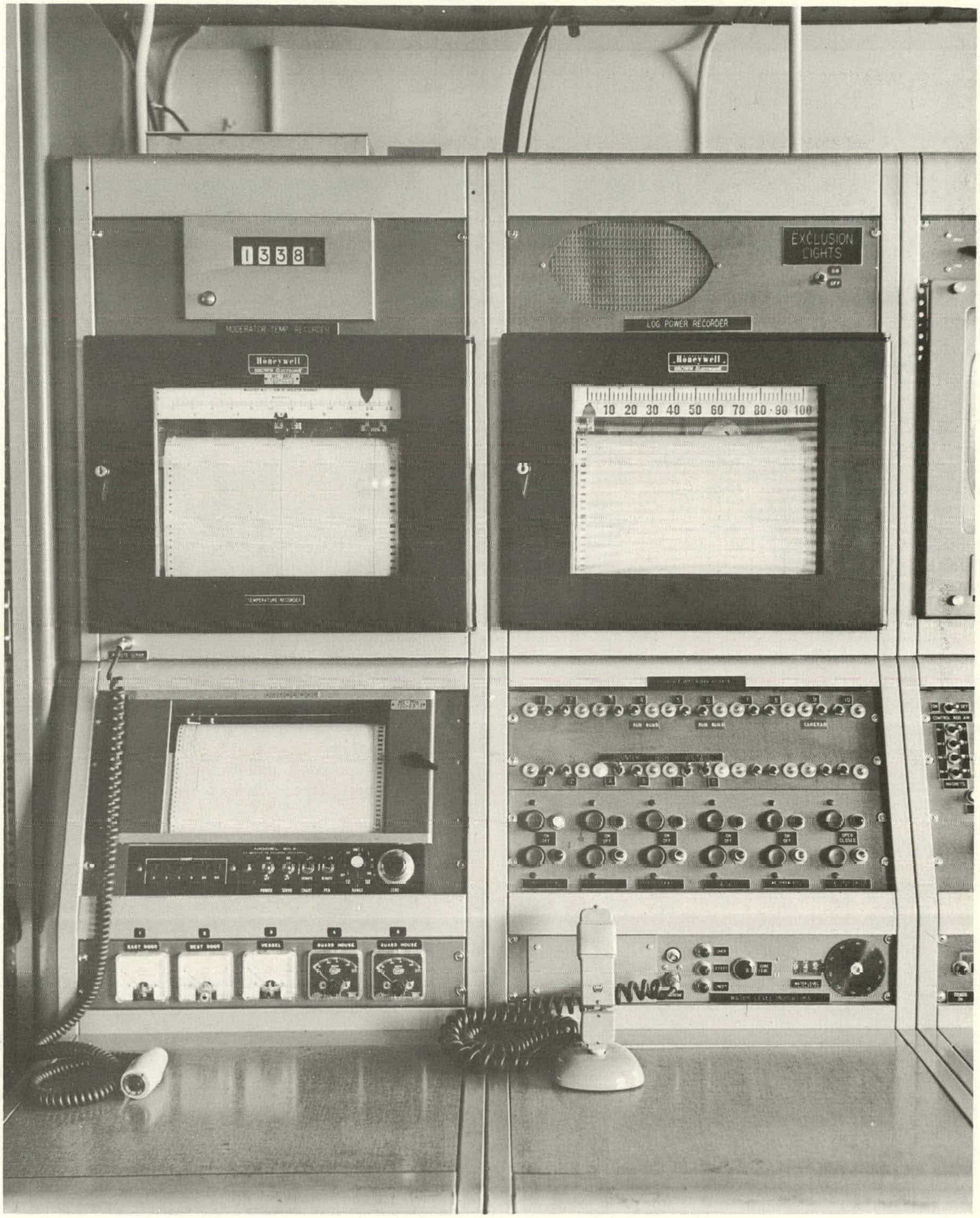

Fig. A-30 Reactor console panels 4 and 5.

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

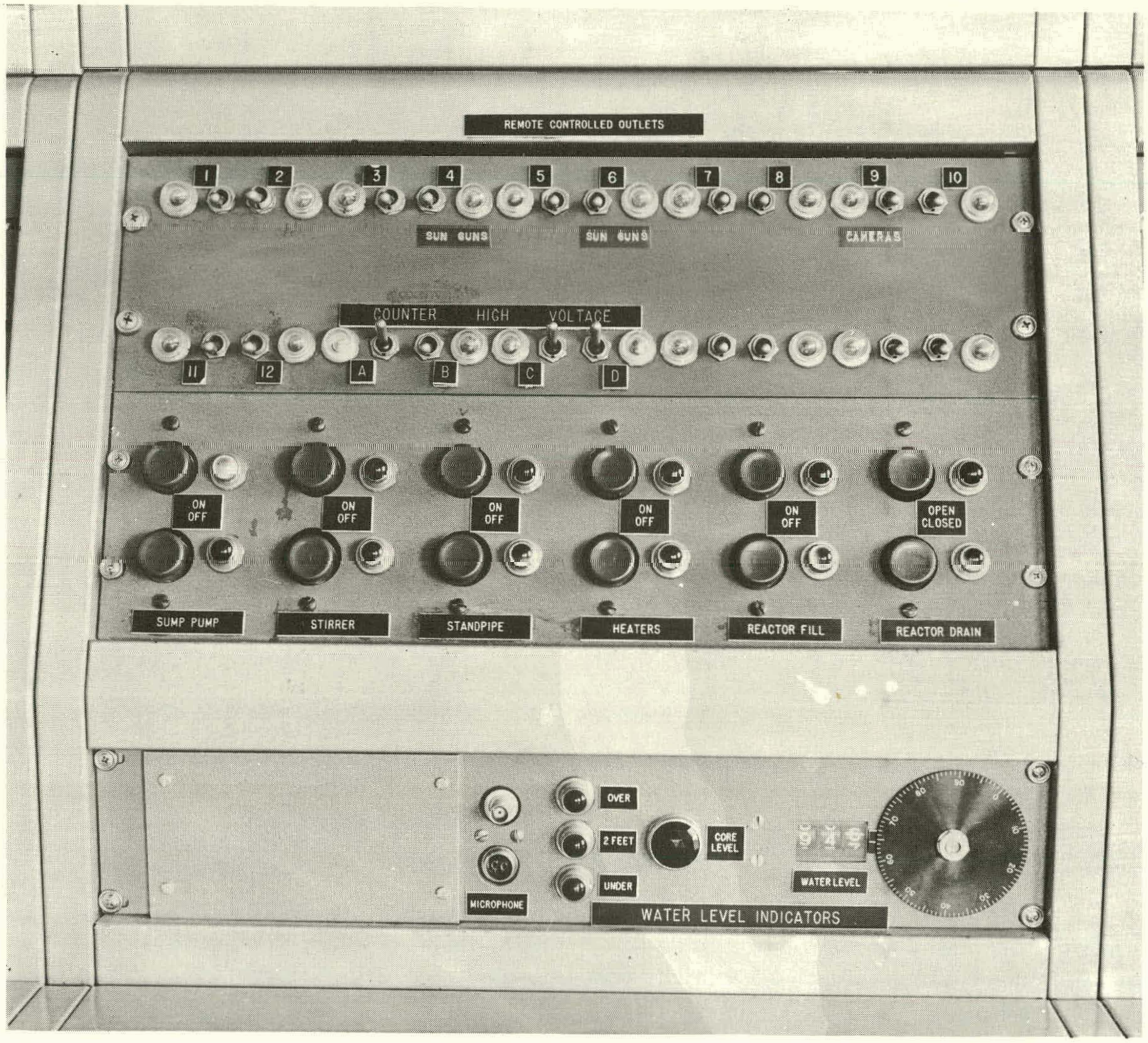

Fig. A-31 Remote outlet control (panel 5).

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.

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, the upper limit lights for the control rod drives, and the shock extend light for each of the control rods.

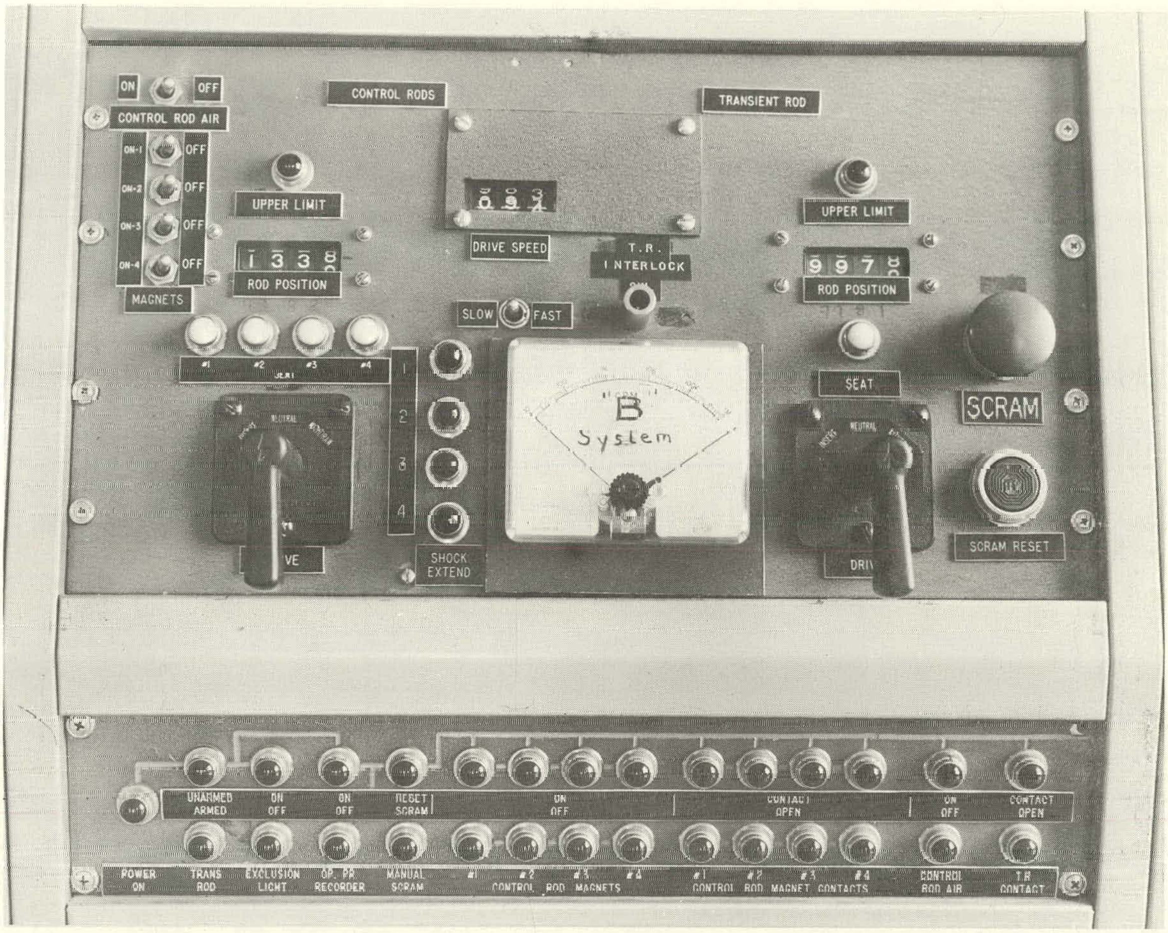

Fig. A-32 Lower section of reactor console panel 6; control and transient rod drive controls.

A small procedure panel (Figure A-32) equipped with red and green lights and located below the control panel supplies the following information to the operator:

(1) Power on

(2) Exclusion light - on or off

(3) Operational power recorder - on or off

(4) Scram - scram or reset

(5) Control rod magnets (1 - 4) - on or off

(6) Control rod contact (1 - 4) - contact or open

(7) Control rod air - on or off

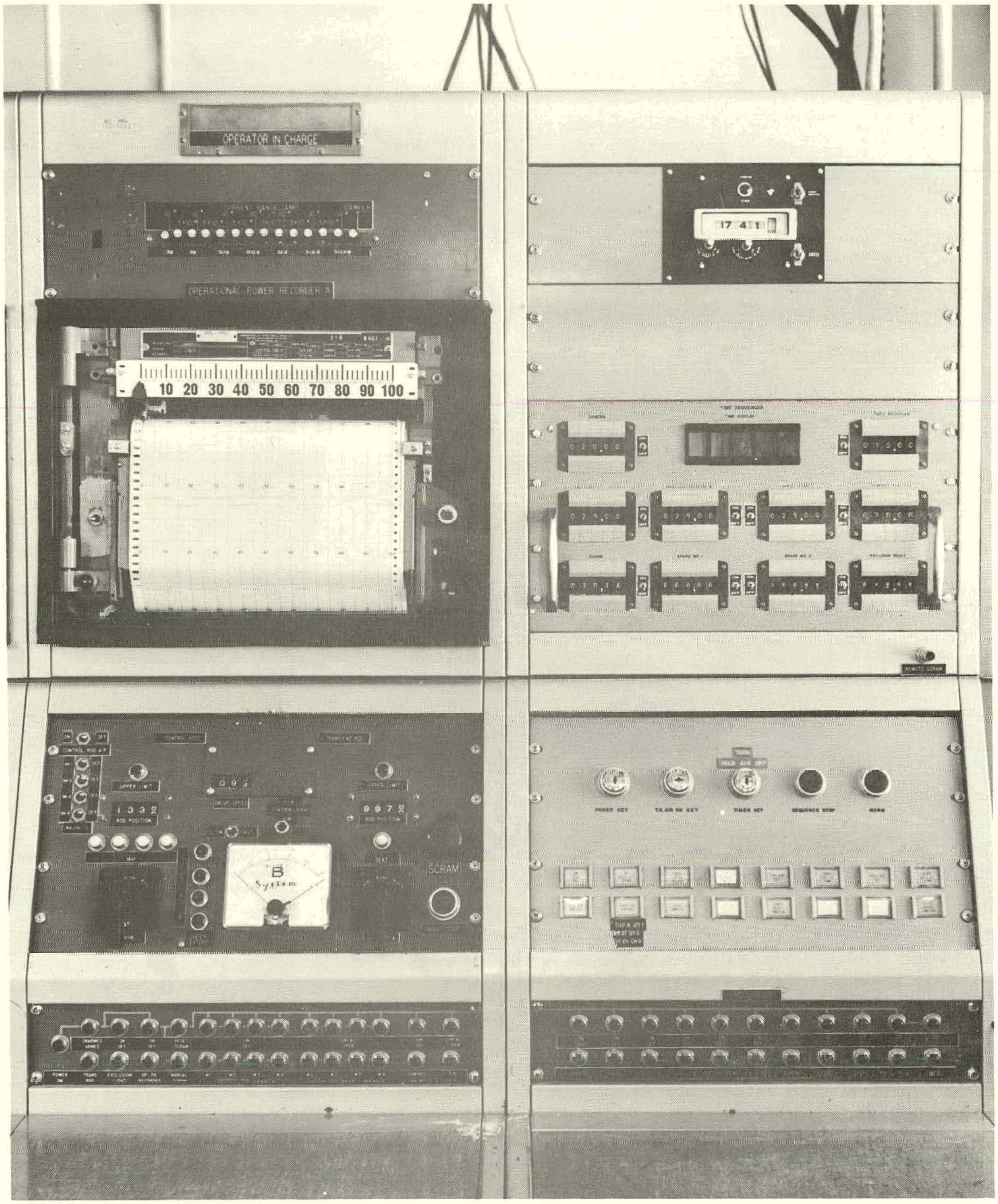

Fig. A-33 Reactor console panels 6 and 7.

The upper part of panel six (Figure A-33) contains the linear power recorder and its range selector.

The last panel contains the sequence timer and controls for the transient rod air system. The timer (Figure A-34), in accordance with a preset schedule,

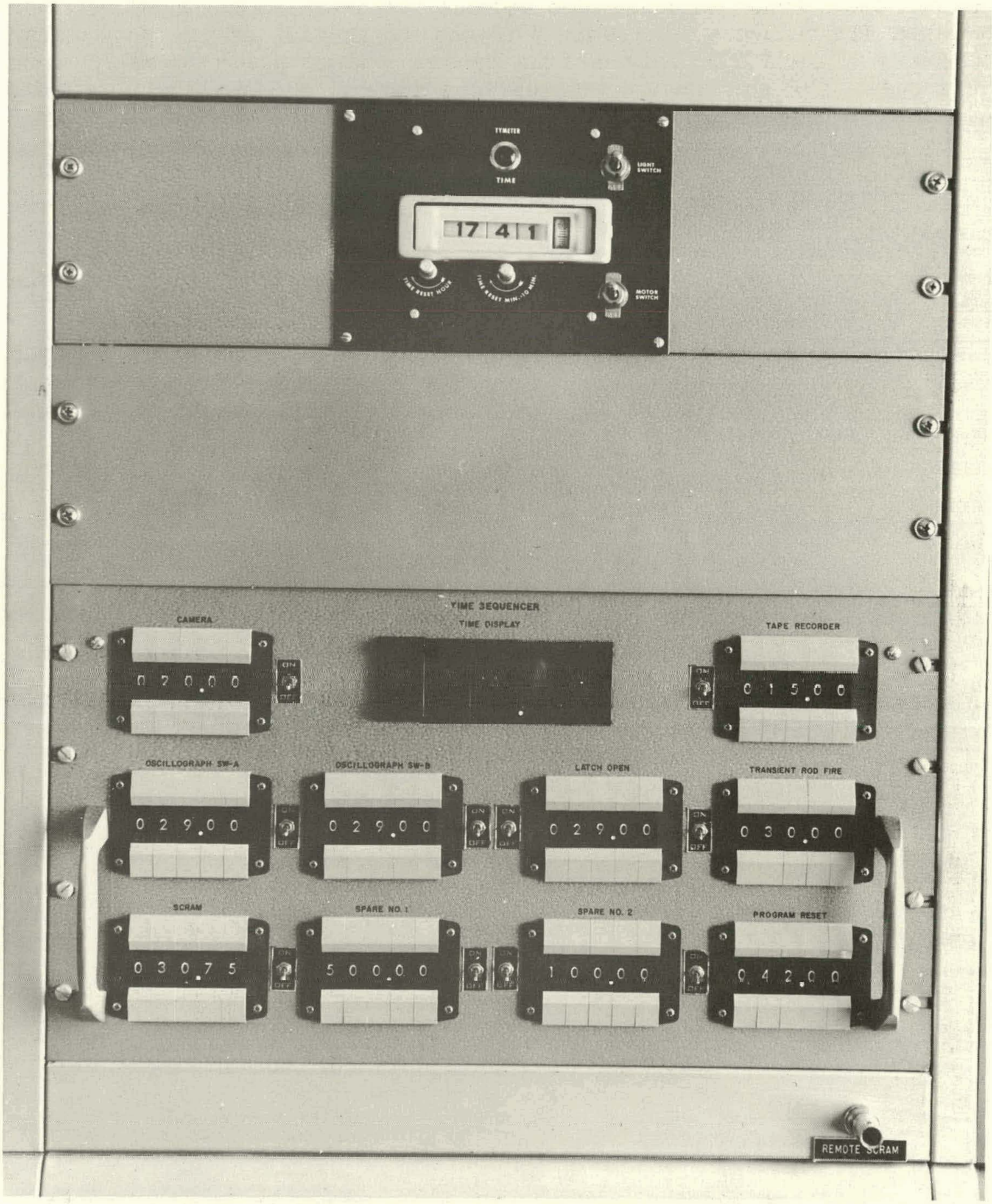

Fig. A-34 Sequence timer (panel 7).

starts cameras and recording equipment, initiates the firing of the transient rod, and at the desired time inserts the control rods and shuts off experimental equipment. In addition, this panel contains key operated switches for main power, transient rod air supply, and timer start. There are also two push buttons, one a timer sequence stop and one is the warning horn in the reactor building. The sequence stop button will stop the 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, 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.

The transient rod air panel, Figure A-35, contains the controls necessary for the operation of the transient rod. The top row contains indicator lights showing the configuration of the transient rod, such as seat, shock absorber extended, latch unlock or lock, fire-air set, etc. The bottom row contains push button switches for such operations as applied fire air and hold air, venting, etc. These switches are self-indicating variety.

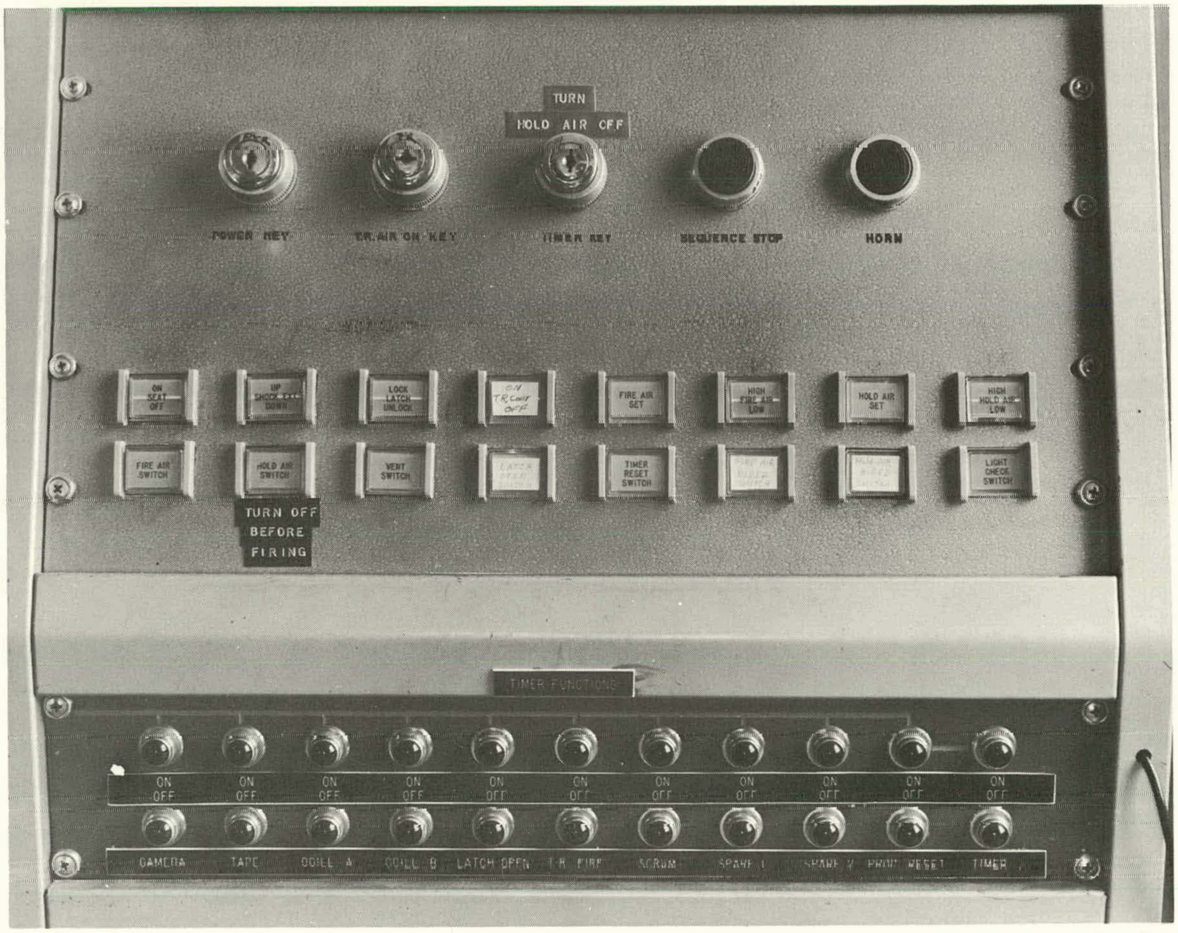

Fig. A-35 Transient rod air control (panel 7).

The bottom section of this panel contains on-off indicator lights for the timer functions.

Located near the reactor console is a small auxiliary console for the use of the control room supervisor. This console, Figure A-36, contains a log count rate meter, a scram button, a timer sequence stop button, and two water level indicator lights showing when the water level is below 6 in. and 3 in., respectively. In addition, the master control switch for a soluble poison injection system is located here. The soluble poison, gadolinium nitrate, is a "backup" shutdown system used in the event that it is not possible to insert the control rods due to damage.

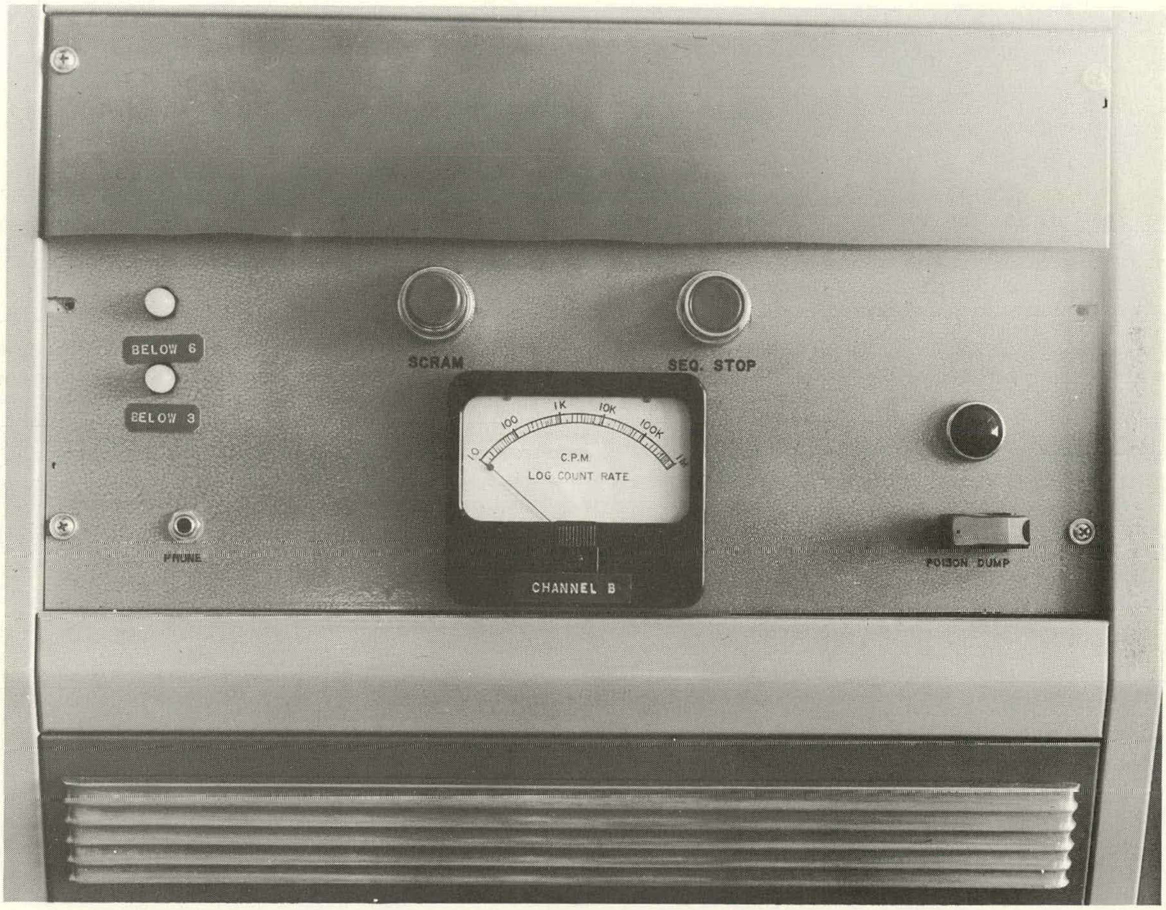

Fig. A-36 Auxiliary control console.

4. OPERATIONAL INSTRUMENTATION

4.1 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 are given in Section 6 of this Appendix.

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 Figure A-37.

4.2 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 while counting rates are being recorded on the other pulse systems. A log count-rate meter is included in this "backup" system. An independent neutron-monitoring system, which is audible in both the instrument bunker and reactor building, is connected to the most sensitive pulse system. This channel is battery powered and will continue to operate during a power outage. During any nuclear operation, at least two pulse systems must be in operation.

In the power range from one watt 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.

4.21 Pulse Channel Electronics. Figure A-38 is a block diagram of a typical pulse channel. This system uses a Spert built integral solid state start-up channel with all major components except the chamber high voltage power supply housed in one instrument.

(1) Preamplifier Circuit. The preamplifier is a two-transistor current-sensitive pulse amplifier whose input impedance terminates the RG-8 coaxial line from the chamber in its characteristic impedance. There are no reflections or losses in the 3000 ft of cable between the chamber and the preamplifier. Extra electromagnetic shielding pipe is required around the chamber cable to minimize noise pickup.

(2) Amplifier Circuit. The amplifier used in this system is a low-level pulse amplifier provided by Eldorado Electronics as part of the scaler. This stage has an adjustable gain with a maximum of 100. With a gain of 25, the average neutron pulse output is about 5 volts peak with a duration of 0.4 μsec. The overload characteristic of the amplifier are such that saturation neutron pulses have little effect on the neutron count rate after recovery.

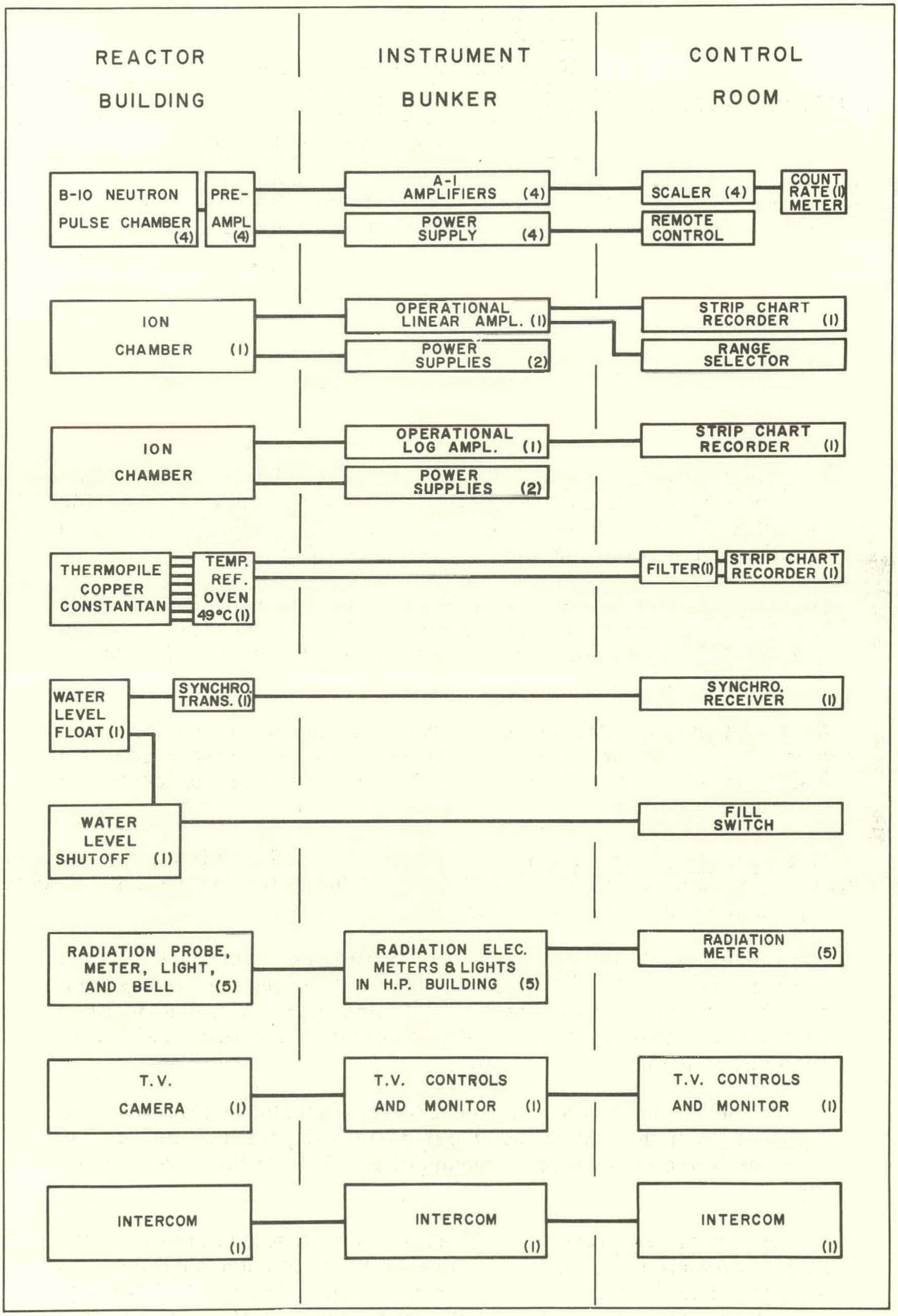

Fig. A-37 Block diagram of operational instrumentation.

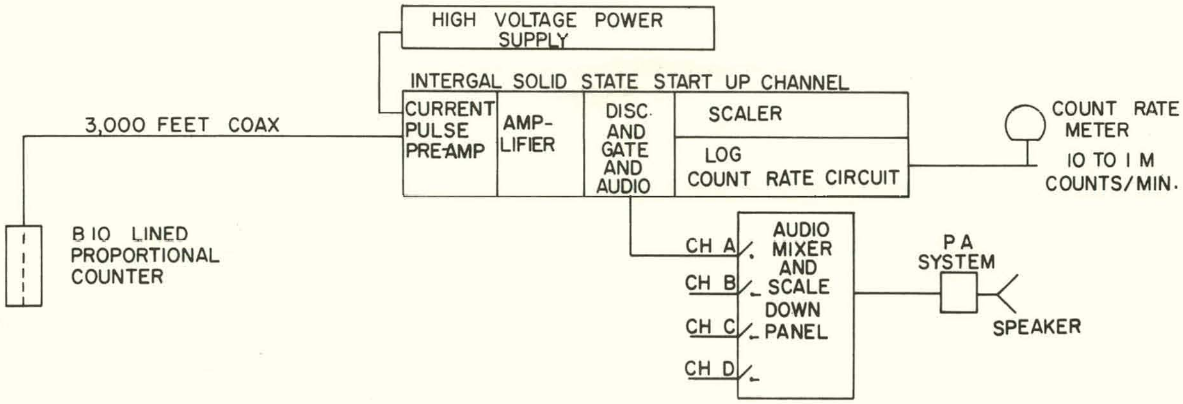

Fig. A-38 Block diagram of typical neutron pulse channel.

(3) Discriminator Circuit. The discriminator circuit, also supplied by Eldorado Electronics as part of the scaler, contains the gating circuitry for the scaler. The minimum pulse height is 1 volt, and it will resolve pulses as close together as 0.5 μsec.

(4) Scaler. The scaler, manufactured by Eldorado Electronics, has seven decades of nixie readout. Neutron counting rates as high as 10^7 counts per minute can be obtained before 10% counting loss can be expected.

(5) Log Circuit. The log-count-rate circuit is a diode log-pump type circuit manufactured by Tracer Lab. covering a counting range of 10 to 1 million cpm. The readout meter is located on the control console.

4.22 Ionization Chamber System. Figure A-39 illustrates a block diagram of a typical ion chamber power-level channel, including the ion chamber detector, power supplies, the electrometer ammeter, and recorders.

(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 Figure A-40. 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 are positioned by means of brackets on the core support legs.

(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^{-2} amp, inclusive, with the desired range selected by a switch on the control console (Figure A-33). The log electrometer was manufactured by Keithley Instruments Co. The device

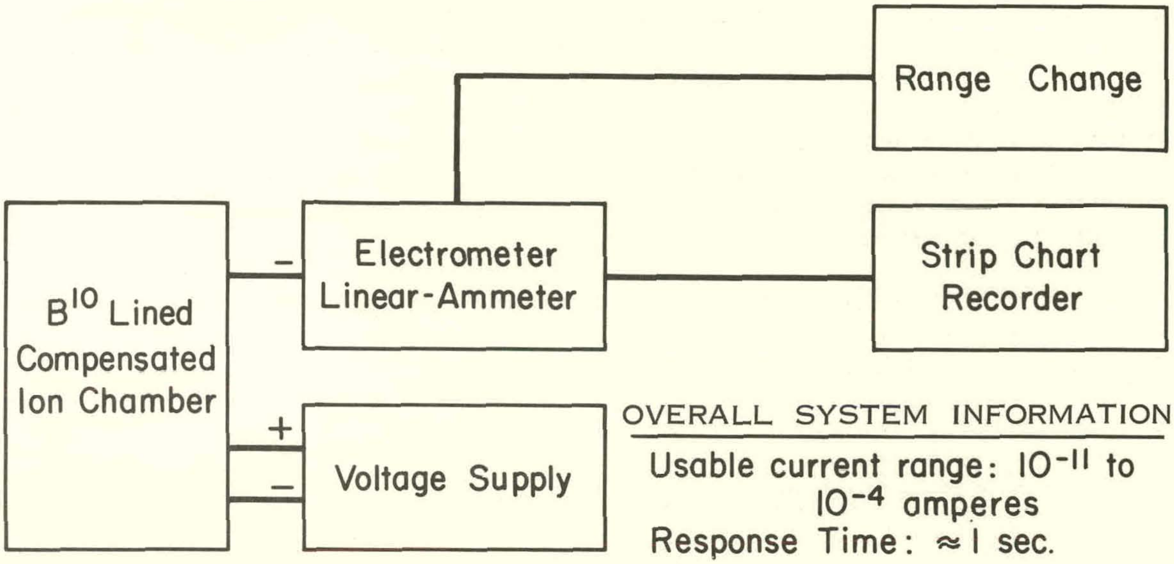

Fig. A-39 Block diagram of typical ion chamber linear power-level channel.

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^{-4} 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 to 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 the Evenvolt power supply to provide a constant bucking voltage.

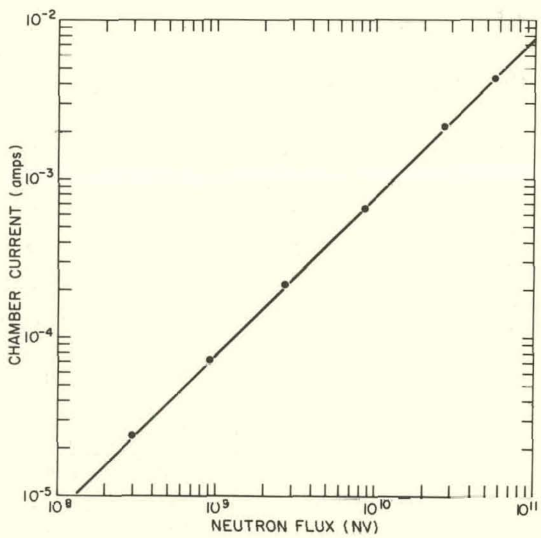

Fig. A-40 Neutron sensitivity curve.

4.3 Bulk-Water-Temperature Indicator

The bulk-water temperature in the Spert I reactor is measured by a thermopile consisting of 12 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 49.1°C. The signal from the thermopile is filtered to eliminate 60-cycle ac noise and transmitted without amplification to the control center. The temperature is read in millivolts on a multirange Minneapolis-Honeywell Brown recorder equipped with five ranges and an automatic range selection. The range span is 0 to 10.2 mv.

4.4 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 Tracer Lab, 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 Corp. constant air monitor (CAM) 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.

4.5 Water Level Indicators

(1) Continuous Level Detector. As a means of monitoring the water level before and after, but not during, a destructive type test in the Spert I reactor tank, a strain gage pressure transducer and purge system will be used. This system consists of a nitrogen gas supply of at least 50 psi throughout the monitoring period, a bubbler to measure the gas flow, a length of 1/4-in. stainless steel tubing immersed to the bottom of the water filled tank, a pressure transducer and a mv recorder.

This system uses the effects of back pressure in a tube which is devoid of the measured media. A pressure transducer placed between the bubbler, and the submerged end of the probe tube, will transmit a linear mv signal as a function of water level change. This signal will be recorded at the control center on a strip chart recorder. The diameter and length of the pressure probe protects the relatively low-range transducer from damage during a destructive pressure pulse.

(2) Unambiguous Level Detector. Two level devices which have an off-on type signal will be placed at the bottom of the reactor tank, one at 3 in. and the other at 6 in. Each of these devices consists of polystryrene float which actuates a miniature microswitch at 3- or 6-in. water levels. The microswitches and floats are housed within heavy wall stainless steel chambers which have small seep holes for water level sensing. The signal wires are protected with stainless steel tubing and exit clear of the reactor tank.

5. AUXILIARY EQUIPMENT

5.1 Water Treatment System

5.11 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-II 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 Figure A-41 consists of a water softener, one demineralizer unit, and associated piping and controls.

TABLE A-II

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

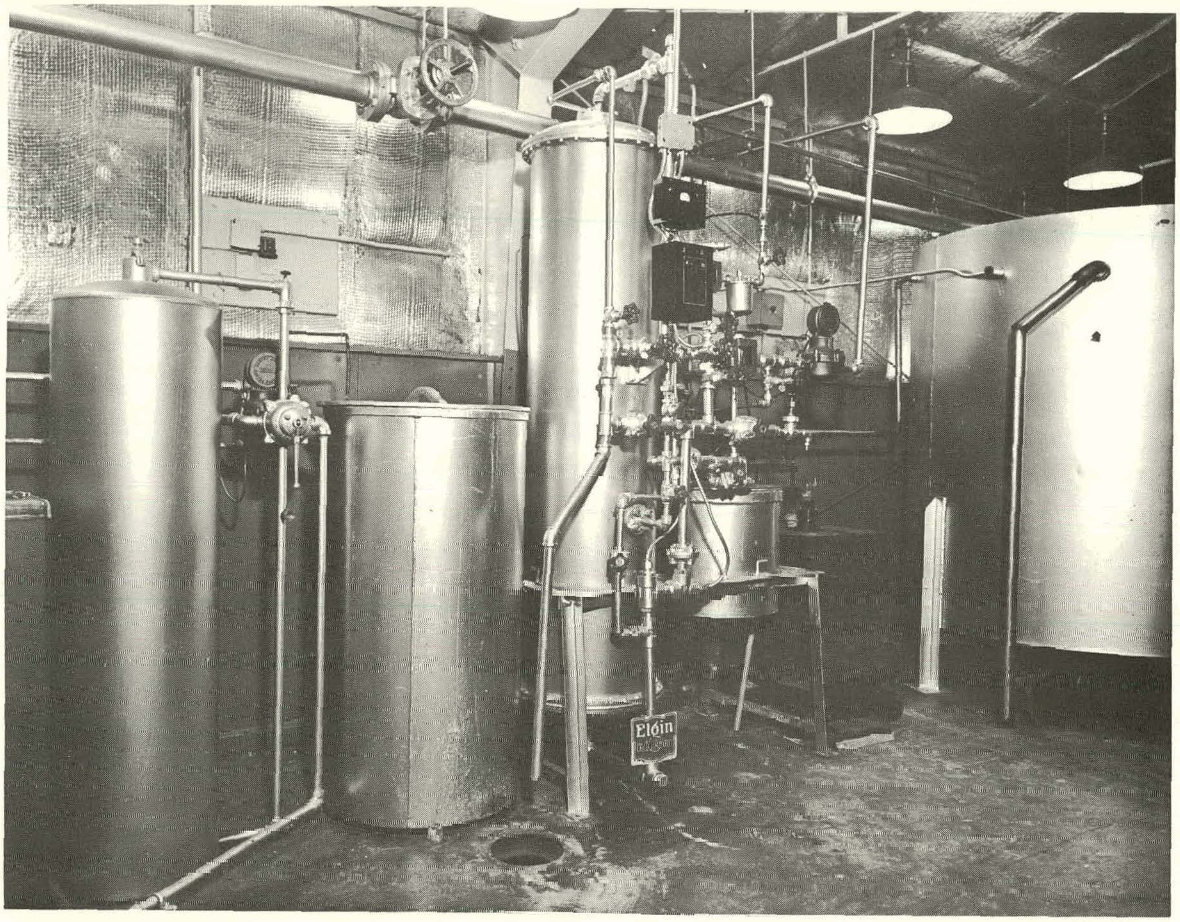

Fig. A-41 Water treatment system.

5.12 Softener. An Elgin Zeolite water softener, using a carbonaceous zeolite 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.

5.13 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. Breakthrough 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.

5.14 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.

5.2 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 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.

5.3 Reactor Equipment

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

5.31 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 Co. and may be operated on automatic control or on manual control from either the reactor building or the Spert I control room. Approximately 60 min are required to drain a 12-ft head 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.

5.32 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 two ft from the tank wall. The stirrer is used to obtain a uniform distribution of the bulk-water moderator.

5.33 Television Cameras. Three remotely controlled, closed-circuit television cameras will be utilized to provide visual observation of the reactor vessel and adjacent areas during and after the destructive test. One television camera equipped with zoom and pan capabilities is to be mounted on the roof of the guard shack directly southwest of the reactor building. This camera will provide visual observation of the area inside the reactor building above the reactor vessel. A second camera will be mounted above a 50-ft mast located near the southwest corner of the reactor building. This camera will provide additional coverage of the interior of the reactor building with the added feature of being able to see into the reactor vessel to a limited degree. This camera also will be equipped with zoom and pan capabilities. The third television camera will be mounted on the end of an arm positioned along the northeast wall of the reactor building during the excursion and may, through remote control, be moved to the edge of the reactor vessel after the test is completed. This camera will provide observation of the reactor vessel after the excursion.

5.4 Emergency Shutdown System

It is conceivable that during a destructive test an electrical or mechanical malfunction of the control rods or physical damage to the core could prevent insertion of the control rods to shut down the reactor. Therefore, it is necessary to provide some means that are 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, 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 shutdown 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. Figure A-42 is a schematic diagram of the emergency shutdown system.

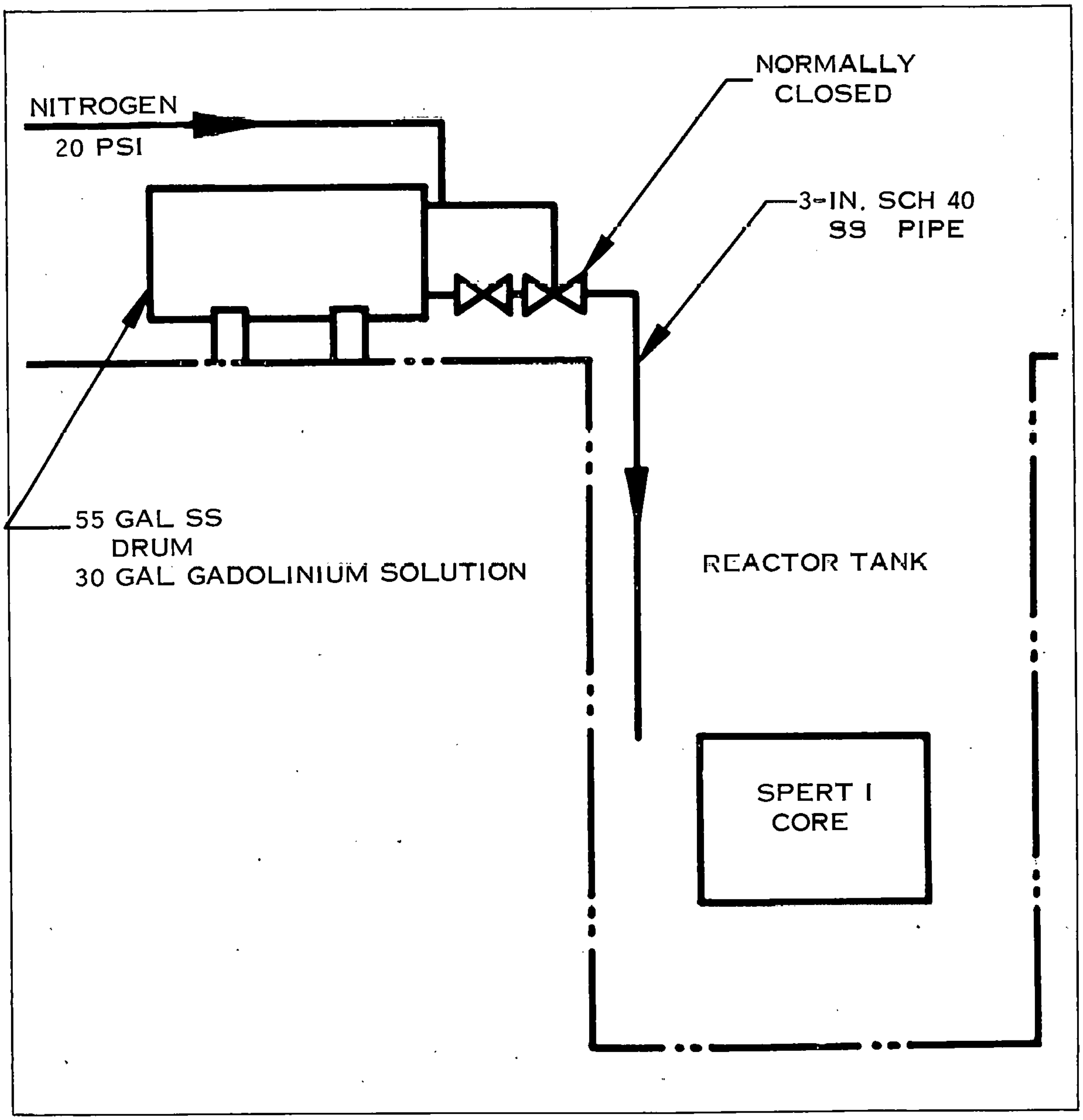

Fig. A-42 Poison injection system.

6. TRANSIENT MEASUREMENTS AND INSTRUMENTATION

6.1 General

The necessity for recording the data at the control center 1/2 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 reentry to 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, transmitted by means of special driver amplifiers, 3000 ft to high-speed oscillograph recorders and magnetic tape recorders in the control center.

The instrumentation system requirements necessary to obtain good data on reactor power, fuel plate temperatures, pressures, 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 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 in $100/2\pi$ times the reciprocal period of the exponential rise [1].

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:

(1) 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 limitation and replacement of these instruments will be made whenever necessary. Protection of transducer leads and ion chamber leads is provided.

(2) 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 also are recorded over two different ranges, differing by a factor of ten, to provide complete data monitoring during the early portion of the transient.

(3) 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 acoustic measurements.

6.2 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 level over a dynamic range of about five decades will be made with two different classes of detectors: (1) miniature B-10-lined ion chambers and/or semi-rad chambers which have linear output vs flux to at least 10^{16} nv, and (2) conventional B-10-lined ion chambers which are linear to only approximately 10^{11} nv. The detectors which are linear to 10^{16} nv will be located within at least 6 in. of the core, and their output will be primarily indicative of the neutron level, whereas the other detectors must be located 5 ft or farther from the core. Consequently, their output will be primarily indicative of the gamma power level. Placement of these chambers is shown in Figure A-43. The linear power recording circuits for the miniature ion chambers and the semi-rad chamber cover a power range of 4 decades, compared to about 2.6 decades for the conventional chambers. A 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. Both of these signals will be taken from conventional ion chambers. The relative dynamic range that is nominally covered by the ion chambers during a transient test is shown schematically in Figure A-44. 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 the absolute power level is made on the basis of a calorimetric measurement similar to that described in Section 6.3 below.

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 after peak power,

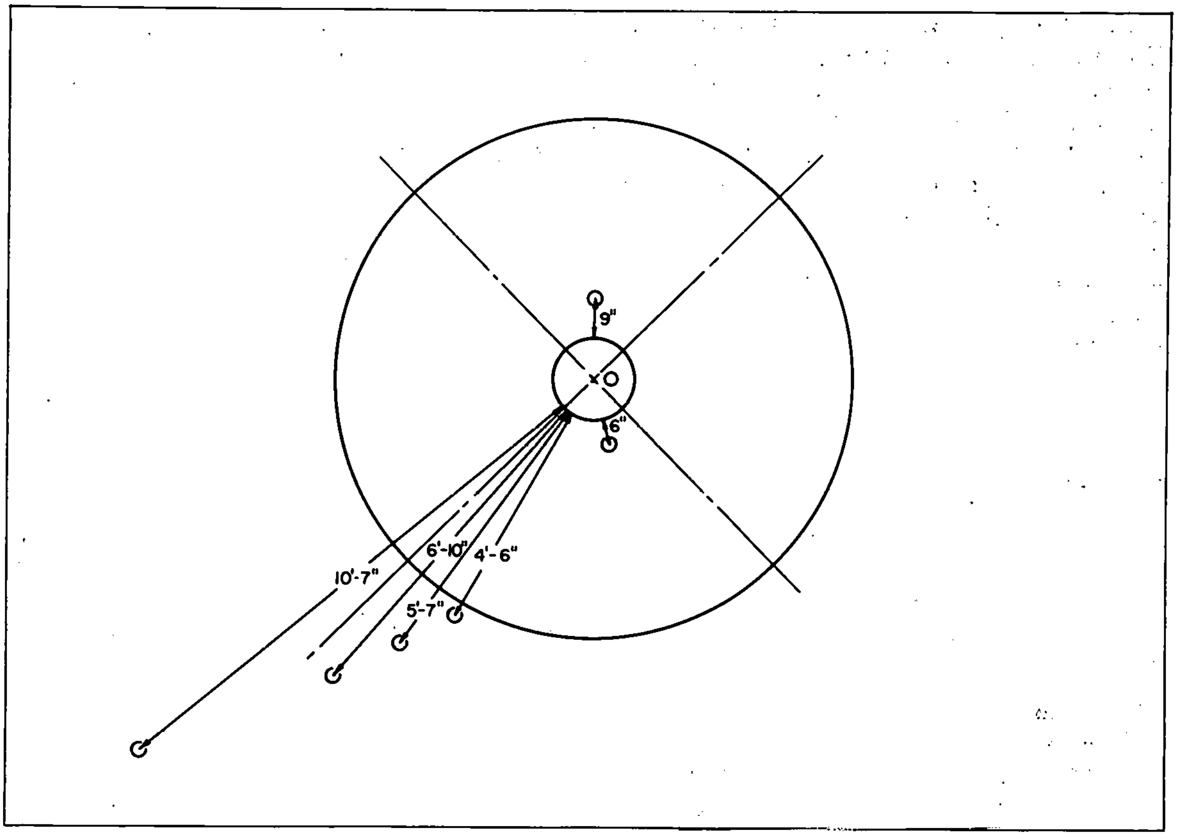

Fig. A-43 Spert I oxide core transient neutron power chamber location.

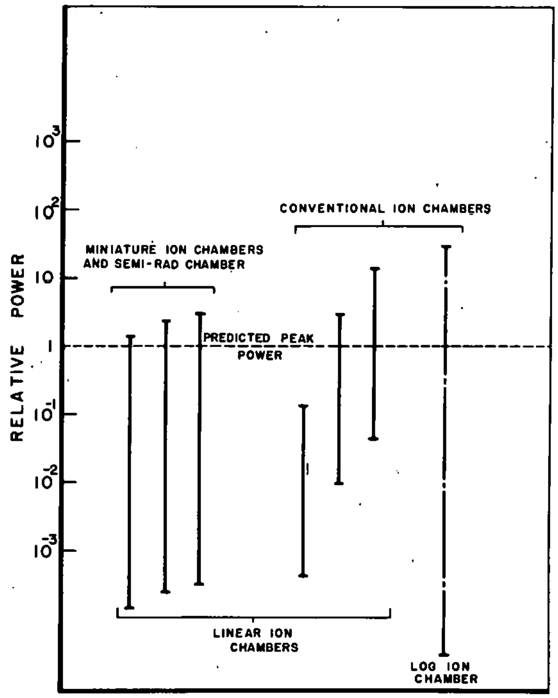

Fig. A-44 Relative dynamic range coverage of transient ion chambers.

when 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 will be placed inside the core. Amplification of the conventional ion 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.

Amplification of the miniature ion-chamber and/or semi-rad chamber signals are accomplished by dc to 20 kc low-level amplifiers. By use of the appropriate range staggering techniques, full scale output levels with attenuation of 1, 3, 10, 100, 300, and 1000 are provided. This provides for up to six channels of data from each chamber and allows the same chamber to be used both for period and peak power measurements. Power measurement 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.

6.3 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 100 Mw-sec energy burst, the calculated uniform bulk water temperature rise is approximately 1.0°C, a temperature change which can be measured to about 1%. 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, provided that little or no water is lost from the reactor tank.

6.4 Water Pressure Measurements

The transient pressure that can be developed during a short period power excursion from rapid steam formation, or other sources constitutes a principal destructive mechanism in reactor accidents, and its measurement during the destructive tests is of importance. The measurement and analysis of transient pressure, however, 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 oxide core destructive test series. Approximately 15 pressure transducers will be used during the destructive tests to obtain data pertinent to (1) magnitude, (2) rise time, (3) duration, (4) propagation and (5) directionality of the pressure pulse. Detailed experimental information on these questions require the use of sensitive or specialized pressure transducers. Additional, less sensitive devices will be used primarily for "backup" (protective) purposes.

Transient pressure measurement of low magnitude and slow rise time pressure pulses 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.

Three different techniques will be used to determine the transient pressure in the high pressure, fast time response region. A bar gage constructed of a strain gage mounted on a 1-in. stainless steel rod will be used to measure the rise time of the pressure pulse and the magnitude if it occurs within approximately 0.4 msec of the initiation of the pressure pulse. The strain gage is used to measure the compression of the rod caused by the pressure striking the end of the rod. Quartz-crystal pressure transducers capable of measuring peak pressures up to 70,000 psig and rise time of approximately 1 msec also will be used. These quartz transducers are being evaluated in a transient radiation environment and these results will dictate the location of these transducers with respect to the core of these transducers. The third approach taken will be the use of a mechanical filter on a strain-gage pressure transducer to attenuate the high-frequency components of the pressure pulse. This type of transducer then will be capable of determining the duration of a pressure pulse by eliminating any resonances of the transducers which may be excited by the high-frequency components of a fast-rise-time pulse.

For short-period transients, "backup" transducers, ie, ball-crusher and/or diaphragm deformation gages, 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.

Mounting brackets are used to position the transducers in the three dimensional array. Approximate transducer positions are given in Figure A-45.

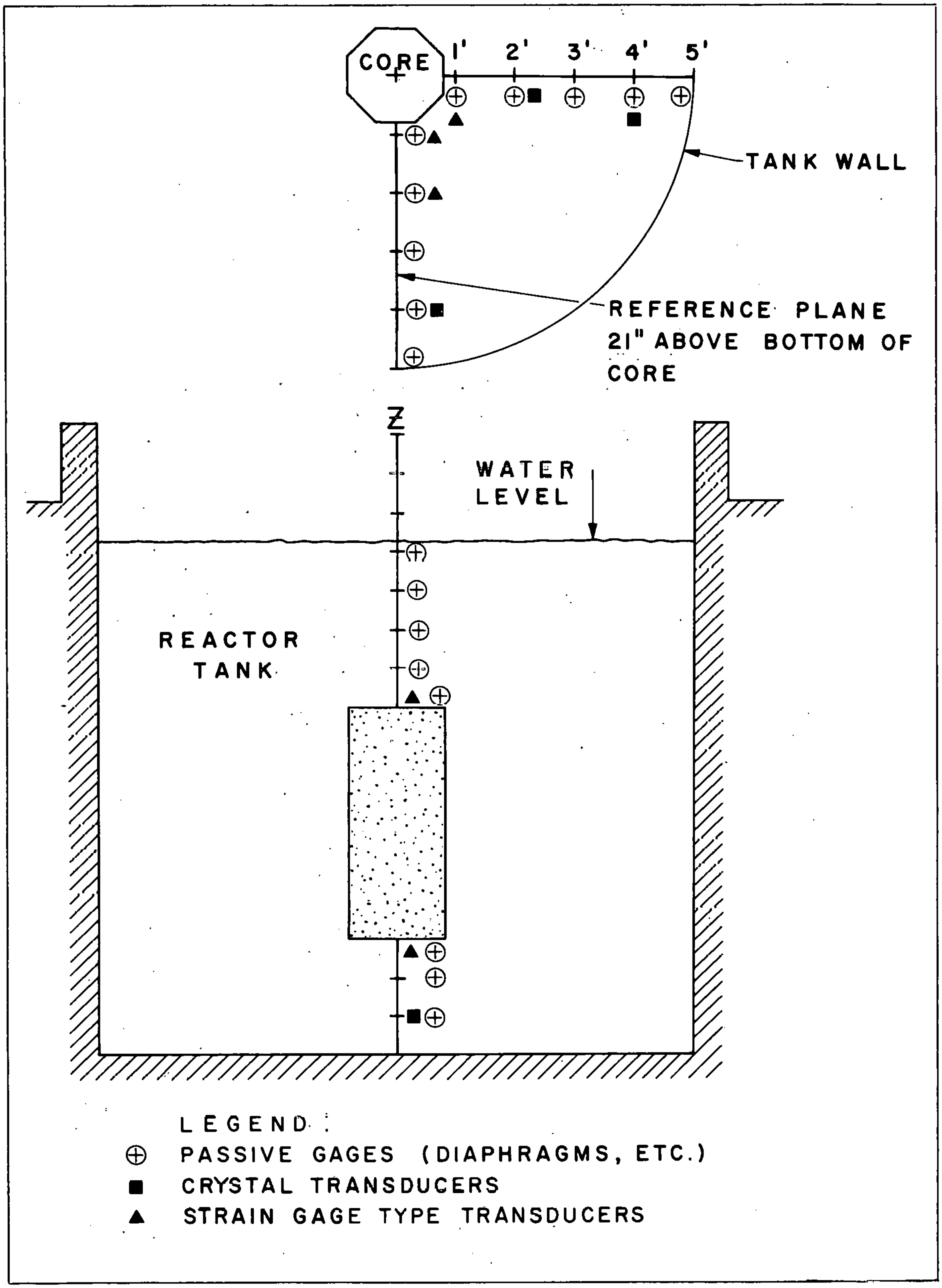

Fig. A-45 Transient water pressure transducer locations.

6.5 Fuel Rod Surface Temperature Measurements

Fuel rod 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 transfer rate to the water, thermal stress, etc, as functions of time during the transient. The onset of boiling at a particular point in the core also is indicated by the shape of the temperature curve [2, 3]. Temperature data also are 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 0.010-in.-diam, chromel and alumel wires attached to the stainless steel rod surface. A photograph of a typical junction is shown in Figure A-46. Each chromel or alumel leg of the thermocouple consists of one flattened contact point, which is spot-welded to the stainless steel rod surface to form a chromel stainless steel alumel junction. The junction wires are flattened to approximately 0.002-in.-thick wafers, to increase the thermocouple frequency response.

The distribution of thermocouples in the core is intended to provide an approximate 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 [4]. As shown in Figure A-47 several thermocouples are attached at various vertical fuel rod positions to measure the axial temperature distribution along a given fuel rod.

6.6 Strain Measurements

Pressure 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 rods and on the reactor vessel wall.

6.7 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 previous Borax and Spert tests.

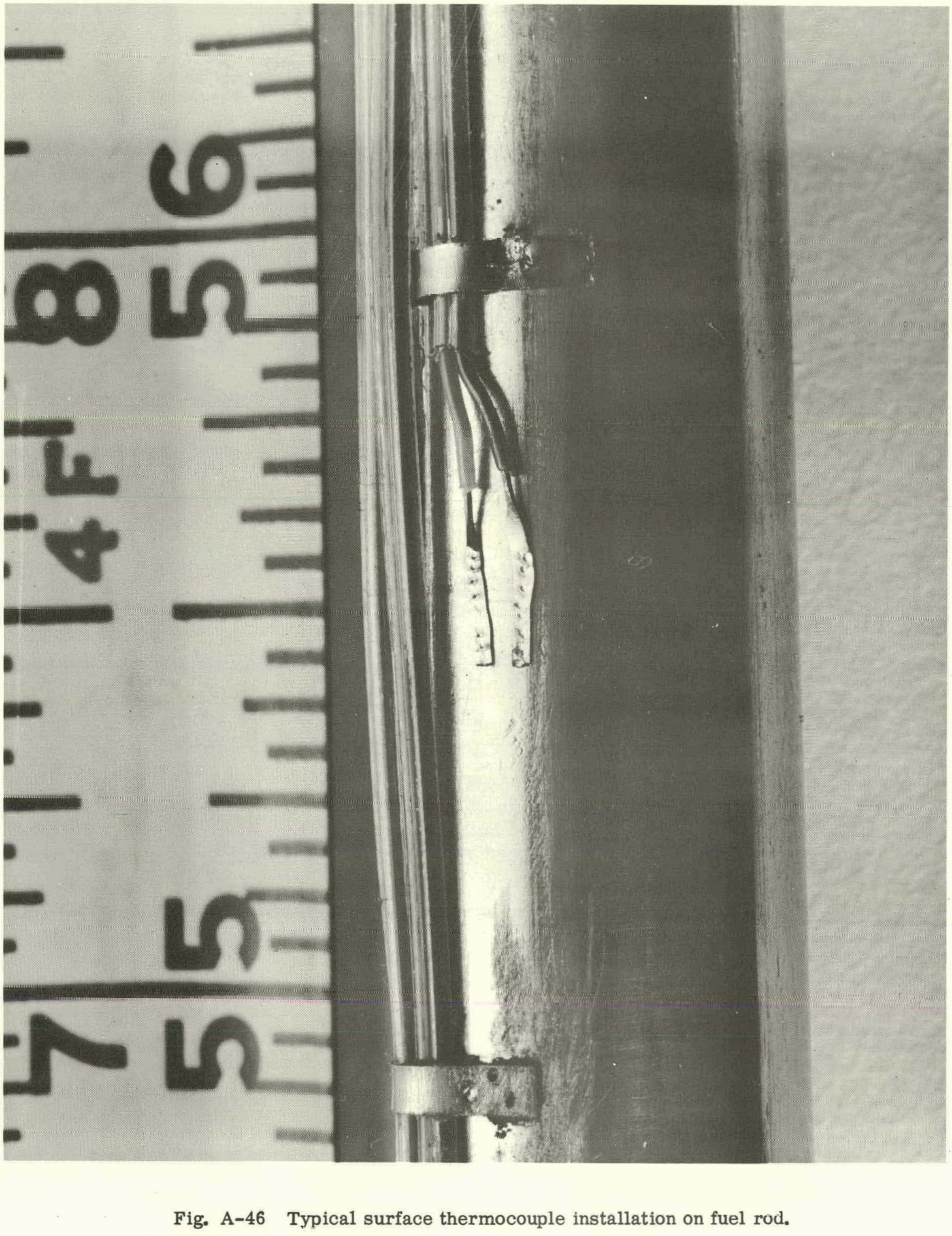

Fig. A-46 Typical surface thermocouple installation on fuel rod.

From acceleration data, calculation can be made of the speed and displacement and of the forces acting on the instrumented component, and can be used to indicate the time of initiation of dislocation.

6.8 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 measurement 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.

6.9 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. 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.

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. The dosage

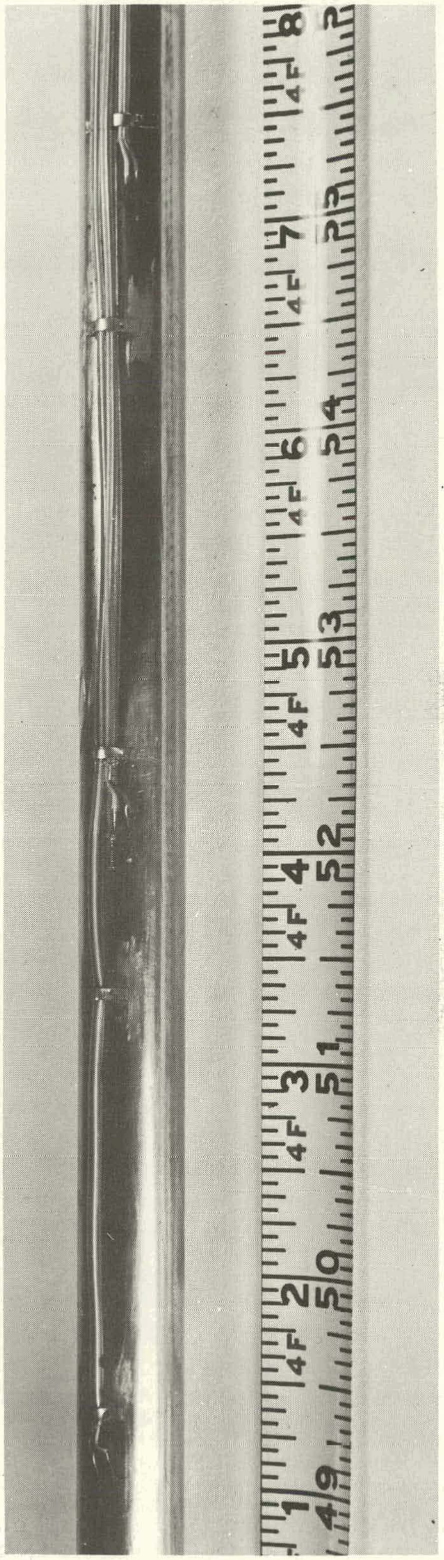

Fig. A-47 View of several surface thermocouples installed on a fuel rod.

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; 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 reentry teams, as described in Section 4. Up to the time that portions of the core may be 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 reentry 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 Test 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 rods, will be used to ascertain the magnitude of the fisside release from damaged fuel pins.

The path of any contamination cloud from a given excursion will be followed with the aid of mobile (including airborne) crews which 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.

6.10 Metallurgical Examination of Reactor Components

The types of damage anticipated to occur in the reactor during the Destructive Test Series will be those associated with high temperatures, thermal stresses to the fuel rods, 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 UO_2 and stainless steel to various thermal shocks, up to and including actual melting of the fuel. 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 rods may be replaced for subsequent excursions. A complete photographic history of core damage will be made by photographing damaged fuel rods as they are removed from the core.

Fuel rod cladding examination consists of hardness determinations, bend tests, chemical analyses, and photomicrographs. Additional testing procedures will be employed as required to maintain a complete history of the core materials throughout the Destructive Test Series.

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

(1) Maximum temperatures of fuel and cladding

(2) Temperature profile of rod

(3) Diffusion of fuel into the cladding

(4) Sintering of fuel

(5) Indication of grain growth

(6) Hot shorting of cladding

(7) Chemical composition of globules of melted fuel rod

(8) Indication of alloying fuel and cladding

(9) Brittleness or ductility of rods

(10) Extent of melting of fuel and cladding.

6.11 Photography

Photographic documentation of the transient tests will be initiated during the fiducial-kinetic 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 any reentry and cleanup operations that are required.

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 Figure A-48 and listed in Table A-III. The cameras located near the core and the reactor building 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 also will permit possible determination of the velocities, momenta, and energies of missiles.

Phase II photography of reactor damage will be performed by photographers accompanying the reentry 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.

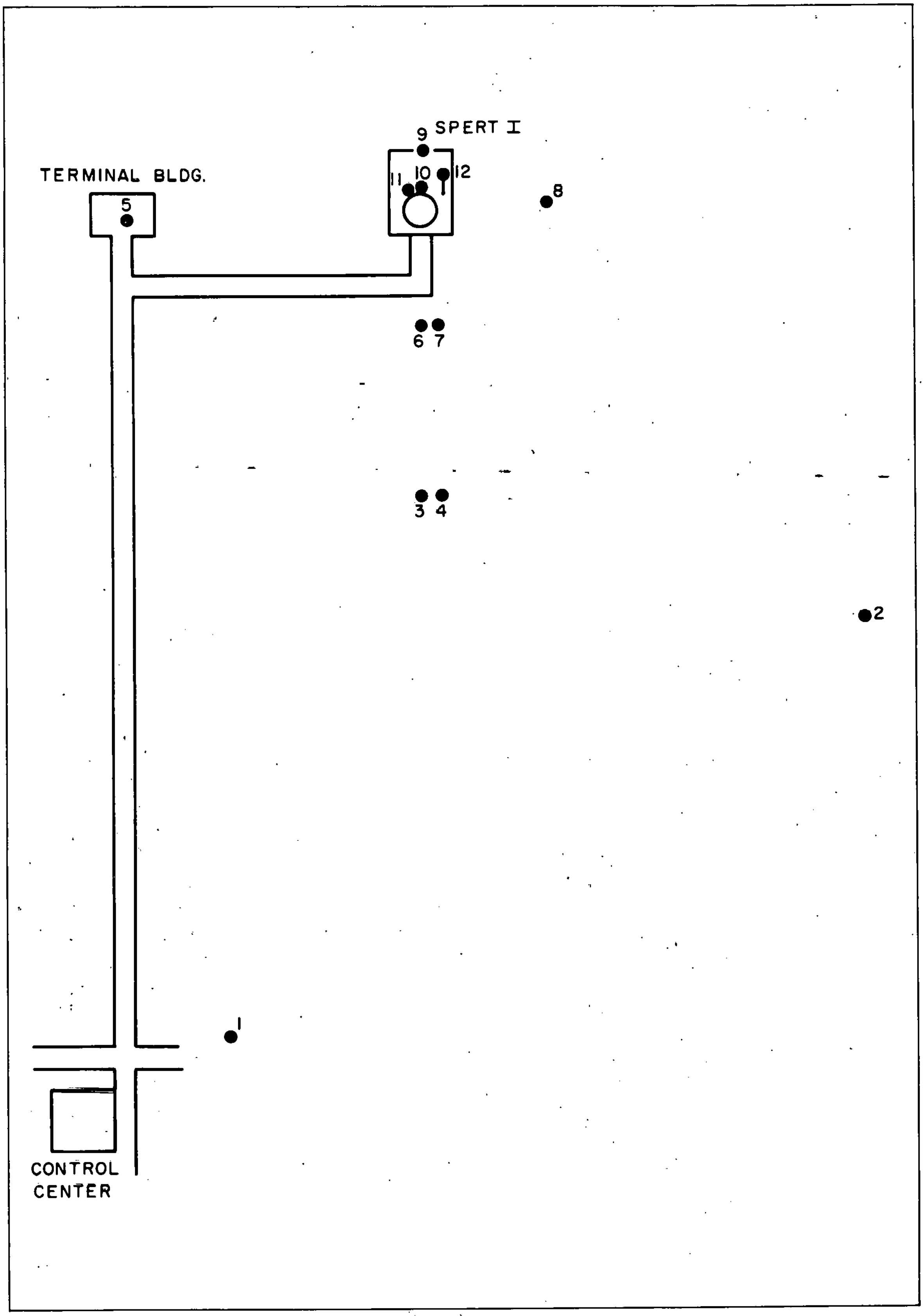

Fig. A-48 Camera locations for Spert I oxide core destructive test.

TABLE A-III

CAMERA LOCATION DESCRIPTION

Camera No.	Description
1	Arriflex 16-mm camera, located near the control center, using color film, a 3-in. lens, and operating at 24 to 48 frames/sec
2	Bell and Howell 16-mm camera, located near the Spert II reactor building, using color film, a 6-in. lens, and operating at 24 to 48 frames/sec
3	Bell and Howell 16-mm camera, located 150 yd in front of Spert I, using color film, a 2- to 3-in. lens, and operating at 128 frames/sec
4	Rapid sequence 70-mm camera, located with camera No. 3, using color film, a 3-in. lens, operating at 20 frames/sec
5	Bell and Howell 16-mm camera, located on the roof of the terminal building, using color film, a 2-in. lens, and operating at 64 frames/sec
6	Bell and Howell 16-mm camera, located 50 yd in front of Spert I, using color film, a 2-in. lens, and operating at 128 frames/sec
7	Waddel 16-mm camera, located beside camera No. 6, using B/W or color film, a 2-in. lens, and operating at 500 to 1000 frames/sec
8	Fastax 16-mm camera, located 50 yd northeast of Spert I, using B/W or color, a 2-in. lens, and operating at 500 to 1000 frames/sec
9	Waddel 16-mm camera, located at the back of the Spert I reactor building, using B/W film, a 16-mm lens, and operating at 500 to 1000 frames/sec
10	Waddel 16-mm camera, located at the periscope looking at the side of the core, using B/W film, a 2-in. lens, and operating at 5000 frames/sec
11	Fastax 16-mm camera, located beside camera No. 10, using B/W or color film, a 2-in. lens, and operating at 500 to 1000 frames/sec
12	Bell and Howell 16-mm camera, located in the remotely operated camera case, using color, a 1-in. lens, and operating at 24 frames/sec

7. LITERATURE CITATIONS

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2. R. W. Miller, "An Experimental Study of Transient Boiling During Spert I Power Excursions", Trans. Am. Nuc. Soc., 4, p 1 (1960). Also reported in Power Reactor Technology, 5, No. 1, pp 22-32 (December 1961).

3. Quarterly Technical Report, Spert Project, 3rd Qtr 1960, IDO-16677, p 3 (May 10, 1961).

4. A. H. Spano, et al, Self-Limiting Power Excursion Tests of a Water-Moderated, Low-Enrichment UO_2 Core in Spert I, IDO-16751 (February 1962).

APPENDIX B - NUCLEAR OPERATION TESTING PROCEDURE*

1. OPERATING PHILOSOPHY

The operation of the Spert reactors is to be carried out in a manner consistent with the overall objectives of the experimental program which requires tests closely approaching and possibly attaining the maximum possible accident with any facility. Since these objectives require the performance of tests under conditions which normally would be considered unsafe for most reactor facilities, administrative control must be relied upon to minimize the possibility of nuclear incidents, to insure the safety of Spert personnel and the NRTS, and to eliminate hazard to the general public.

Because of the nature of the experimental program, the various reactor control systems are, in general, not provided with automatic safety circuits. The control systems of the various Spert reactors do, however, contain numerous interlocks, both electrical and mechanical, in order to reduce the probability of unplanned reactor excursions and to prevent procedures from being followed which could lead to unanticipated situations or unsafe operating conditions. No reactor operation is permitted with any person within approximately 1/3 mile of the reactor.

The application of safe operating practices and recognition and prevention of potentially unsafe acts and situations is the individual responsibility of every member of the Spert organization.

The attitudes reflected in the operating philosophy are based on the physical principles of nuclear chain reacting systems and fissionable materials, the standard procedures employed in similar non-nuclear industrial applications, and acceptable practices for remote testing.

2. 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 Engineering Section is responsible

(*) 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.

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. The Nuclear Test Section is responsible for ensuring that the required operational instrumentation is 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.

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 Section 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. The reactor operator may then initiate routine evacuation of the reactor building.

3. EVACUATION OF REACTOR FACILITY

The reactor area must be evacuated throughout all nuclear 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 by signaling 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. He will then have the security guard turn off the area key switch and remove the key. The reactor area access

gate cannot be opened from the guard house unless the area key switch is activated.

(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 and give the area key to the responsible supervisor in the control room.

The Nuclear Test Section chief, or his designated group leader, is in charge of the facility during reactor operation, and is the responsbile 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.

4. 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 the timer is cycled to insure that it cycles completely and resets. Timer functions are not selected until later. (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.

5. REACTOR SHUTDOWN

Upon completion of the transient test, the reactor is safely shut down 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 reentry of the reactor building has been made.

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

(1) All control rods and the 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 judgment 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 shut down, the responsible supervisor notifies the engineer-in-charge that the reactor has been shut down 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, and reentry is established.

Subsequent to each destructive test, approval for reentry will be given by the responsible supervisor in the Spert I control room with concurrence of either the Spert manager or deputy manager. Special reentry procedures and equipment have been provided for a safe reentry following such approval.

Approval for reentry depends upon a knowledge that the reactor is in a safe shutdown condition as determined by information available at the control center. A safe shutdown condition will not be assumed unless:

(1) Remote operation of the control system will allow, by evaluation of the response of neutron sensing instruments, an unambiguous determination that a safe and stable subcritical state exists.

and/or

(2) Drainage of water from the tank can be effected to a level of no more than approximately 6-in. above the vessel floor as shown by an unambiguous indication of the water level.

In the event that the safe condition cannot be determined by these criteria, special reentry precautions and procedures will be employed. These must be chosen depending on the conditions at the time and must be approved by the AED manager.

APPENDIX C - PREVIOUS TEST RESULTS

The following provides a summary of the previous experimental results and the conclusions derived from them. This was reported previously in IDO-16751 and is reproduced here as extracted from that report.

1. SUMMARY AND CONCLUSIONS

The dynamic behavior of a water-moderated, low-enrichment UO_2 fuel rod reactor has been investigated in the unpressurized Spert I facility. In such a long thermal-time-constant reactor, reactivity compensation for limiting accidental power excursions is inherently available through the long-delayed mode of moderator density effects and through the prompt mode of Doppler broadening. Data on the relative importance of these two modes as a function of initial reactor period were obtained from step- and ramp-induced, self-limiting power excursion tests.

The test core, 6 ft long and 19 in. in equivalent diameter, was comprised of 592 rod-type fuel elements arranged in a square lattice with 0.663-in. pitch. The fuel rods consisted of 0.5-in.-diam steel tubes, each containing 1600 g of 4%-enriched, unsintered UO_2 compressed to an effective density of 9.45 g/cm^3. Experimental measurements of the isothermal temperature coefficient as a function of temperature showed it to be determined primarily by the expansion properties of the water moderator and equivalent to a uniform void coefficient of -39¢ per percent decrease in moderator density. The results of step-initiated power excursion tests, performed over a range of initial reactor periods from 30 sec to a few msec, revealed a period-dependent burst shape, which changed from a single power peaking shape in the long period region ($\tau_0 \gtrsim$ 100 msec) to one having a multiple peaking behavior in the region $\tau_0 \lesssim$ 100 msec. The source of this varying burst shape behavior was determined to be bowing of the long fuel rods, which were supported only at the ends. Subsequent installation of a mechanical constraint prevented rod bowing and eliminated the multiple power peaking behavior.

The results of the Spert oxide core test program lead to the following conclusions:

(1) The energy coefficient of the unconstrained oxide core consisted principally of: (a) a prompt negative term of approximately -3¢ per Mw-sec for 10-msec period tests, due to the Doppler broadening of the U-238 absorption resonances; (b) a delayed negative term of large magnitude, resulting from an effective decrease in moderator density by such mechanisms as fuel rod expansion, moderator heating and boiling; (c) a slightly delayed positive effect, arising as a result of the systematic bowing of the fuel rods during the power burst; and (d) a small prompt negative contribution, arising as a result of direct moderator expansion by radiation heating. In the constrained oxide core, the effects of fuel rod bowing and of reactivity compensation to the time of the first peak by moderator boiling were absent.

(2) Of major importance in considering the safety aspects of low-enriched oxide cores was the actual test demonstration of the effectiveness of the prompt Doppler shutdown mode in the short-period region, where the large negative feedback due to moderator density effects cannot, because of its long time constant, contribute to reactivity quenching. In particular for the Spert core, step-induced power excursions with asymptotic reactor periods as short as 3.2 msec were successfully quenched by the Doppler effect without core damage. Further, analysis of the short-period power excursion data provides information on the magnitude and energy dependence of the Doppler energy coefficient, from which may be deduced a value of the Doppler temperature coefficient to permit comparison with calculation.

(3) The fuel pin bowing effects observed in the unconstrained oxide core indicate that, for cores such as the one tested, positive reactivity effects are possible, which can alter very markedly the burst behavior, give rise to substantially greater energy releases, and consequentially increase the probability of fuel pin rupture. The more violent non-coherent fuel pin motion, which followed after the termination of the power burst as a result of steam pressures and hydraulic effects, also points to the importance of providing adequate constraints on fuel pin motion. Even in the so-called "constrained" core, sizable fuel pin motion was still obtained for the shortest period tests as a result of the modest transient steam pressure generated.

(4) The fact that the core was able to sustain without damage a 3.2-msec period self-limiting power excursion, which required a step reactivity insertion in excess of 2$, implies the following:

(a) In order to obtain a minimum period of 3.2 msec in a ramp-induced, self-limiting power excursion, an estimated ramp rate of about 8 $/sec is required for a sufficient time to inject at least 2$ reactivity into the system. This ramp rate is perhaps a factor of 50 or more greater than that obtainable in a rod withdrawal accident from most control rod drive systems presently in use. The 8 $/sec rate also may cover any reasonable fuel loading accident in an oxide reactor.

(b) Whereas no damage occurred in the oxide core for a 3.2-msec step test, the same test in a plate-type, water-moderated reactor would in all likelihood result in substantial fuel plate melting and in mechanical damage by transient steam pressures. That damage occurs in the one case and not in the other appears to be a consequence of the rapidity with which heat in the plate-type core is transferred from the meat to the cladding, and thence (after the cladding surface temperature has risen substantially above the saturation point), transferred quickly to the water. The sudden violent boiling that occurs gives rise to the destructive transient pressures and, as a result of the decreased plate surface heat transfer coefficient, to the subsequent increase in surface temperature up to the melting point. On the other hand, an oxide fuel rod core with adequate Doppler coefficient need not depend on heat transfer to effect shutdown, and its low heat transfer rate is advantageous in the event of a severe power excursion in helping to maintain the fuel rod cladding temperature near the saturation point and in suppressing rapid steam formation and consequent violent pressures. The implication of these results is that rod-type oxide core may be safer than a plate-type core for a research reactor, and also may provide the basis for an efficient burst facility.

In regard to the overall question of reactor safety, however, consideration must be given to the problem of the consequences of fuel rod rupture. In the event of sudden rupture of a fuel rod containing powdered UO_2 during a short-period transient, there is the possibility of a rapid dumping into the water of nearly all of the burst energy stored in the rod. In this event, since the energy in the powder would be dispersed throughout the water, the transient pressures generated by sudden steam formation might be expected to be much higher than those obtained in the equivalent plate-type reactor accident, so that much greater mechanical damage would then ensue. In fact, if a single hot fuel rod with powdered fuel were to rupture, the resulting pressures might rupture the surrounding rods, which may then cause still others to rupture, etc, giving rise to a sort of chain reaction. This situation, however, can be expected to be significantly improved for the case where the fuel rod contains sintered pellets rather than powdered UO_2, since the dispersal of energy in the event of rupture would then be considerably less rapid than in the powdered oxide case, in accordance with the decrease in overall heat transfer surface-to-mass ratio of the oxide fuel.

2. STEP-TRANSIENT SUMMARY DATA

The following tables contain summary data on control rod positions, reactor period, reactivity insertion, power, and integrated power at the time of peak power for the step-transient test series performed on both the constrained and the unconstrained cores.

TABLE C-I

DATA SUMMARY - UNCONSTRAINED CORE STEP-TRANSIENT TEST SERIES

Run No.	Rod Positions (in.) Critical	Transient	Reactivity Insertion (¢)	Reactor Period, τ_o (msec)	Reciprocal Period, σ_o (sec^{-1})	Maximum Power, ϕ_m (Mw)	First Peak Power $\phi(t_{ml})$ (Mw)	Energy to First Power Peak, $E(t_{ml})$ (Mw-sec)
2	7.92	8.44	20.7	30×10^3	0.033	-----	---	---
64	8.20	9.00	30.6	15.3×10^3	0.066	0.098	---	---
3	7.92	8.85	36.0	10.5×10^3	0.095	0.21	---	---
5	7.92	9.17	47.8	5.9×10^3	0.17	0.69	---	---
63	8.20	9.66	54.1	4.3×10^3	0.232	1.14	---	---
6	7.94	9.68	65.1	2.3×10^3	0.43	3.25	---	---
7	7.96	10.02	76.0	1.2×10^3	0.833	7.07	---	---
8	7.98	10.18	80.5	822	1.22	0.82	---	---
9	7.96	10.28	84.7	542	1.85	13.7	---	---
10	7.97	10.47	90.3	273	3.66	20.9	---	---
11	8.03	10.71	95.3	150	6.67	32.7	---	---
12	7.97	10.71	97.6	114	8.77	37.6	---	---
13	7.99	10.77	99.3	84.2	11.8	39.6	---	---
14	8.02	10.92	102.9	51.8	19.3	48.1	37.7	5.52
15	8.03	10.99	104.5	45.2	22.1	53.7	46.4	5.38
16	8.08	11.11	106.4	35.5	28.2	64.3	63.5	5.67
17	8.06	11.15	108.4	31.2	32.1	77.4	77.4	5.41
62	8.08	11.23	110.0	27.4	36.5	77.8	77.8	4.90
18	8.05	11.22	111.0	25.8	38.7	101	101	5.77
19	8.10	11.40	114.4	20.4	49.0	146	146	6.69
21	8.16	11.59	117.7	17.2	58.1	198	198	7.57
22	8.22	11.74	119.9	15.7	63.7	239	239	8.16
27	8.23	11.84	123.0	14.2	70.4	260	260	7.88
28	8.22	12.05	129.0	11.2	89.3	428	428	10.2
29	8.20	12.14	132.0	10.1	99.0	494	494	11.0
65	8.15	12.13	134.0	9.7	103	622	622	12.6
66	8.16	12.14	134.0	9.7	103	571	571	11.9
75	8.27	12.31	134.0	9.7	103	546	546	11.7
30	8.20	12.40	140.0	8.5	118	720	720	13.0

TABLE C-II

DATA SUMMARY - CONSTRAINED CORE STEP-TRANSIENT TEST SERIES

Run No.	Rod Positions (In.) Critical	Rod Positions (In.) Transient	Reactivity Insertion (¢)	Reactor Period, τ_o (msec)	Reciprocal Period, α_o (sec^{-1})	Peak Power $\phi(t_m)$ (Mw)	Energy to Peak Power, $E(t_m)$ (Mw-sec)
44	10.36	10.97	20.8	31×10^3	0.032	—	—
46	10.41	11.18	26.3	20×10^3	0.050	—	—
47	10.41	11.32	30.6	15×10^3	0.067	—	—
41	10.32	11.41	36.8	10.5×10^3	0.095	0.24	5.93
56	10.45	11.67	40.7	8.1×10^3	0.12	0.31	5.76
45	10.39	11.90	50.3	4.8×10^3	0.21	0.59	7.51
48	10.44	12.46	65.9	1.8×10^3	0.56	1.52	8.36
42	10.36	12.70	75.5	957	1.05	2.48	7.82
49	10.35	13.02	85.5	416	2.41	4.14	6.14
43	10.44	13.24	89.0	273	3.67	6.01	6.50
40	10.31	13.27	94.5	144	6.96	8.70	5.45
52	10.42	13.46	96.4	108	9.26	12.11	3.21
39	10.29	13.41	99.3	73.9	13.5	15.9	2.79
53	10.37	13.60	102.3	44.1	22.7	32.3	3.02
68	10.36	13.69	106.0	33.5	29.9	48.6	3.40
50	10.45	13.89	107.4	29.2	34.2	74.4	4.40
38	10.20	13.92	110.7	17.6	56.8	190	6.76
51	10.49	14.36	118.2	14.0	71.4	259	8.40
54	10.45	14.71	129	10.2	98.1	515	10.76
55	10.43	14.90	135	9.4	106	669	13.51
59	10.49	15.18	140	7.9	127	869	15.12
69	10.38	15.21	145	7.0	143	1126	16.93
61	10.49	15.49	148	6.8	147	1345	18.69
70	10.40	15.50	151	6.3	159	1440	18.35
71	10.42	15.92	161	5.0	179	2010	24.0
72	10.46	16.41	172	4.8	208	2820	29.0
73	10.41	17.06	190	3.8	263	4930	40.0
74	10.47	17.97	210	3.2	313	7480	51.3

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