~~Bk Rawlton 12-12~~ ~~6120 53092~~
~~D. Nishimoto~~

080
46852

IDO-16491
Reactors General
TID-4500 (14th Ed.)

INEEL Technical Library
122735

SPERT II HAZARDS SUMMARY REPORT

G. O. Bright and J. E. Grund

February 17, 1959

AEC RESEARCH AND DEVELOPMENT REPORT

LOAN COPY
THIS REPORT MAY BE RECALLED
AFTER TWO WEEKS. PLEASE
RETURN PROMPTLY TO:
INEL TECHNICAL LIBRARY

LIBRARY LOAN COPY

LOAN COPY
THIS REPORT MAY BE RECALLED
AFTER TWO WEEKS. PLEASE
RETURN PROMPTLY TO:
INEL TECHNICAL LIBRARY

~~J. A. McClure 4-11-63~~
~~M. A. Kramer 8-1-67~~
~~Pete Pagel 4/1/83~~
~~D. W. KNIGHT 7-25-86~~
~~Lee Plansky 5-8-91~~
~~6-23-93~~
Rusty Grant 2-6-95

PHILLIPS PETROLEUM CO.
ATOMIC ENERGY DIVISION
(UNDER CONTRACT NO. AT (10-1)-205)
IDAHO OPERATIONS OFFICE
U. S. ATOMIC ENERGY COMMISSION

04-15-2010

PRICE $2.00

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

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission to the extent that such employee or contractor prepares, handles or distributes, or provides access to, any information pursuant to his employment or contract with the Commission.

122735

SPERT II HAZARDS SUMMARY REPORT

by

G. O. Bright
and
J. E. Grund

Date Report Written

December 10, 1958

PHILLIPS PETROLEUM COMPANY
Atomic Energy Division
At(10-1)-205

IDAHO OPERATIONS OFFICE

U. S. ATOMIC ENERGY COMMISSION

SPERT II HAZARDS SUMMARY REPORT

by

G. O. Bright
and
J. E. Grund

ABSTRACT

Spert II will be a heterogeneous, water-cooled and -moderated reactor designed for operation at pressures up to 300 psig and temperatures up to 400°F. The facility will provide for both static and flow tests with reactor cores employing either light or heavy water as a moderator-reflector, and also tests utilizing solid reflector materials. Its primary function is to provide a facility to permit study of the influence of various types of moderators and reflectors on reactor kinetic behavior. This report discusses the major hazards present in the operation of this facility, the precautions to be taken to reduce the probability of an accident, and the consequences of the maximum possible accident. Brief descriptions of the Spert II design and the site location are also presented.

It is concluded that the proposed method of operation and the site location make possible the operation of the Spert II facility without hazard to the general public.

TABLE OF CONTENTS

APPENDIX A CONTROL SYSTEM DESIGN

APPENDIX B OPERATING RULES AND PROCEDURES

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

LIST OF TABLES

SPERT II HAZARDS SUMMARY REPORT

by

G. O. Bright
and
J. E. Grund

I. INTRODUCTION

The Spert project is operated by Phillips Petroleum Company at the National Reactor Testing Station as a part of the Atomic Energy Commission's reactor safety program. The Spert II reactor is chronologically the third of a series of Spert reactor facilities to be placed in operation for the investigation of the kinetic behavior of heterogeneous, water-moderated reactor systems. Spert II will provide a moderate pressure and temperature facility, 300 psig and 400°F, whose primary function will be to permit the study of the influence of various types of moderators and reflectors on reactor kinetic behavior. The facility will provide for both static and flow tests with reactor cores employing either light or heavy water as a moderator-reflector, and also tests utilizing solid reflector materials. A description of the facility including preliminary design data is presented in section III of this report.

The experimental program for Spert II includes tests of several types:

(1) Tests in which reactivity is inserted instantaneously into the nearly critical system;
(2) Tests in which reactivity is added to the system at various rates in an approximately linear fashion with time;
(3) Tests in which the stability of the system is investigated;
(4) Static tests, which are essentially critical experiments.

These tests will be performed under a variety of initial conditions of reactor power level, water temperature, pressure, flow rate, flow direction, core geometry, moderator and reflector properties, and will involve reactivity insertions above prompt critical.

The facility is intended only as a transient test facility and consequently no heat removal equipment is provided. Extended high power operation is not possible, which insures that the fission product source contained in the Spert II reactor core will always be small in comparison to that contained in conventional power reactors. The fission product activity which might be released in an accident will in fact consist almost exclusively of those short-lived products of the initiating incident.

It is the purpose of this report to set down briefly the major hazards inherent in the operation of this facility, the precautions to be taken to reduce the probability of an accident, and the possible consequences of such an accident.

II. DESCRIPTION OF SITE

The Spert II facility is located within the Spert area at the National Reactor Testing Station in Idaho. The Spert area site is approximately 4 miles east-northeast of the Central Facilities area, as shown in Figure 1 which is a map of the testing station. Figure 2 shows the general site plan for Spert and the location of three Spert reactors and two possible future site locations[1]. The principal work area at the Spert site is the Control Center area. Each reactor area is approximately 1/2 mile from the Control Center and 1/2 mile from the adjacent reactor sites.

Meteorological, hydrological, and seismological data for the NRTS have been described in previous reports[2][3].

Figures 3 and 4 show wind roses at the 20 ft and 250 ft levels for the Weather Bureau Office located in the Central Facilities area, approximately 4 miles distant from the Spert area. The prevailing southwesterly winds occur most frequently in the afternoons during all seasons. The most favorable conditions for diffusion of radioactive cloud material likewise prevail during the afternoon.

The closest population center in the general down wind direction is the Mud Lake-Terreton area 30 miles to the northeast. The site boundary in that direction is approximately 20 miles distant. Other nearby points of importance are: MTR-ETR about 5 miles northwest; CPP about 3-1/2 miles west-northwest; Central Facilities about 4 miles west-southwest; OMRE about 3 miles south-southwest; ALPR about 3-1/2 miles southeast; nearest point of U. S. Highway 20 about 3 miles south; and the nearest site boundary about 6-1/2 miles south.

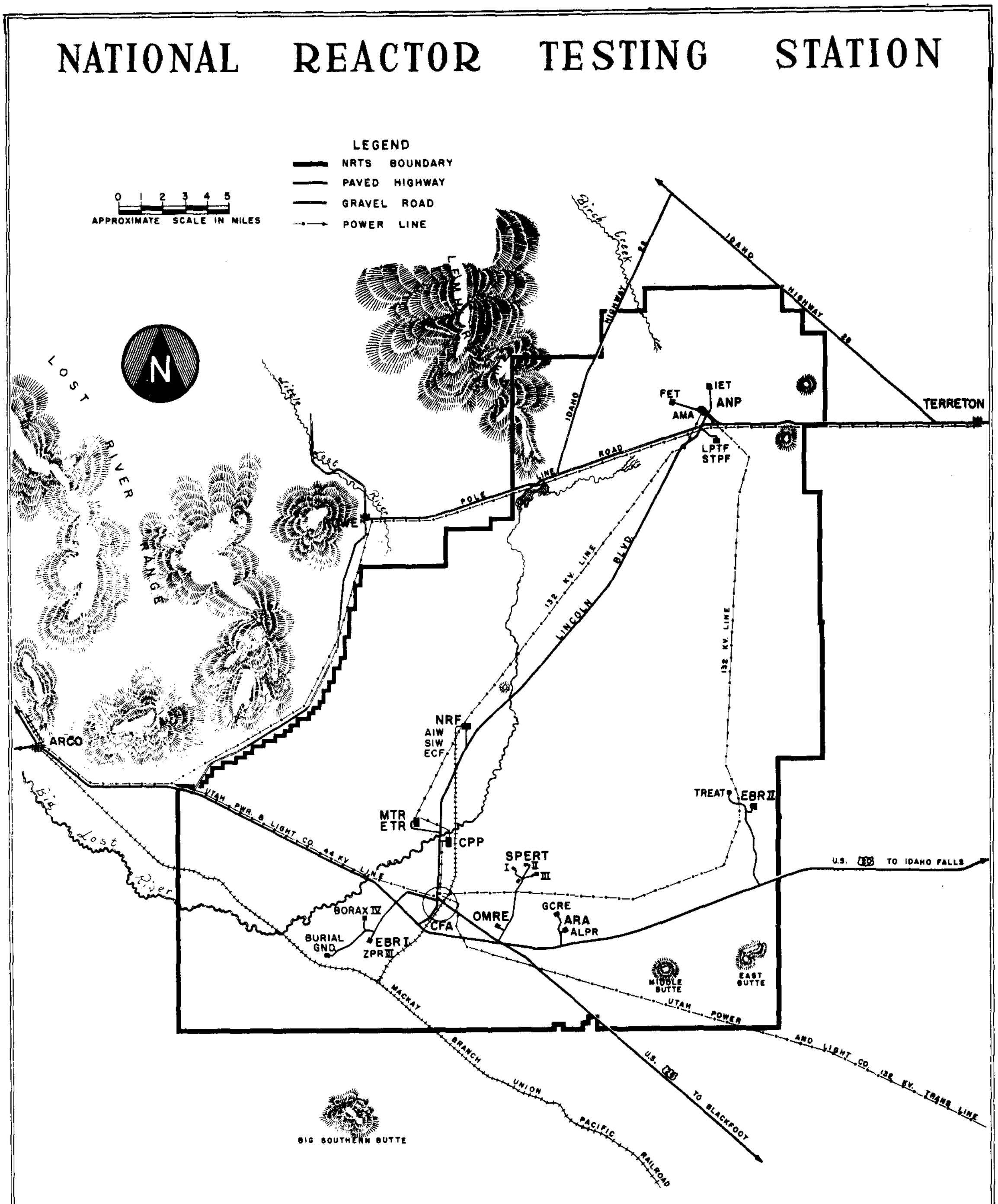

NATIONAL REACTOR TESTING STATION
LEGEND
NRTS BOUNDARY
PAVED HIGHWAY
GRAVEL ROAD
POWER LINE
0 1 2 3 4 5
APPROXIMATE SCALE IN MILES
N
LOST RIVER RANGE
LEMHI RANGE
Birch Creek
IDAHO HIGHWAY 88
IDAHO HIGHWAY 28
Little Lost River
HOWE
POLE LINE ROAD
TERRETON
FET
IET
ANP
AMA
LPTF
STPF
132 KV LINE
LINCOLN BLVD
ARCO
Big Lost River
NRF
AIW
SIW
ECF
MTR
ETR
CPP
UTAH PWR. & LIGHT CO. 44 KV LINE
TREAT
EBR II
SPERT
I
II
III
U.S. 20 TO IDAHO FALLS
BORAX IV
BURIAL GND
EBR I
ZPR III
CFA
OMRE
GCRE
ARA
ALPR
MIDDLE BUTTE
EAST BUTTE
UTAH POWER AND LIGHT CO. 138 KV. TRANS LINE
MACKAY BRANCH UNION PACIFIC RAILROAD
U.S. 26 TO BLACKFOOT
BIG SOUTHERN BUTTE

Figure 1

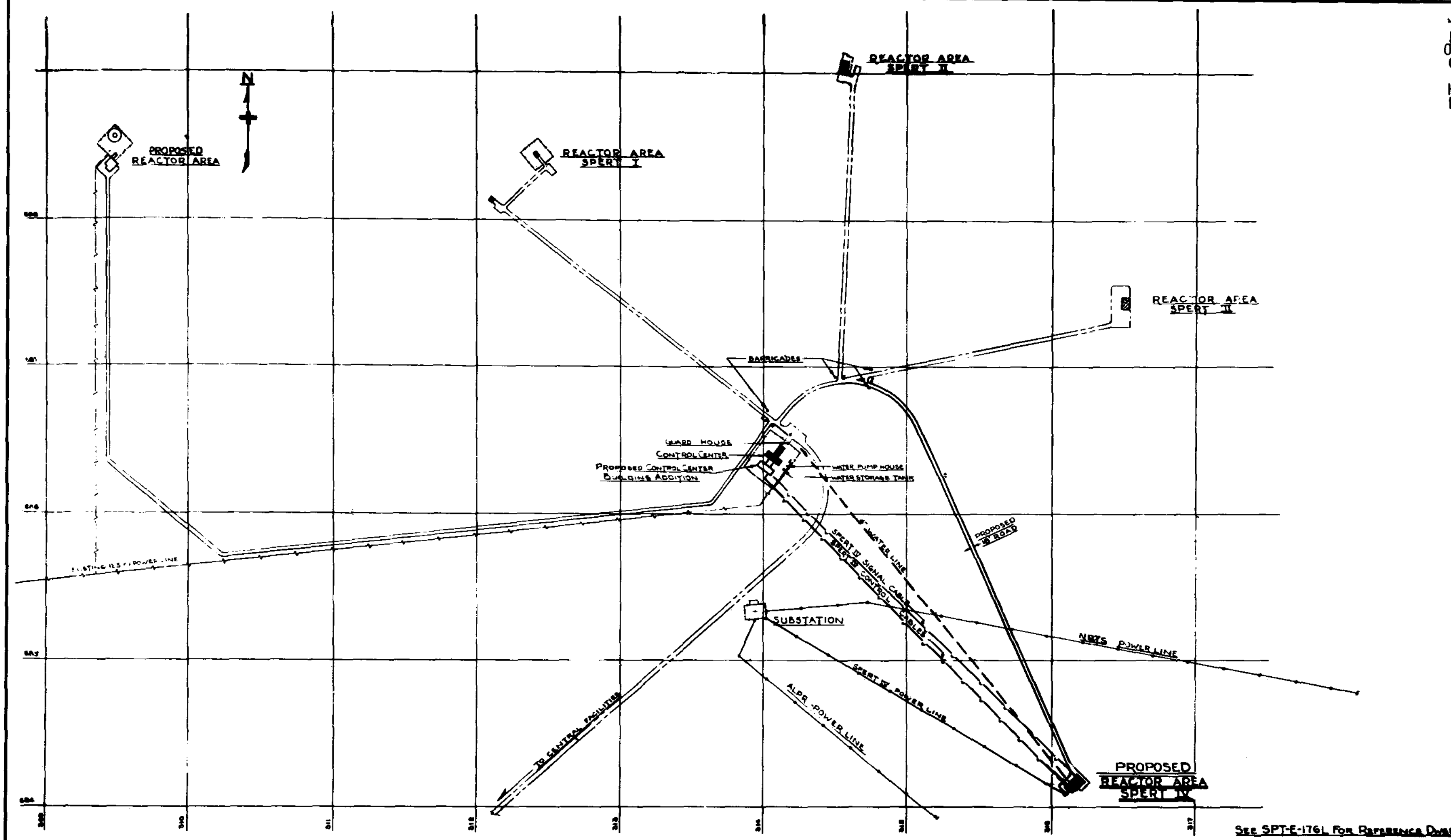


Figure 2

SPERT
GENERAL SITE PLAN
PROPOSAL

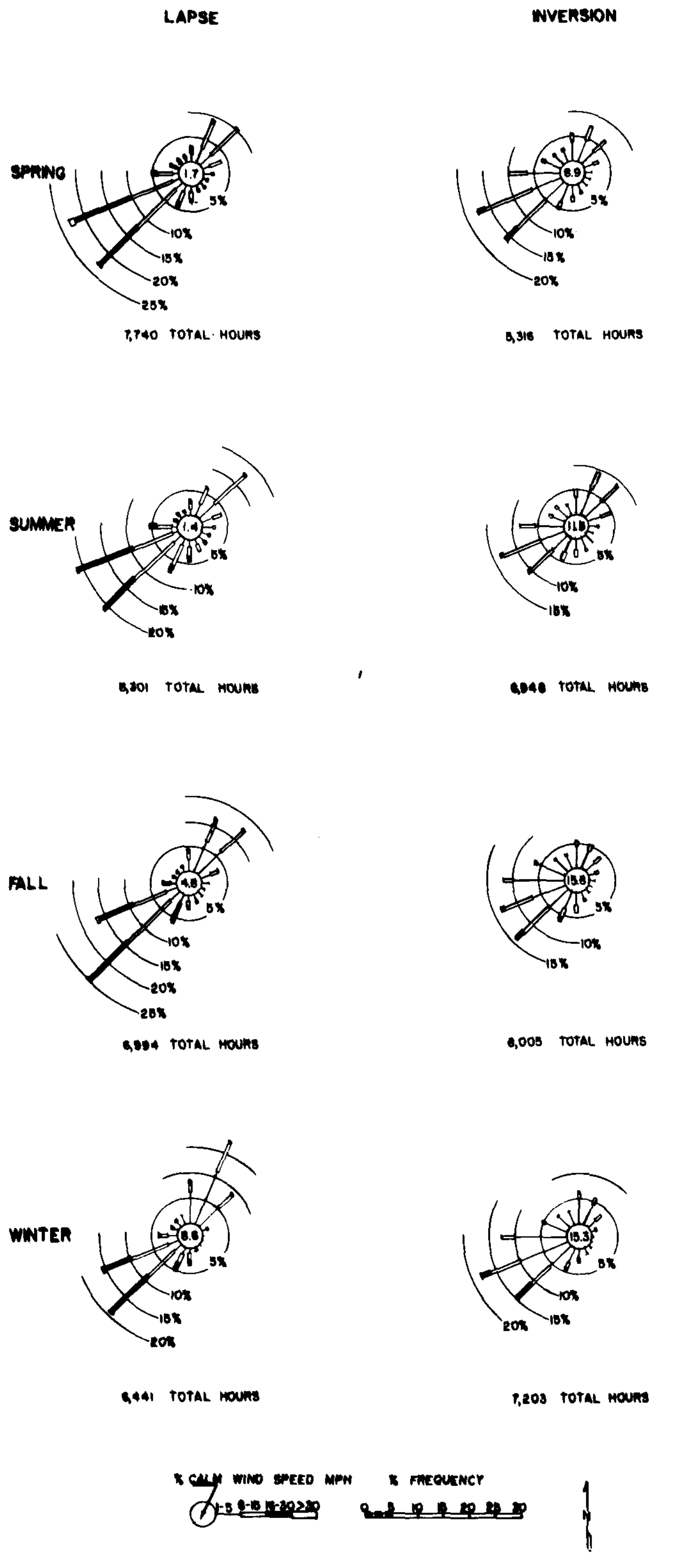


FIGURE 3 WBO 20 FOOT LEVEL WIND ROSES
1950 THROUGH 1956

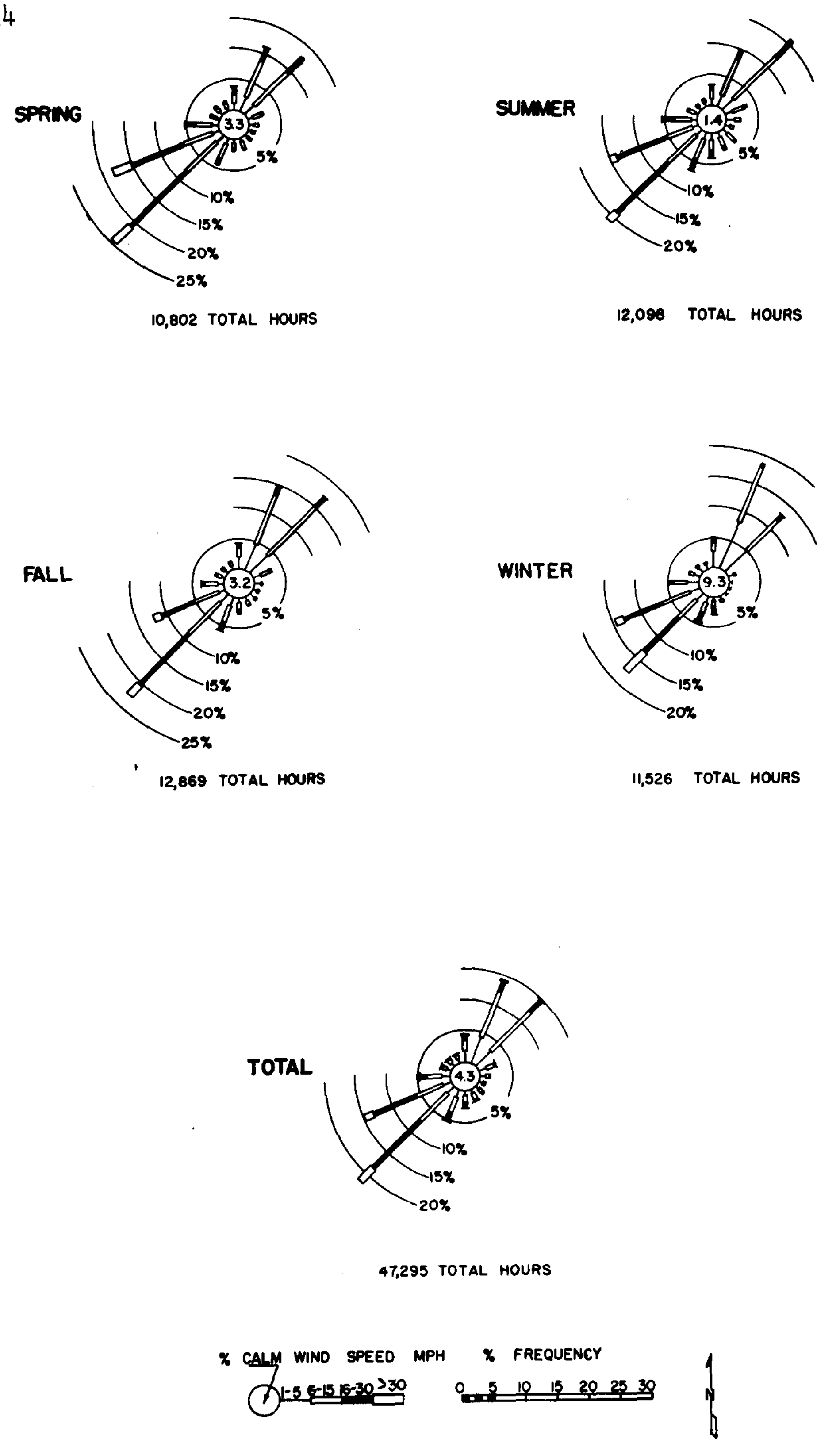


FIGURE 4 WBO 250 FOOT LEVEL WIND ROSES
JULY 1951 THROUGH DEC 1956

III. DESCRIPTION OF THE FACILITY

The following description of the Spert II reactor facility has been drawn from previous reports[4][5][6]. It is reproduced here in part with such additions and changes as are applicable.

A. Design Objectives

The Spert II reactor is a research facility whose purpose is to provide basic experimental information on reactor kinetic behavior and reactor safety. The design objectives are to provide a flexible facility for reactor transient and oscillator studies. The general design criteria for the reactor and coolant system are as follows:

(1) Operation up to 300 psig and 400°F;

(2) Coolant flow, both upward and downward through the core, at velocities up to about 25 ft/sec; however, no provisions for heat removal (i.e., power operation) are included;

(3) Use of both light water and heavy water as the coolant-moderator-reflector with means for varying the reflector thickness. In addition, provisions for the possible use of solid reflector materials are included;

(4) Unobstructed access to the reactor core as well as provisions for changing cores with a minimum of expense;

(5) Remote operation and control of the reactor and principal process system parameters from a distance of 1/2 mile.

B. General Description

1. Physical Plant

The Spert II facility is comprised of two separate areas; a control center area and the reactor area.

The control center area is 1/2 mile distant from the reactor area and contains all the mechanisms for remote operation of the reactor and control of important process system variables.

The reactor area includes the reactor building which encloses the reactor vessel, coolant flow system, heavy water system, and the necessary auxiliary equipment for operating a reactor of this type; such as, a water treating system, compressed air system, etc. The reactor vessel and coolant flow systems are designed for operation at pressures up to 300 psig, temperatures up to 400°F, and flow rates up to 20,000 gpm with either light water or heavy water.

2. Reactor Physics

The design of the Spert II facility is such that various fuel loadings and moderator-coolant-reflector configurations having different physical properties and nuclear parameters may be investigated. Consequently it is impractical to state the specific properties of all possible cores. As is discussed in Section IV of this report, this does not influence the treatment of the hazards presented here since a maximum accident is postulated. Therefore, no discussion will be presented of the calculated nuclear properties of the various core loadings under consideration. For general information and orientation, the following sections will describe briefly the reactor plant along with the initial fuel and core structure to be used.

C. Reactor Components

The Spert II reactor assembly, Figure 5, consists of the reactor core with the control rods and drives, the reactor core structure, removable inner shells with their support structure for the containment of water in and around the core in varying thicknesses, and the reactor pressure vessel with a full diameter flanged top head. This arrangement of the reactor provides for cores of various diameters and heights with varying thicknesses of reflector, either liquid or solid, up to 10 ft in diameter. Inlet and exit coolant piping is attached to the reactor vessel bottom head.

1. Reactor Vessel

The reactor vessel is a stainless steel clad, carbon steel vessel, 10 ft ID by about 16 ft high. The vessel is designed for a pressure of 375 psig at 400°F, and is equipped with a full opening hemispherical head upon which is mounted a 42 in. diameter top plug containing the control rod drives. During normal shutdown operations only the top plug will be removed; however, the full opening head may be removed to change or remove inner shells, or to place solid reflector pieces around the core.

The vessel bottom head contains a 42 in. diameter spool extending downward from the head. The spool is equipped with electric immersion heaters having a capacity of 200 kw for heating the moderator-coolant to the desired temperature, and also contains a 24 in. coolant nozzle. In addition to the spool, the vessel bottom head is penetrated by four additional nozzles for the introduction or removal of coolant, depending upon the desired direction of coolant flow through the core.

Nozzles for the removal of instrument leads are provided in the top head, and at several elevations around the vessel shell. In addition to the electric heaters in the bottom well, an additional 400 kw of heating capacity is supplied by side arm heaters in the vessel shell.

2. Internal Structure

The reactor core and the removable inner shells are supported by an internal structure from the bottom of the vessel. An aluminum flow skirt 42 in. in diameter and surrounding the core, directs the flow of the coolant

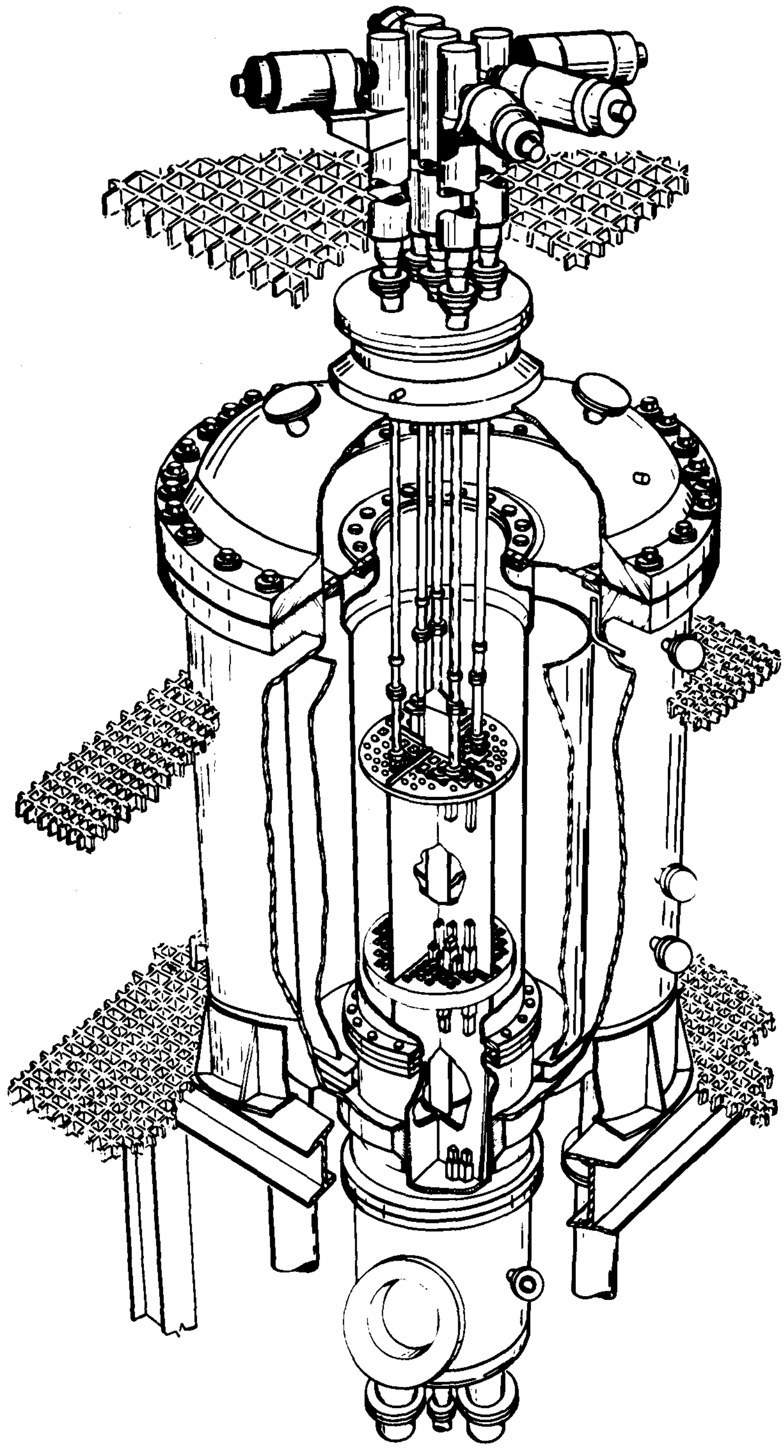


Figure 5

SPERT II
REACTOR VESSEL ASSEMBLY

through the core, and separates the core from the balance of the vessel to a height of about 4 ft above the core. Coolant flow may be directed either upward or downward through the core. Coolant entering the bottom well flows either upward through the core, over the top of the core skirt, downward outside the core skirt, and exits from the bottom of the vessel, or vice versa.

In order to achieve the desired reflector thicknesses using light or heavy water, a series of removable inner shells of different diameter may be employed. The shells contain a poison material which effectively establishes the reflector thickness by preventing neutrons which reach the shell from being reflected back to the core. Normally, only one shell of the desired diameter will be used at a time.

Low pressure, non-flow tests utilizing light water as a moderator and heavy water as a reflector, or vice versa may be conducted by use of a diaphragm extending from the top of the core skirt to the vessel shell. The diaphragm separates the two liquids by isolating the core region from the region outside the core skirt.

Figure 6 is a cross-sectional view of the reactor core. The core is divided into four symmetrical quadrants by a 3/4 in. thick cross consisting of aluminum and water. In the center of the cross is a blade type transient rod containing poison in the bottom section, which may be ejected to initiate reactor power excursions. The aluminum cross was dictated by experimental rather than engineering requirements. The advantages of the cross are that it provides space for the transient rod while utilizing nuclear fuel assemblies of only one size, and duplicates the geometry and nuclear characteristics of the Spert I reactor, enabling a close experimental tie between the two facilities.

The reactor will be controlled by eight fuel-poison type control rods in the positions labeled "C". Shown in Figure 6 is a 28 assembly core; however, positions are provided in the core for 96 fuel assemblies including the control rod fuel sections. Unoccupied fuel positions will be blocked during experiments requiring coolant flow.

Surrounding the core is an aluminum core skirt. Shown also in Figure 6 is a poison skirt for establishing the reflector thickness as previously described.

3. Fuel Assemblies

Figure 7 is an isometric sketch of the initial fuel assembly type to be used in the Spert II reactor. This assembly type was selected since experiments are now being conducted on transient behavior with these assemblies in the Spert I reactor, and will allow experimental correlation between the two reactors. The assembly is a plate type uranium-aluminum unit containing four permanently brazed fuel plates and 20 removable fuel plates. Design of the assembly was for the specific purpose of determining the effect of plate spacing on reactor behavior. With the full complement of 24 plates, the assembly contains 168 g of U^{235} and the plate spacing is

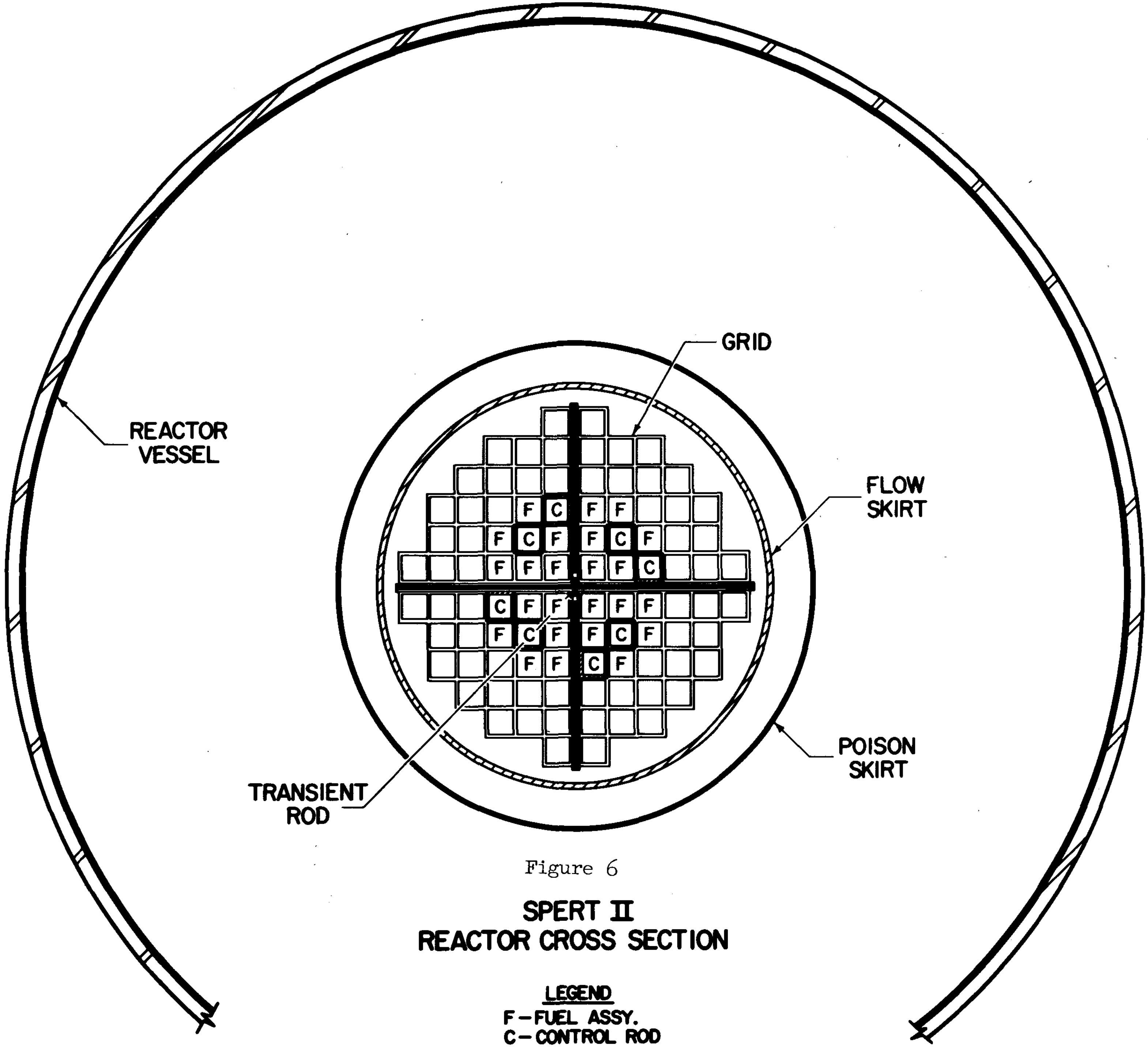


Figure 6

SPERT II
REACTOR CROSS SECTION

LEGEND
F – FUEL ASSY.
C – CONTROL ROD

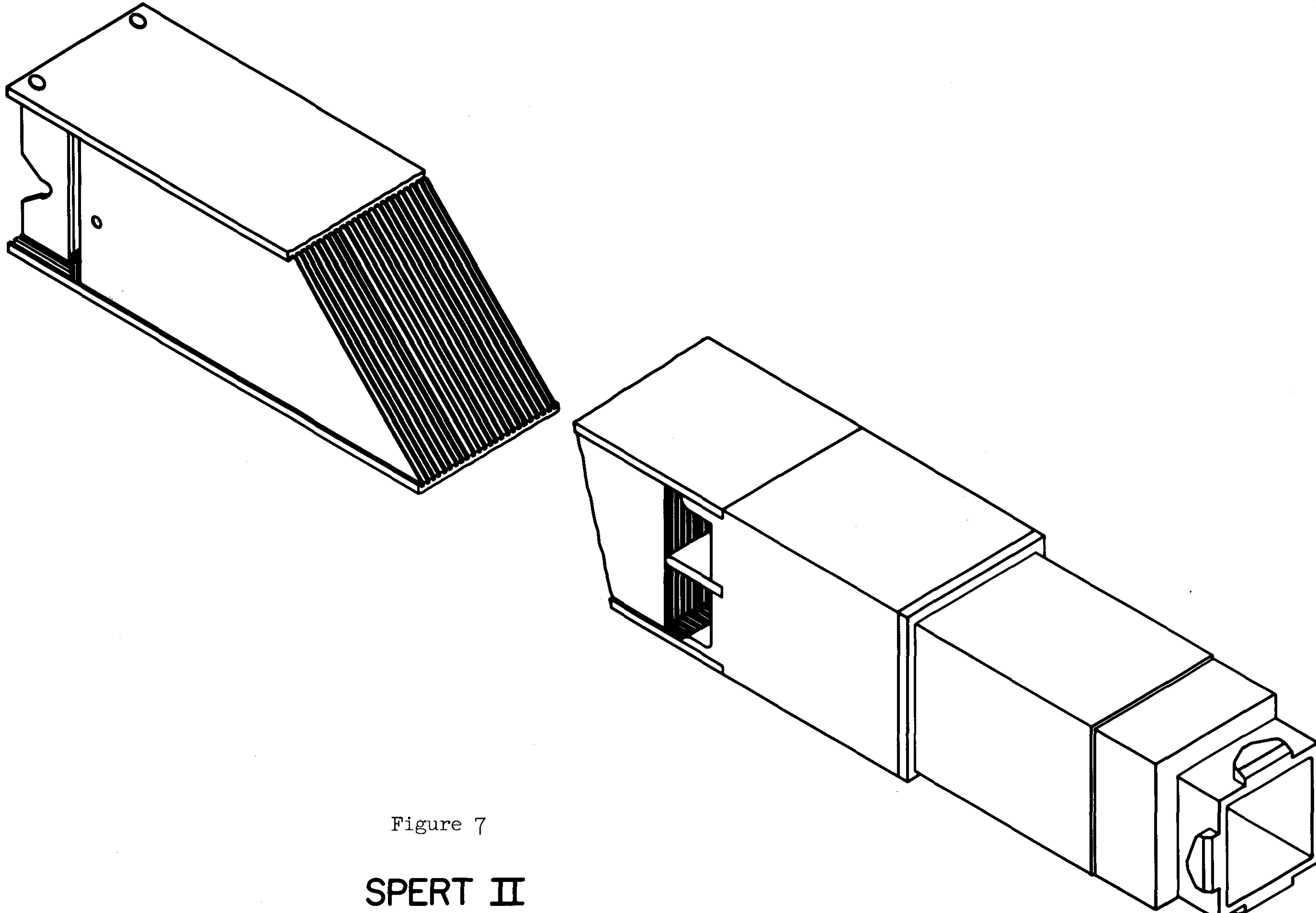


Figure 7

SPERT II
FUEL ASSEMBLY

~1/16 in. By removing every other plate, the unit will contain 84 g of U^{235} and have a plate spacing of ~3/16 in., etc. End box adapters which mate with the lower support grid are riveted to the lower end to complete the fuel assembly.

Each fuel plate consists of a uranium-aluminum alloy core 0.020 in. thick and clad on each surface with 6061 aluminum alloy 0.020 in. thick. The active core height of the plates is 24 in.

4. Control Rods

A pair of control rods is shown in Figure 8. The control rods, containing fuel in the lower section and a cadmium poison sandwich in the upper section, are driven by a single drive mechanism. This type of rod was dictated by experimental requirements; since it was necessary to minimize the number of rods to permit access to the core for instrumentation purposes, the reactivity worth per rod must be maximized. The fuel section of the rod is constructed to permit the adjustment of the number of fuel plates in the rods to agree with that in the fuel assemblies. The poison section consists of a stainless steel box on which a 0.020 in. layer of cadmium and a 0.040 in. layer of stainless steel has been metal sprayed. The eight control rods are used for normal start-up and shutdown of the reactor.

Initially the transient rod is to be a blade-type rod containing poison in the lower section. The rod will be fabricated of 0.020 in. cadmium sheet clad with aluminum. The cadmium will be held in position by seal welded rivets extending through the Al-Cd-Al sandwich.

5. Control Rod Drives

Five rod mechanisms are mounted on the reactor top plug as shown in Figure 5. Four of the drives position the eight control rods and the fifth positions the transient rod. Although the function of the two rod drive types is somewhat different, the mechanical design is essentially identical. In an effort to standardize the mechanical components of the drives for both the Spert II and Spert III reactors, a combination air and three speed electric motor drive designed for the Spert III reactor will be used.

A cross section of a rod drive assembly is presented in Figure 9. The drive basically consists of a cylinder, a piston, a shock absorber, and a drive screw and motor. No direct connection exists between the piston, which is joined to the control rod, and the drive screw and motor. Air pressure on the bottom side of the piston maintains the piston in contact with the screw, thus permitting positioning of the control rod with the drive motor and screw. Air at slightly less pressure, yet sufficient to drive the rod in against reactor pressure is maintained on the top side of the piston. Rapid insertion of the rods or "scram" is accomplished by releasing the air on the bottom of the piston. An air reservoir of sufficient capacity to drive in and hold the rod in the down or safe position is built into the upper cylinder.

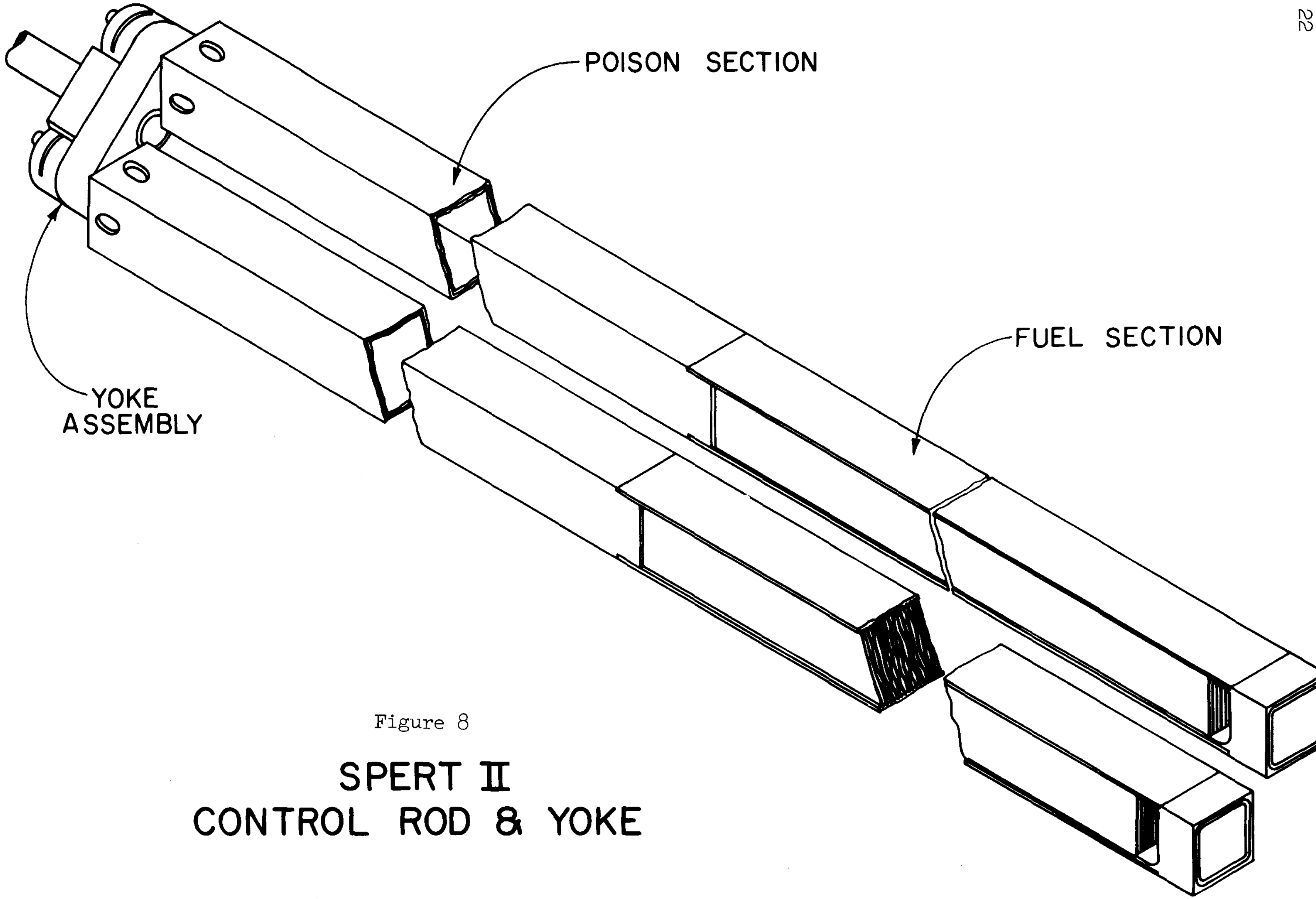


Figure 8

SPERT II
CONTROL ROD & YOKE

In the case of the control rod drives a spring-return latch is provided which mechanically locks the rod pistons in the seated position unless electrically unlocked by the operator. The transient rod drive contains a latch mechanism to mechanically secure the rod piston to the ball-nut extension until unlocked by the operator in preparation for a test. Neither of these two latches is shown in Figure 9.

D. Process Systems

A schematic flow diagram of the Spert II process system is shown in Figure 10. The principal components are the primary circulating loop, a pressurizer vessel with helium pressurizing system, and a heavy water system.

1. Primary Coolant System

The primary coolant circulation loop is designed for use with either light or heavy water, and, by means of a crossover, permits the circulation of coolant either upward or downward through the reactor core. The primary loop piping is stainless steel, type 304L, and is designed to assure maximum leak integrity and minimum holdup of the coolant. Where feasible, valves have been located in vertical runs to minimize liquid holdup. Generally speaking, when light water is in use in the reactor vessel and primary coolant loop, the heavy water system will be completely isolated from the loop by removable spools, and vice versa.

The two primary coolant pumps are conventional horizontal, single stage, centrifugal pumps having a maximum capacity of 10,000 gpm each and a net differential head of 175 ft of water. A unique feature of the pumps is that they are coupled to their electric motor drives through fluid drives. The fluid drives permit the adjustment of pump speed; thus, the flow rate may be varied from 10,000 gpm to as low as 2,000 gpm. Further flow reduction down to 200 gpm is permissible with an 8 in. flow control valve.

The flow rate is obtained by either of two flow tubes designed to accurately measure the flow rate over the range of flows from 200 to 20,000 gpm.

2. Pressurizing System

The reactor and coolant system are pressurized by means of the pressurizer vessel, utilizing high pressure helium. The pressurizer is directly connected to the reactor vessel and is mounted in relation to the reactor vessel approximately as shown in Figure 10.

In addition to system pressurization, low pressure helium will be utilized to purge and/or blow down the system as required. Helium released to the atmosphere for any purpose, such as depressurization, will be first directed through a cold trap to recover D_2O vapor from the helium.

3. Heavy Water System

The heavy water system is comprised of a number of small systems including the heavy water storage vessel, a transfer system, a clean-up loop, and a D_2O recovery system.

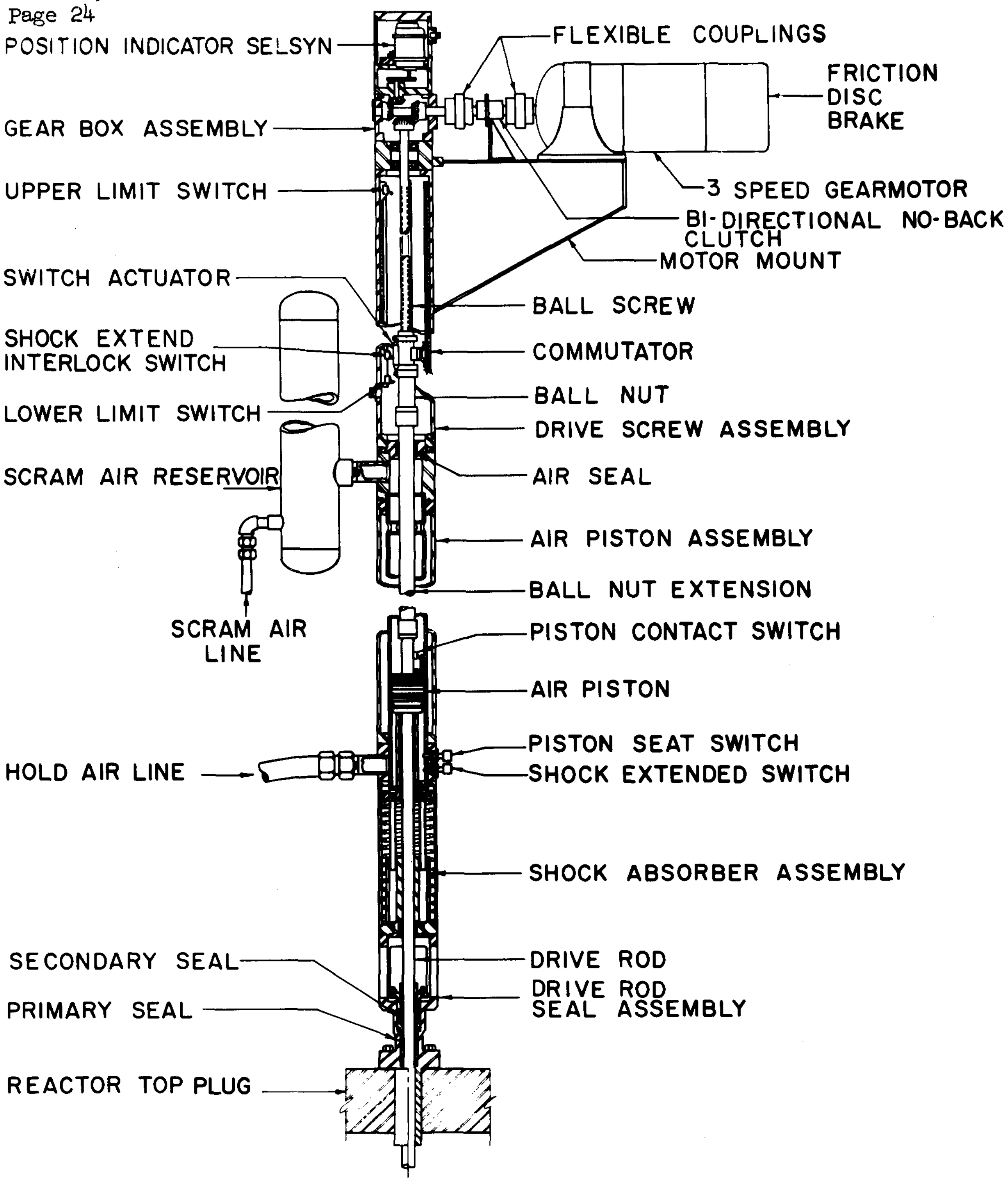
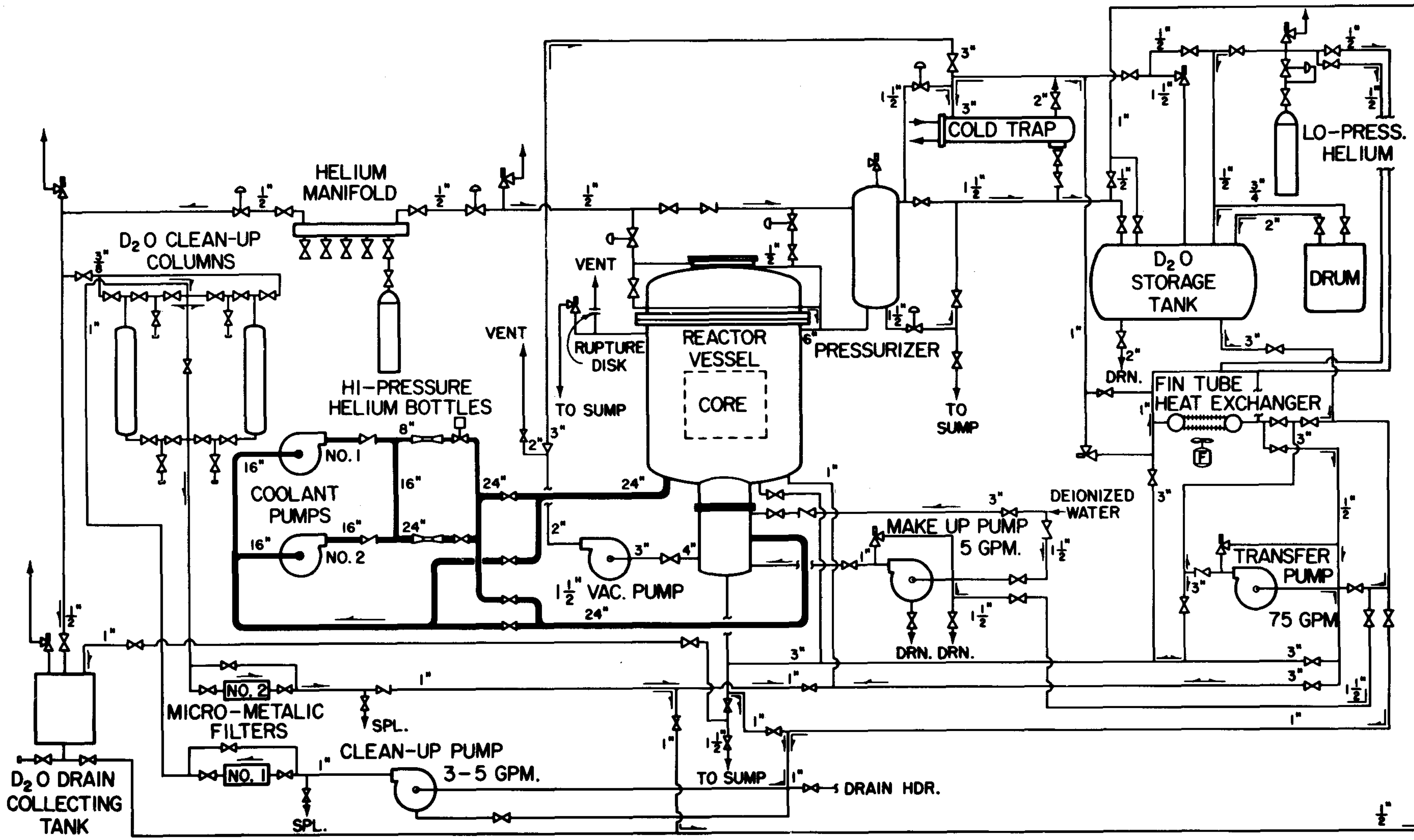


Figure 9

SPERT II
CONTROL ROD DRIVE
SCHEMATIC ELEVATION



SPERT II
PROCESS FLOW DIAGRAM

Figure 10

Heavy water is stored in a 14,500 gal stainless steel tank, which is provided with a helium gas blanket to a positive pressure of about 0.5 in. of H_2O. Heavy water may be transferred from storage to the reactor vessel, or vice versa, by means of a 75 gpm canned motor pump. In order to cool the heavy water in the vessel below the boiling point before returning it to storage, or opening the vessel, a forced draft fin-tube heat exchanger has been incorporated in the transfer loop. The heat exchanger is capable of reducing the temperature of the D_2O from 400°F to 100°F in about 12 hours.

Heavy water purity is maintained by a clean-up loop consisting of a 5 gpm canned motor pump, two 35 micron stainless steel filters, and two mixed-bed ion exchange columns. Heavy water located in storage or in the primary system may be directed through the loop and returned as desired. The ion exchange columns contain about 1 cu ft of mixed anion-cation resin each and are valved for removal without draining. Secondary equipment includes both pH and conductivity meters and a sampling station. The pH of the D_2O will be maintained at 6.5 to 7.0 and the resistivity to about 10^6 ohm-cm.

For experimental purposes, it is necessary to alternately use light and heavy water in the reactor vessel and coolant system. The possibility of contaminating heavy water has necessitated that means be provided for thoroughly drying the vessel and coolant system. Drying will be accomplished by placing the system under vacuum. A vacuum pump having an air displacement of 130 cfm and capable of reducing the pressure to about 40 microns is included for this purpose.

E. Auxiliary Equipment

The auxiliary equipment associated with the facility includes a water treating system, two compressed air systems, and an emergency power supply.

High purity deionized water having a resistivity of approximately 10^6 ohm-cm will be used in the primary coolant system. The deionized water is to be supplied by a 30 gpm softener and a 30 gpm conventional mixed-bed deionizer. The softener-deionizer combination has been employed to reduce operating costs by increasing the capacity and quality of the deionizer unit. A 14 000 gal capacity, rubber lined, deionized water storage tank located outside the building is also included in the water treatment system.

A 100 psi compressed air system is to be supplied for instrument and general use, and in addition a 15 scfm, 600 psi system will be installed for operation of control rod drives.

A 15 kw gasoline-driven, emergency power supply will be available for operation of the control rod drive motors in the event of a commercial power failure.

F. Reactor Building

A plan and section view of the reactor building are shown in Figures 11 and 12. The building is 60 ft x 90 ft and is of concrete block construction. The main building housing the reactor and process equipment is 26 ft in height, while the wing building housing the process controls, auxiliary equip-

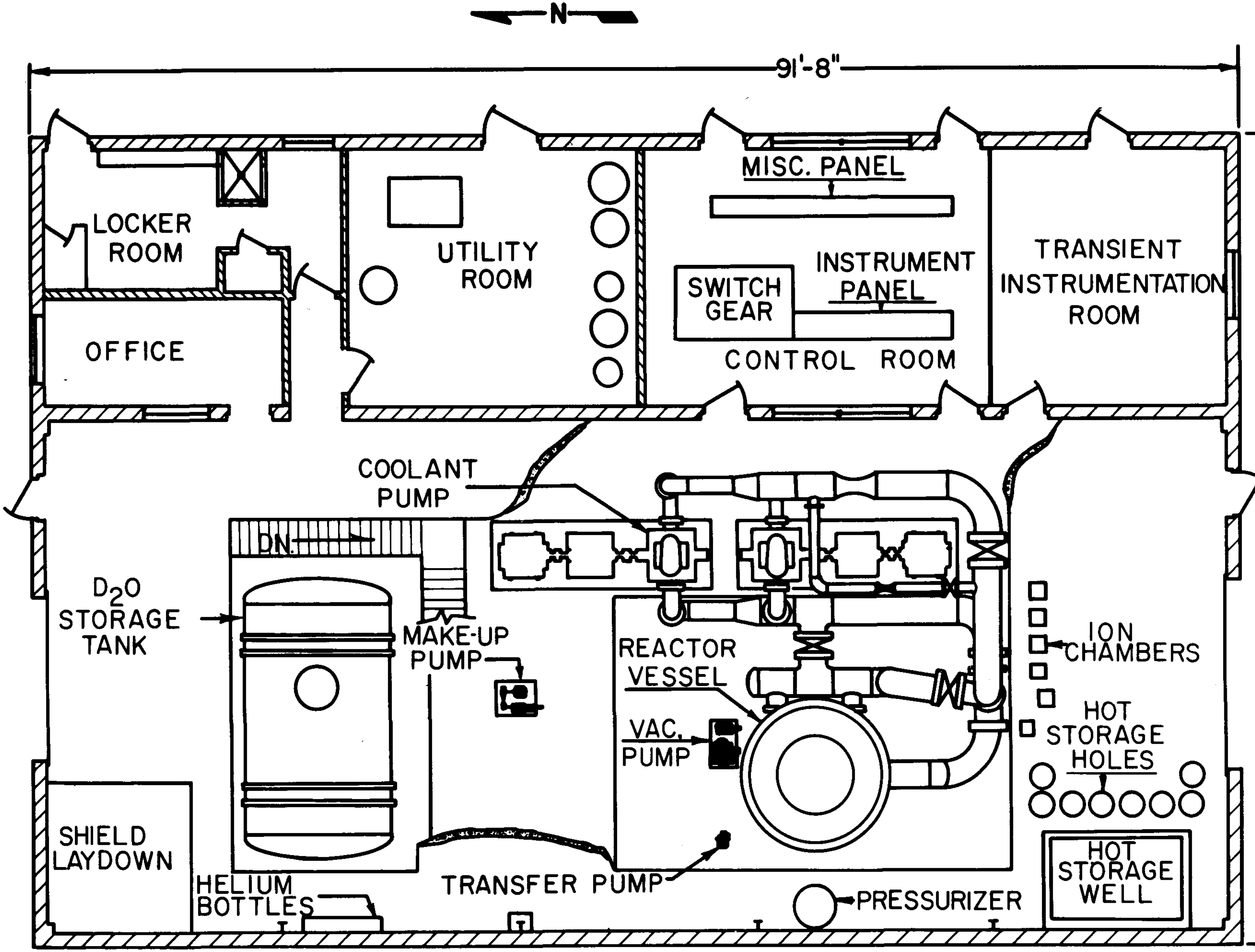


Figure 11

SPERT II

REACTOR BUILDING PLAN

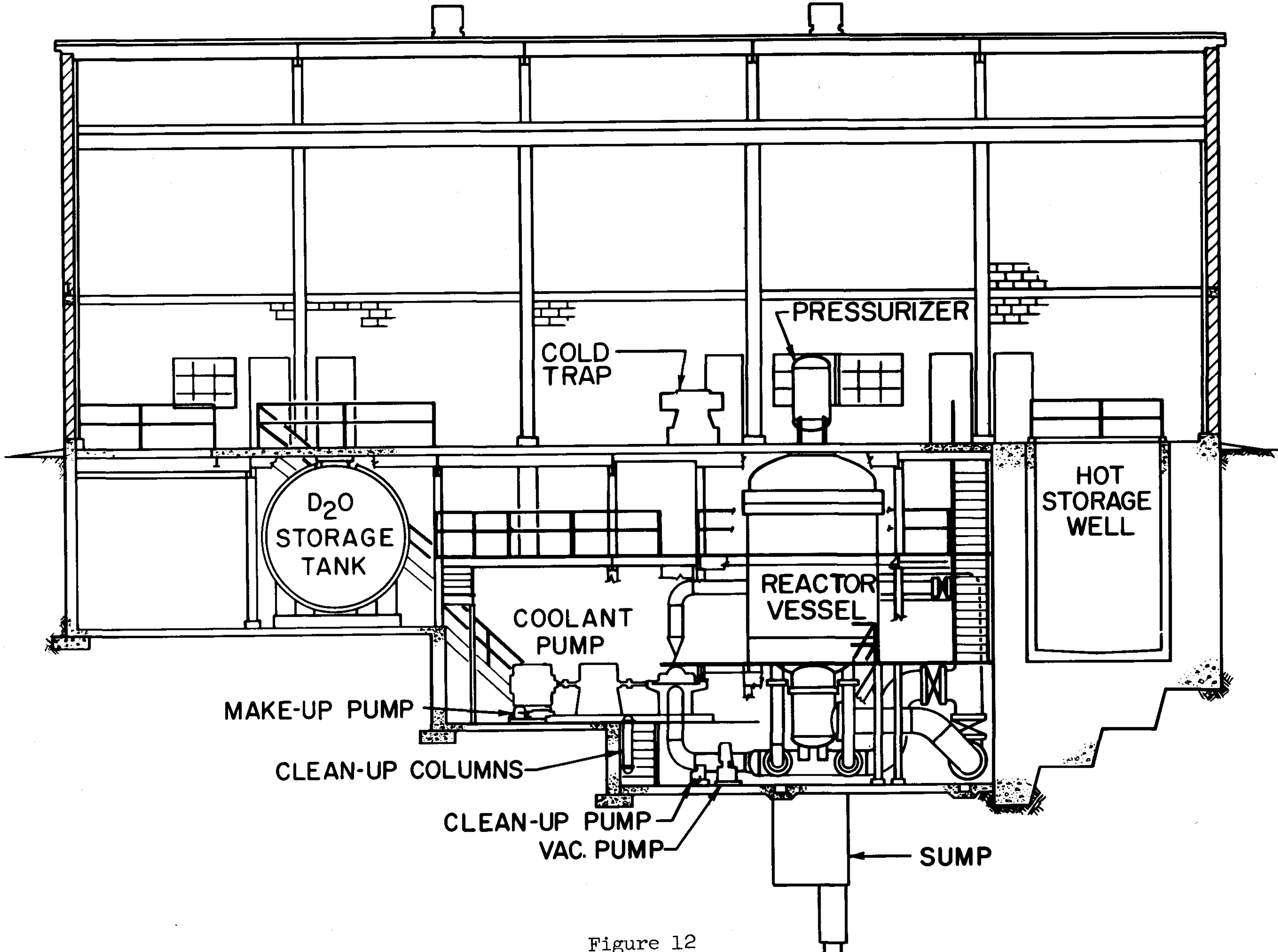


Figure 12

SPERT II

REACTOR BUILDING SECTION

ment, etc., is 10 ft in height. The main building is equipped with a 10-ton crane spanning the width of the building. The reactor vessel and process equipment are located in a leakproof pit. Access around the reactor vessel and elevated process equipment is provided by platforms at two elevations surrounding the vessel. Fuel storage facilities include a miniature canal or hot storage well and a number of dry storage holes.

G. Instrumentation

The experimental instrumentation for the Spert II facility is divided into three categories; namely, the process instrumentation, the reactor control instrumentation, and the transient instrumentation. The transient instrumentation, which is designed to measure and record the transient behavior of various system parameters, does not affect the operation or safety of the reactor facility and is not discussed in this report.

1. Process Instrumentation

The process instrumentation, as referred to here, consists of the instrumentation required for operation of the coolant system at various temperatures, pressures, and flow rates. A block diagram is shown in Figure 13 which indicates the process variables to be measured and/or controlled, the type instrument, and the physical location of the instruments. The system will normally be operated from the control center; however, for convenience in start-up and check-out, the control of essential process variables is also permissible from the reactor building. This, of course, necessitates control panels at the reactor building and the control center, and somewhat complicates the instrumentation because of the dual controls required. Because of the 3 000 ft distance separating the reactor and the control center, electronic transmission is required for all signals transmitted to the control center. The signals terminating at the control center will be transmitted via multi-conductor or coaxial cables as required, which will be laid on the surface of the ground. Cables transmitting power signals will be separated by a distance of at least 15 ft from cables transmitting instrument signals.

2. Reactor Control Instrumentation

Portions of the reactor operational instrumentation are currently in the design stage and therefore the system will be described by reference to a few of its features. A more complete description of the control system design is contained in Appendix A of this report.

a. Two boron-lined ionization chambers will be used for neutron level determination under operational condition (above $\sim$ 5 watts reactor power). The output of these will be displayed on two neutron level recorders mounted in the reactor console.

b. Two boron-lined, or BF_3, neutron pulse chambers will be used for low level neutron measurements (up to $\sim$ 5 watts reactor power). The output of these chambers will be displayed on scalers mounted in the reactor console; in addition, one of these will provide a count-rate meter indication on the console.

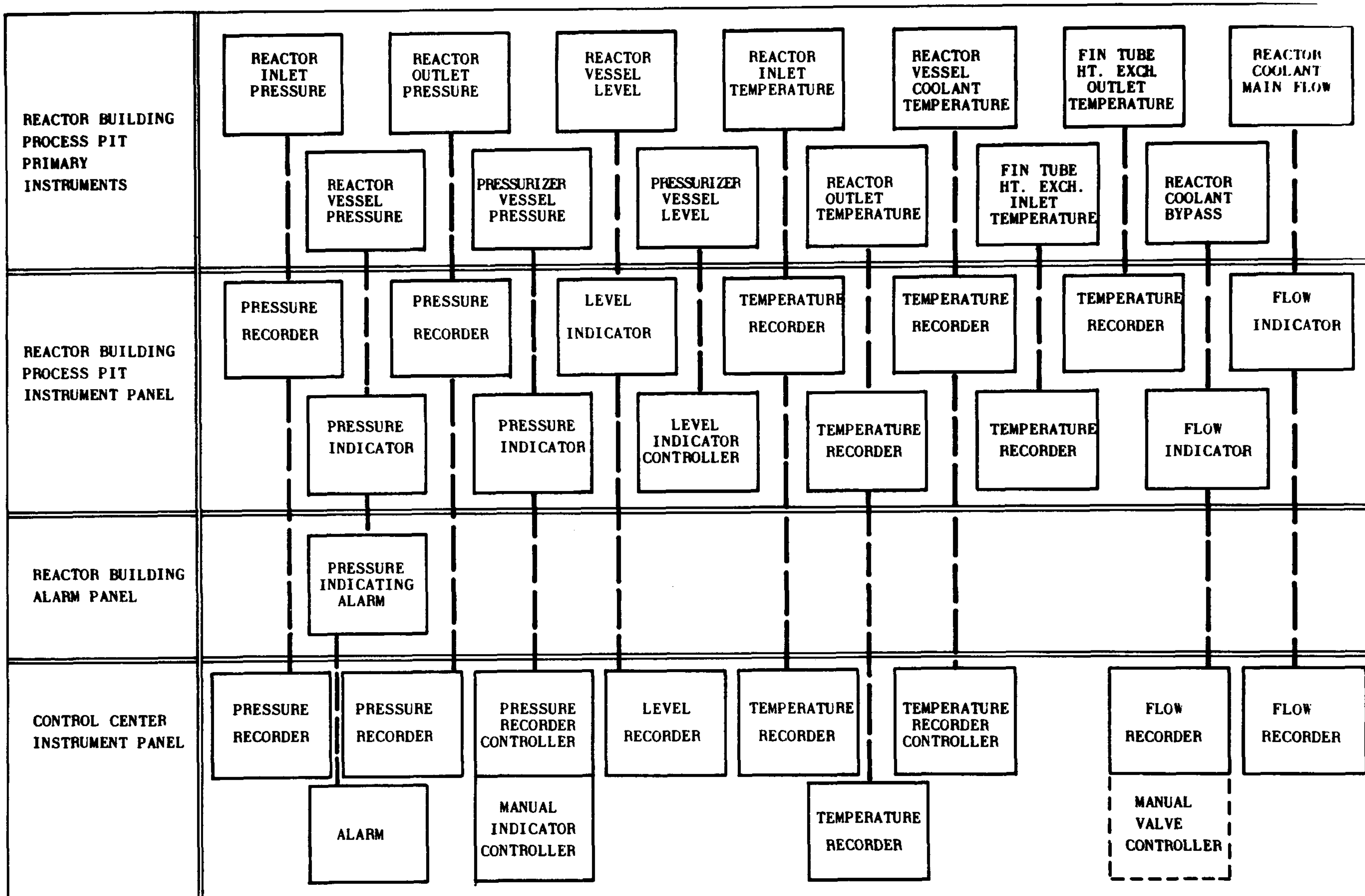


Figure 13

SPERT II SIMPLIFIED PROCESS INSTRUMENT BLOCK DIAGRAM

c. A closed circuit TV system, controlled from the console, will permit the operator to survey the reactor and process area of the reactor building.

d. Two sequence timers are provided, covering timing ranges of 30 sec and 5 min. These timers may be used to initiate tests, to control auxiliary equipment, and to terminate tests by means of scram or fast rundown of the control rods.

e. Indication is provided at the console of adequate scram air manifold pressure and transient rod top air pressure.

f. Control rod drive motor system.

(1) Each rod drive system is composed of a three-speed induction motor with an integral gear box.

(2) The control system permits selection of individual or group operation of all the control rods.

(3) The system provides for selection of rod speeds of about 19, 13 or 6.5 in/min.

(4) The system will automatically switch to high speed insertion upon command from an automatic rundown system.

(5) The system will provide automatic switching between the commercial power source and the standby power supply, in event of power failure.

g. The transient rod drive has essentially the same features as the control rod systems.

H. Shielding

During normal shutdown operation only the top plug will be removed from the vessel. No shielding is provided, since the water in the vessel serves as a shield around the core. During operation no personnel will be allowed in the area, therefore shielding under these conditions is not necessary.

I. Radioactive Materials Handling

1. Materials Handling

Two premises formed the basis for considering the radioactive materials handling problem in Spert II, namely (1) the equipment provided shall be as simple as possible commensurate with reasonable economy; and (2) since the Spert II reactor is not a continuous neutron or power producer requiring frequent fuel changes, unloading of the core will be necessary only to permit access to the reactor vessel for internal structure modification and repair, or to install special experimental equipment.

With these considerations in mind, unloading of fuel assemblies or other radioactive material from the reactor will be accomplished either by long handled grappling devices or by use of a transfer coffin.

Two types of "hot" storage facilities are provided at Spert II.

a. Eight 6-in. diameter, 16-ft deep dry storage holes, sunk in the floor near the southwest corner of the reactor building will accommodate small, moderately radioactive items. Lead shielding plugs will be used as closures.

b. A 6 ft x 10 ft x 16 ft deep hot storage well which is normally filled with water is located in the southwest corner of the building. Fuel assembly storage racks lined with cadmium or fabricated of boron steel will be provided in the bottom of the well, adjacent to the wall. The center of the well may be used for placement of the transfer coffin. A storage facility such as the "hot" well is necessary to provide some means for transferring radioactive assemblies to a coffin in order to permit removal of assemblies from the Spert area to other locations for chemical processing, re-use, etc.

Operation with heavy water is not expected to lead to a problem of excessive tritium activity. The heavy water will initially have a tritium concentration of ~ 6 μc/liter which is well below the human tolerance level. No build-up is expected since the total nvt accumulated during operation will be small.

2. Waste Disposal

The term waste disposal system as used herein refers to the piping, valves, pumps, etc., required to dispose of or contain liquid radioactive materials from the reactor vessel or coolant loop. To determine the necessary waste disposal system the major considerations are as follows:

a. Means must be provided for removal of the normally non-radioactive light water which must be drained periodically from the reactor vessel.

b. The reactor pit should contain heavy water and permit recovery in the event the reactor vessel should be ruptured.

c. Means of handling fission products in the primary coolant from a possible rupture and/or melting of fuel plates must be provided.

Normally the radioactivity of the light water will be low enough to permit discharge directly to a leaching pond. To this end, all building and process drains extend to a reactor building sump located in the reactor pit. A 75 gpm sump pump transfers the waste to a leaching pond located outside the reactor building. The sump pump is automatically controlled by the building sump liquid level with provisions for a manual start-stop station.

When utilizing heavy water, the sump pit will be drained and dried and the pump disconnected, thereby permitting the recovery of heavy water in the event of major spills. Normal cleanup of any radioactive contamination will be accomplished by processing the heavy water through the cleanup loop. Removal of contaminants will be accomplished in the ion-exchange columns which can then be disposed of.

No provisions have been included for the disposal or containment of highly radioactive water. Since the total nvt accumulated by any test core will be relatively small, there is little likelihood of a reactor accident which would seriously contaminate the reactor coolant. In the event of fuel plate rupture and/or meltdown the procedure followed would be:

a. Initially a period of containment while short-lived fission products from the initiating accident decays for the case of either light or heavy water coolant;

b. For light water coolant, dilution to tolerable levels and transfer to ICPP for disposal;

c. For heavy water coolant, circulation through the cleanup loop until a radioactivity level is attained whereby it can be returned to a processing plant for further purification and recovery.

IV. HAZARDS DISCUSSION

A. General

The objectives of the Spert II experimental program require that many tests be planned for, and carried out under, conditions which would normally be considered unsafe(7). It is obvious that if the test program is to yield the maximum of information, these tests must, and will, approach closely those circumstances leading to a maximum possible accident with the facility. Normally, a discussion of possible means of insertion of reactivity into the system would be called for in this type of report. However, this system is designed for studies on the response to such insertions, and such a discussion would be essentially an outline of the test program. The pressurized water experimental plant of the Spert II facility will be subjected to a large number of thermal cycles and thermal shocks. This type of operation may cause the system to fail, due to fatigue and resultant weakening of components, even in the absence of a nuclear incident. The nature of the tests to be conducted also requires that most of the automatic safeties and interlocks, which are normally provided on a system of this type, be by-passed. In lieu of such automatic devices, the philosophy of the design has been to provide no safety features which act independently of the operator, but rather to rely on administrative procedures to provide safety for the plant and personnel. This philosophy has been in use for Spert I for nearly three years and has proven itself a workable doctrine during that period.

The considerations stated above lead to the conclusion that, for the Spert II facility, the probability of occurrence of the maximum possible accident, while low, is not negligible in comparison with the probability of any other accident. This accident then, for Spert II, becomes also the maximum credible accident and is the one considered in this hazards discussion.

B. Accident Initiation

As has been previously stated, the test program for Spert II is one which deals with reactor safety. As a consequence of this program, many of the tests performed with Spert II would normally assume unsafe proportions in other locations and under different conditions. For the purposes of this discussion, accidents are classified as those events which occur in an unexpected manner, thereby creating dangerous conditions for personnel and/or resulting in equipment damage. Accidents will be further classified as to the circumstances surrounding their initiation.

1. Unpredictable Behavior

Careful extrapolation and interpretation of previous test results, consideration of equipment ratings, operating experience and systems inspection would not have predicted such behavior by the system (either nuclear or process). The results of such an accident, while unfortunate from a hazards standpoint, would constitute a discovery and hence provide a valuable contribution to the reactor safety program.

2. Unexpected Behavior

This type of accident would be caused by the behavior of the system (either nuclear or process) which was not expected due to faulty or careless predictions or actions. This type of accident must always be considered as a source of hazard due to the human factor. Caution and care in the interpretation of previously obtained results and carefully prepared and checked experimental procedures conducted by trained personnel are the only effective means for reducing the probability of such accidents.

3. Unscheduled Nuclear Excursions

An accident of this character would originate from premature or unplanned nuclear tests, or assemblies of supercritical systems which occur due to operator error or equipment malfunction. Rigid administrative controls, sound operating practices and procedures, proper mechanical designs and, where possible, system interlocks are the only effective means of reducing the probability of such an accident. The measures adopted at Spert II for this purpose are given in Appendix A and Appendix B. A discussion and outline of the major features of the Spert II control system is given in Appendix A. Appendix B contains a discussion of the operating rules and procedures.

C. System Response to an Initiating Incident

The Spert II facility has been designed to provide a planned means for the introduction of variable amounts of reactivity in various means or modes of insertion into the system under different operating conditions. The pressurized water experimental plant provides, by its very nature, that a minor nuclear incident or excursion may cause, or occur at the same time with, a failure of system components. The Spert II experimental program has been devised so that basic information on the response characteristics, under such operation conditions, may be obtained to provide a basis for response calculations and to secure information which may lead to the safer design of future reactors(7). Since a detailed analysis of system response to an incident requires a prior knowledge of response characteristics which is the item to be experimentally evaluated, and inasmuch as the type of accident postulated is the maximum one (destruction of the reactor vessel and building with the ensuing release of the core contents to the atmosphere) it is consequently unrealistic to make detailed calculations of the system response as a function of the initiating incident.

The assumed accident will then be that such a destructive failure of the system does occur. This postulation of an accident makes unneccessary, in this hazards evaluation, the need to consider the calculated nuclear parameters of each of the various core loadings and configurations which may be used. The nuclear properties will only influence the probability of an accident which is already assumed to be significant. The detailed procedures for the individual tests will be importantly influenced by the calculated and experimentally measured nuclear properties of the cores under investigation.

D. Consequences of Possible Accidents

While this discussion will be concerned primarily with the destructive incident mentioned above, several categories of consequences will exist, depending upon the circumstances surrounding the accident. These are categorized in five groups.

1. Local Radiation Exposure Only

This category does not properly follow from the destructive accident under discussion but is presented for completeness. An accidental assembly of a supercritical mass during fuel manipulations, or a non-explosive core melt-down, might produce such limited consequences. These accidents will be briefly discussed later in this report.

2. Reactor Area Damage Only

In this category, physical damage and nuclear and/or non-nuclear hazards are limited to the immediate reactor area.

3. Minor Site Hazards

In this category, serious radioactive hazards extend to the Spert control center area with minor danger to other NRTS areas, but no hazard exists to off-site areas.

4. Major Site Hazards

Serious radioactive hazards extend to downwind areas within the NRTS, but little or no danger exists to population centers outside the NRTS.

5. Hazards to the General Public.

The accident assumes catastrophic proportions with serious hazards to the general public.

The non-nuclear perils associated with malfunctions, misoperations or failures of the Spert II pressurized water system constitute a safety hazard which is of more immediate danger to operating personnel than the radioactive hazards. The water system has a volume of approximately 1700 ft^3. When operating at 300 psig and 400^oF this system has a potential energy release of approximately 10^7 BTU (10^4 Mw-sec). Using an energy release of 2.33×10^3 BTU per pound of TNT, the system has an energy equivalent of about 8,000 lbs of TNT.

For the purposes of this report, it is assumed that a serious accident will involve the sudden explosive release of this energy. No attempt is made to make detailed estimates of the resultant physical damage. Instead, it is further assumed that the reactor building would be demolished and personnel in the immediate area might be killed or seriously injured. Since no attempt has been made in the design to provide for an exterior containment vessel, the contents of the reactor core would be scattered in the reactor area but would constitute only a local radiation hazard unless the failure occurred under nuclear conditions which might allow melting or vaporization of the core material.

Any destructive incident is assumed to lead to category two consequences or worse. It will be shown in the next section that the maximum possible accident will lead to category three or four consequences depending on the meteorological conditions at the time of the incident. The particular type of accident initiation will not affect the consequences to the off-site personnel. However, accident initiations of the type previously listed as Unscheduled Nuclear Excursions and non-nuclear plant accidents of the Unpredictable Behavior type might lead to serious injury to operating personnel who might be in the reactor area at the time of occurrence. Category five consequences in which the hazard extends to the general public are shown not to exist for Spert II.

E. Discussion of the Maximum Possible Accident

The postulation of the maximum possible accident with the Spert II facility is one which includes a severe nuclear excursion during which a steam explosion occurs, demolishing the reactor and reactor building and releasing the accumulated fission product source in the reactor core to the atmosphere as debris or as a radioactive cloud.

The hazard will be confined to the blast area if no melting of the fuel occurs and fuel assembly fragments are the only radioactive items ejected. However, should the steam explosion be accompanied by a nuclear incident in which the fuel is melted or vaporized, consideration must be given to hazards associated with the formation of a radioactive cloud.

The extent of such hazard will be, to a large degree, influenced by the atmospheric conditions prevailing at the time of the incident and the fission product source present in the core at that time.

The Spert II facility, as previously stated, will not operate at appreciable powers for extended lengths of time due to the absence of heat removal facilities. The normal operation will consist only of transient tests lasting a few minutes at the maximum where reactivity is inserted as a step or ramp addition. Tests will, in general, be similar in severity to those conducted on Spert I except for the added risks due to flow, pressure, and temperature differences. The maximum energy input to the Spert I system for any one nuclear test has been of the order of 800 Mw-sec. Much of the accumulated fission product source will be due to the operating history of previous power excursions in which 10 to 20 Mw-sec were released in less than a second. It is then apparent that the major components of the fission product source following an excursion will be relatively short-lived products.

Estimation of the hazards for the maximum possible accident presumes a fission product source consisting of two parts. A long-lived source is assumed to account for an accumulated equilibrium source due to 800 Mw-sec of operation every day for an extended length of time. Practical considerations indicate that such assumptions are not realizable, and therefore this assumption is an upper limit for the long-lived source. The accident is postulated to have occurred at the end of a stability test where 800 Mw-sec of integrated power has been introduced into the system. The energy release of the actual destructive excursion will be estimated to have produced 100 to 200 Mw-sec which is the estimate of that released in the Borax destructive test[8]. Consequently the short-lived source is taken as that produced by an integrated power of 10^3 Mw-sec released instantaneously.

In order to investigate the influence of weather conditions on the consequences of an accident, the appraisal of effects has been made for average weather conditions, for inversion conditions, and for strong wind conditions.

The postulation of the maximum possible accident is summarized in the following paragraphs:

(1) The reactor vessel is ruptured and 100% of the melted or vaporized reactor core is released to the atmosphere in a steam cloud.

(2) The cloud is of negligible radius and is released at ground level. (Both of these assumptions will cause over-estimates of doses at points close to the incident).

(3) Three different sets of weather conditions are used in the estimates; average meteorological conditions, inversion conditions, and strong wind conditions. The meteorological constants used for these cases were taken from Table 8.2 of reference (9) and are shown in Table 1.

TABLE I

Suttons Meteorological Constants[9]

	Average Meteorological Conditions	Inversion Conditions	Strong Wind Conditions
Stability Index, n	0.25	0.50	0.25
Diffusion Parameter, C	0.20 $(\text{meters})^{1/8}$	0.05 $(\text{meters})^{1/4}$	0.20 $(\text{meters})^{1/8}$
Wind Velocity	3 meters/sec	1 meter/sec	15 meters/sec

Estimates have been made of the total integrated external gamma dosages which would be received by an individual standing directly in the path of the radioactive cloud as it passes overhead. Two cases were considered: The long-lived gamma source is treated as one produced by steady plant operation at 10 kw for an indefinite period of time (equivalent of 800 Mw-sec of operation per day for an indefinite period of time). This source is assumed to decay as $t^{-0.21}$. The short-lived gamma source is treated as that which would have been generated by a short-term power excursion with an energy release of 10^3 Mw-sec. This source is assumed to decay as $t^{-1.21}$. The nomograms due to Holland(10) have been used in making these estimates. These nomograms assume no fallout or rainout. The results for the above cases are shown in Figures 14 and 15 for distances up to 100 miles from the incident. Even under the worst weather conditions, these dosages would not constitute a serious hazard to areas other than the immediate vicinity of the Spert site. The nearest population center (Mud Lake-Terreton) is about 30 miles from the Spert site in the prevailing downwind direction. No other NRTS installations are located in this direction.

In order to estimate the internal doses from inhalation, it is assumed that the radioactive cloud falls immediately to the ground and thereafter moves along the ground with a hemispherical shape. The Sutton equation for a puff was used to describe the concentration of fission products as a Gaussian function of the distance measured from the cloud center. It is assumed that the cloud moves with constant velocity and the width of the distribution is a function of the distance the cloud has traveled. The doses shown in Figures 16, 17, 18, and 19 are estimated for an individual standing directly in the path of the cloud during its passage. The approximate dosages to bone and to the thyroid have been estimated for infinite time after inhalation. The constants and equations used in arriving at these estimates are given in Appendix C of this report. Two sets of conditions have been used here in estimating the dosages. The dosages labeled "Steady Power" in Figures 16 and 17 assume 100% fission product release following operation at 10 kw for an indefinite period of time. Those labeled "Power Excursion" in Figures 18 and 19 assume 100% fission product release following operation totaling 10^3 Mw-sec. Except for the inversion condition, the dosages for distances greater than 5 miles are substantially less than the maximum "once in a lifetime" exposure (which is assumed to be 25 rad). The large inhaled dosages shown for inversion conditions indicate the necessity for meterological control in cases where there is a reasonable chance for the occurrence of such an incident.

It should be stressed that every attempt has been made in these calculations to postulate the worst possible cases. The assumption in the above calculations of release at ground level as a point source results in an over-estimate of the dosages at short distances from the incident. Almost certainly the cloud would be of at least 10 meters in diameter and would rise rapidly since it would be largely a steam cloud. The assumptions of 100% fission product release and an impracticably large (for Spert II) long-lived source term are further over-estimates. A more reasonable estimate of fission product release and long-lived source term would indicate that the estimated dosages would be reduced by two orders of magnitude.

In view of the isolated location of the incident within the NRTS and the absence of installations in the prevailing downwind direction, the effects of fallout, rainout and specific isotope uptake have not been considered in drawing conclusions as to the consequences of the incident.

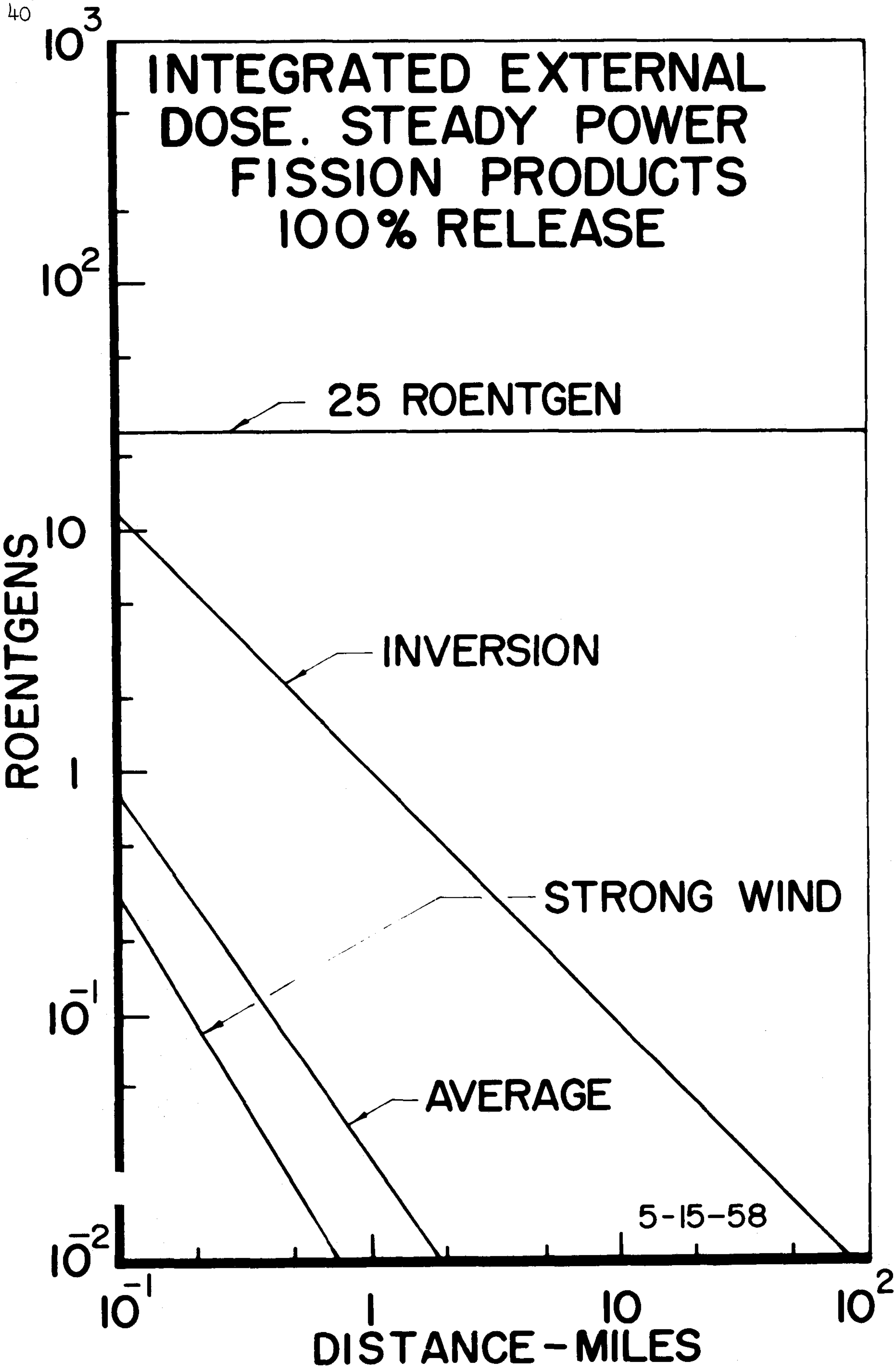

INTEGRATED EXTERNAL DOSE. STEADY POWER FISSION PRODUCTS 100% RELEASE
25 ROENTGEN
INVERSION
STRONG WIND
AVERAGE
5-15-58
ROENTGENS
10^3
10^2
10
1
10^{-1}
10^{-2}
10^{-1}
1
10
10^2
DISTANCE-MILES

Figure 14

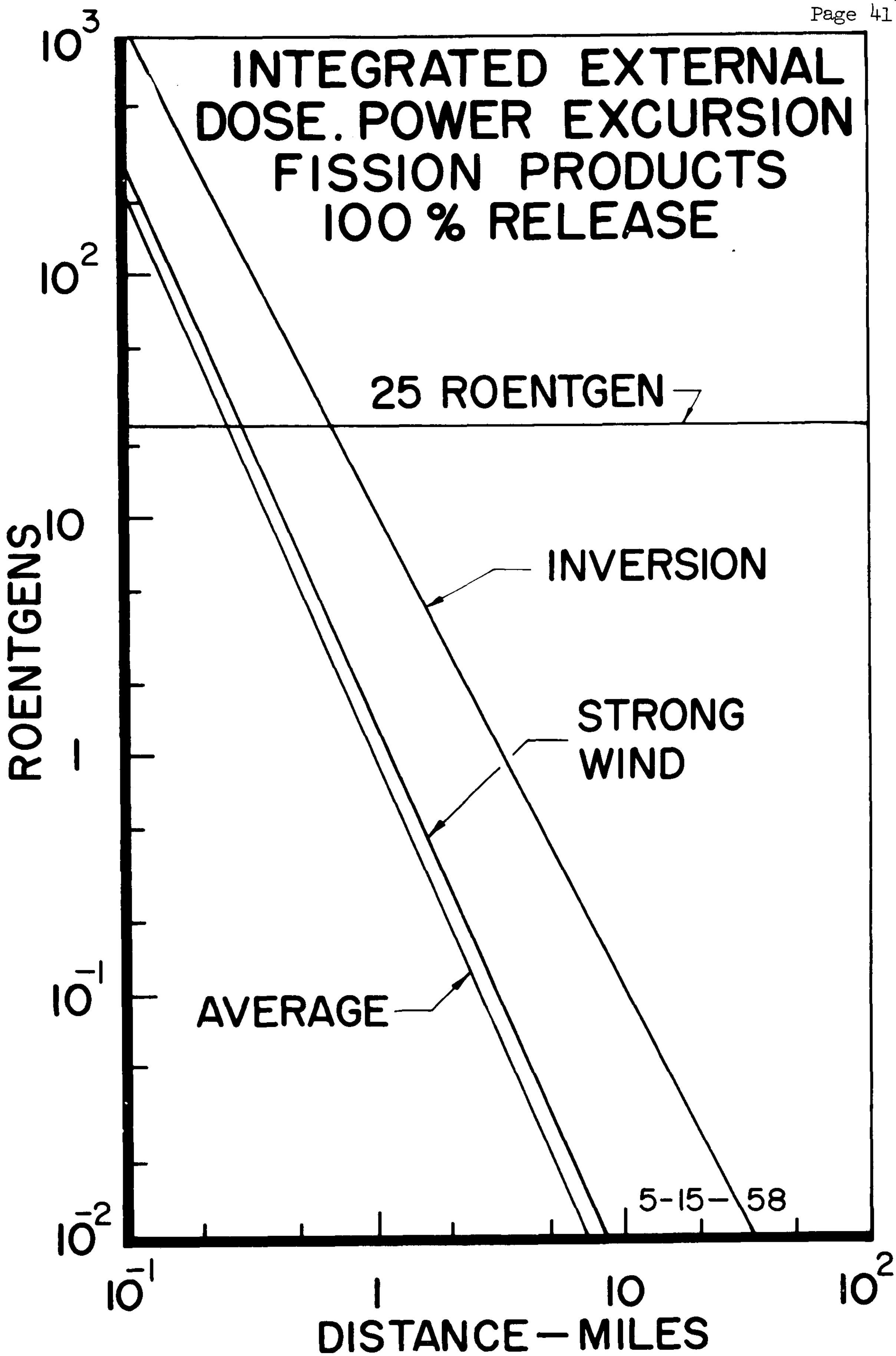

INTEGRATED EXTERNAL DOSE. POWER EXCURSION FISSION PRODUCTS 100 % RELEASE
25 ROENTGEN
INVERSION
STRONG WIND
AVERAGE
5-15-58
ROENTGENS
10^3
10^2
10
1
10^{-1}
10^{-2}
10^{-1}
1
10
10^2
DISTANCE — MILES

Figure 15

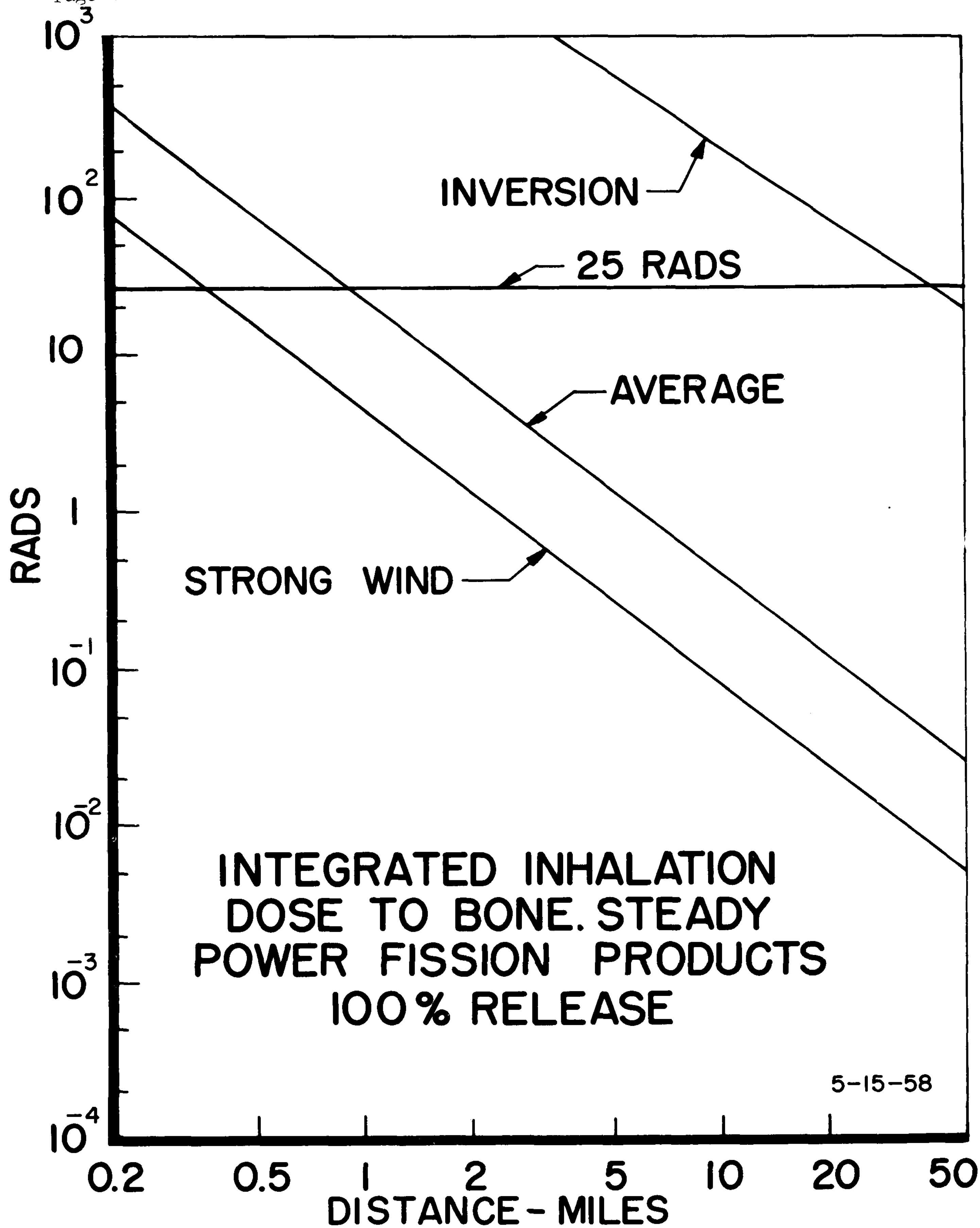

10^3
10^2
10
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
RADS
INVERSION
25 RADS
AVERAGE
STRONG WIND
INTEGRATED INHALATION
DOSE TO BONE. STEADY
POWER FISSION PRODUCTS
100% RELEASE
5-15-58
0.2
0.5
1
2
5
10
20
50
DISTANCE - MILES

Figure 16

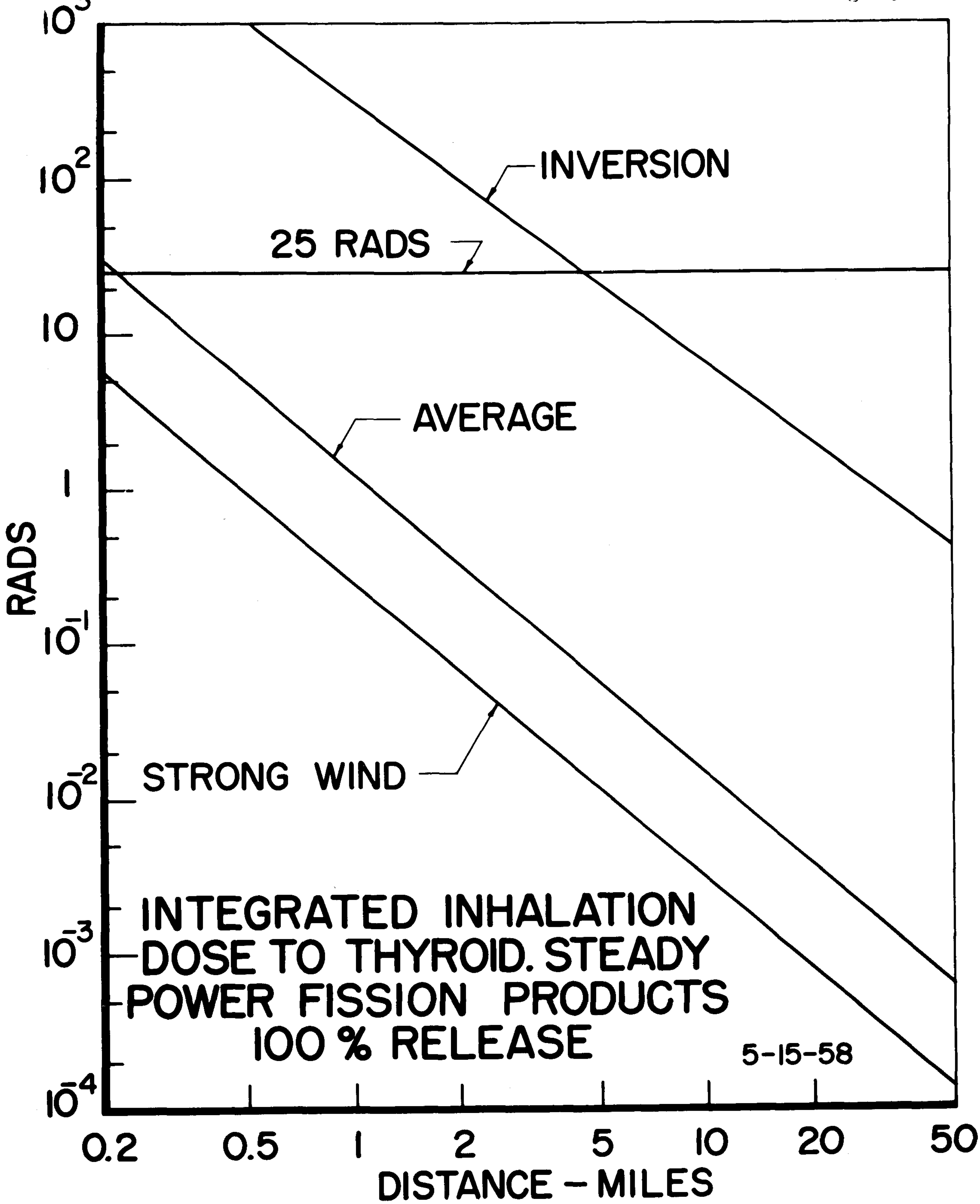

RADS
10^3
10^2
10
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
INVERSION
25 RADS
AVERAGE
STRONG WIND
INTEGRATED INHALATION DOSE TO THYROID. STEADY POWER FISSION PRODUCTS 100 % RELEASE
5-15-58
0.2
0.5
1
2
5
10
20
50
DISTANCE – MILES

Figure 17

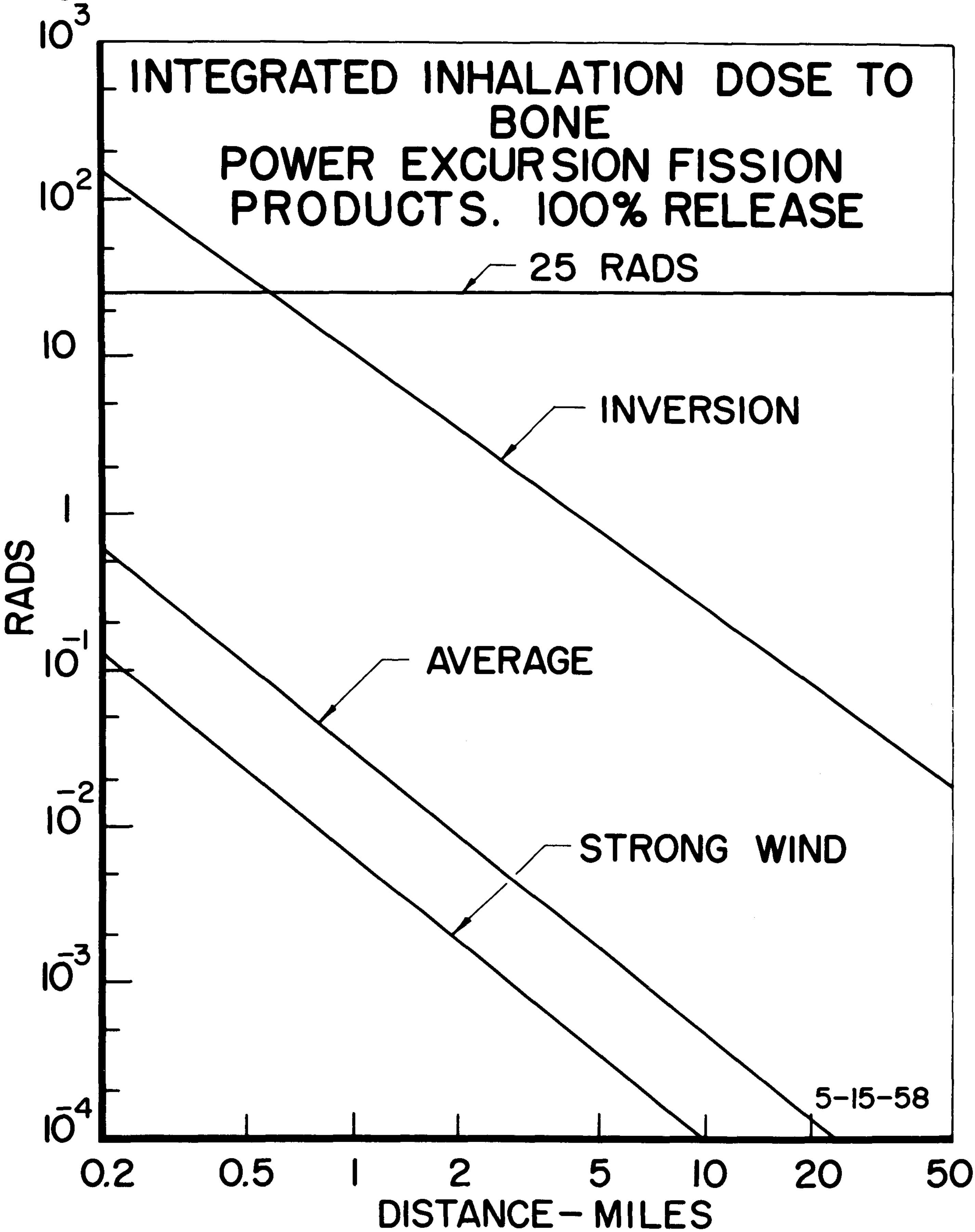

INTEGRATED INHALATION DOSE TO BONE
POWER EXCURSION FISSION PRODUCTS. 100% RELEASE
25 RADS
INVERSION
AVERAGE
STRONG WIND
5-15-58
RADS
10^3
10^2
10
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
0.2
0.5
1
2
5
10
20
50
DISTANCE - MILES

Figure 18

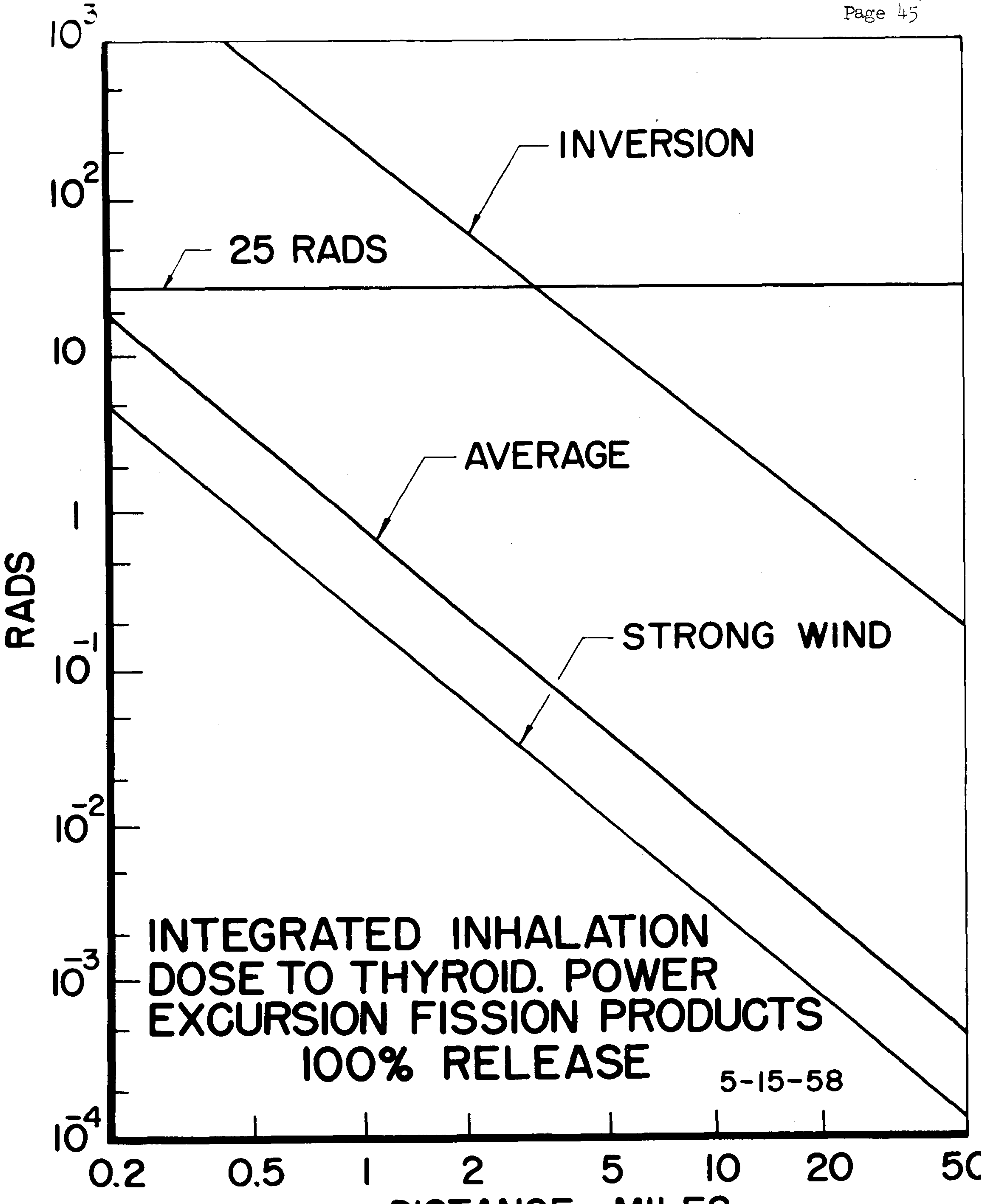

INVERSION
25 RADS
AVERAGE
STRONG WIND
INTEGRATED INHALATION DOSE TO THYROID. POWER EXCURSION FISSION PRODUCTS 100% RELEASE
5-15-58
RADS
10^3
10^2
10
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
0.2
0.5
1
2
5
10
20
50
DISTANCE — MILES

Figure 19

These would be important only to those persons in the open within a few miles downwind of the incident.

This treatment leads to the conclusion that meteorological control is mandatory for an experiment where there is a reasonable chance of an explosive release. In particular, no such experiments will be conducted when conditions are such that it is possible for other NRTS installations to be downwind of the Spert II facility.

The operating procedures outlined in Appendix B should prevent any accumulation of circumstances leading to the accident postulated here. However, a small but finite possibility will always exist for such an incident. In the event of such an occurrence the normal NRTS disaster plan would be placed in effect and the AEC Idaho Operations Office would be alerted immediately in order to aid in the evaluation of hazards and to direct the precautionary effort.

The conclusion drawn here is, that even under the most unfavorable conditions, the isolation provided in the site location is adequate to protect the general public. Meteorological control will aid in providing additional protection to the NRTS areas by decreasing the probability of accidents while unfavorable weather conditions exist. The control system features and administrative procedures, outlined in the appendix, are designed to provide additional protection to Spert personnel by reducing the probability of unscheduled tests with personnel in the vicinity of the reactor.

F. Criticality Accidents

Serious radiation exposures to operating personnel could result from an incident in which a supercritical assembly is accidentally achieved during loading of the reactor core. Kinetic testing at maximum system temperature will require a reactor loading which provides about 10% flexible reactivity with light water as a moderator and about 25% flexible reactivity with heavy water as a moderator at room temperature. All operations on the reactor vessel with the top plug, and hence the control system, removed must be classified as critical experiments and administrative control procedures as outlined in Appendix B will be invoked in order to reduce the probability of such accidents.

G. Hazards of Non-Explosive Release of Fission Products to the Reactor System

There exists a very real probability of partial melting or cladding failure of the reactor fuel assemblies with a resultant release of fission products to the reactor primary coolant system. The system will be initially leak tight in view of the pressurized water system and utilization of heavy water. Any violation of the system integrity will become a reason for immediate shutdown for repairs.

Should such a leak develop simultaneously with the fission products release it might lead to the release of fission products into the air or the reactor building.

If the primary system is intact following such a fission product release the only hazard of immediate consequence would be of a local nature due to the unshielded piping and reactor vessel. Normal health physics procedure for re-entry to the reactor building following operation will provide protection against such exposures. The reactor pit may be flooded with potable water by remotely operated valves to provide additional shielding, should this be necessary.

The next point of hazard will occur at the time that the system pressure is reduced. Air sampling and monitoring systems with remote alarms will be provided at the helium pressurizing system vents. These alarms will be monitored during system depressurization to assure that gaseous products are not released to the atmosphere. If such products are existent, system pressure relief will be done only under health physics control and with safe meteorological conditions. Samples will be taken and checked for fission product activity prior to any dumping or transfer of primary system water.

Failure of the primary system which results in leakage to the building will, of course, be readily apparent if operation is at greater than atmospheric pressure. Otherwise reliance must be placed on continuous air monitors in the building and on health physics surveys to detect the presence of a hazard.

In summary, the hazards associated with the release of fission products into the primary water system are largely concerned with the detection of such a condition. Once a condition has been detected suitable measures can be taken to minimize the consequences and any additional release to the atmosphere. Hazards to operating personnel are slight because of the remote operation of the reactor. The nature of experiments with the Spert II facility, while increasing the probability of such occurrences, does reduce the severity of consequences due to the fact, as mentioned previously, that the fission product concentration will be quite low. In any case, these hazards will be orders of magnitude less than those described for the maximum possible accident.

V. CONCLUSIONS

Because of the uncertainties surrounding the possible failure of the Spert II pressurized water system, the probability of an accident in which the reactor system and building are destroyed is significant. This is so whether or not the accident is a result of a nuclear incident. In cases where no significant fission product source is present in the reactor at the time of the incident, the hazards would be limited to the immediate reactor area. The planned investigations of the response of the system to what are considered unsafe operations also increases the probability of a destructive incident. It must be remembered, however, that it is the rupture of the vessel along with the core meltdown which provides the serious hazard, since core meltdown can occur without vessel rupture. The discussion has been concerned with the "maximum possible accident", one in which the reactor system and building are completely destroyed and the reactor contents are liberated to the atmosphere. Thus the nuclear and physical constants do not enter the evaluation of hazards presented here and the discussion is applicable to all core loadings which might be used in Spert II.

It is concluded from the discussion that under favorable meteorolgoical conditions radioactive hazards will be serious only in the immediate vicinity of the reactor with minor hazards to downwind areas a few miles away. Under unfavorable weather conditions serious radioactive hazards may extend as far as five miles downwind but no major hazards will be inflicted on the general public.

Procedures have been outlined to reduce the probability of such accidents and to insure favorable meteorological conditions for those tests which have moderate or unknown probability of accident, or which involve sizeable concentrations of fission products in the reactor. Inasmuch as the radioactive hazards arise primarily from the short-lived products produced during the actual experiments in which the accident might occur, previous nuclear operating history of the core in the Spert II reactor will contribute only insignificantly to the hazards.

The site location, control design, and operating procedures proposed permit operation of the Spert II facility without hazards to the general public.

VI. ACKNOWLEDGEMENT

The author wishes to express his thanks to F. Schroeder for his many helpful suggestions during the writing of this report. Many of the definitions, operating procedures, and methods of hazard evaluation are based on those used by him in the Spert III Hazards Summary Report.[11]

VII. REFERENCES

1. Leyse, C. F. and Wilson, T. R.; Proposed Spert Reactor Sites, PPC-249 (August, 1955).

2. McLain, S., and Winkleblack, B. K.; Classified Report, ANL-SM-236.

3. Humphrey, P. A. and Wilkins, E. M.; The Climatology of Stack Gas Diffusion at the National Reactor Testing Station, IDO-10020 (March, 1952).

4. Leyse, C. F., McPherson, R. S. and Wilson, T. R.; Conceptual Proposal Spert II Reactor Facility, PPC-326, (January, 1956).

5. Wilson, T. R.; The Spert II Reactor Facility Preliminary Design Report IDO-16386 (October, 1957).

6. Montgomery, C. R., Norberg, J. A. and Wilson, T. R.; Summary of the Spert I, -II, -III Reactor Facilities, IDO-16418 (November, 1957).

7. Nyer, W. E. and Forbes, S. G.; Spert Program Review, IDO-16415 (September, 1957).

8. Dietrich, J. R.; Experimental Investigations of the Self-Limitation of Power During Reactivity Transients in a Sub-Cooled Water-Moderated Reactor, AECD-3668 (August, 1955).

9. U. S. Department of Commerce, Weather Bureau, Meteorology and Atomic Energy, AECU-3066 (July, 1955).

10. Figures 8.1 and 8.3 of reference 9.

11. Schroeder, F.; Spert III Hazards Summary Report, IDO-16425 (January, 1958).

APPENDIX A - CONTROL SYSTEM DESIGN

I. INTRODUCTION

As previously mentioned, the Spert II control system will contain no automatic safety circuits. The circuitry is, however, designed to be fail-safe, that is, it will fail in such a way as to protect the reactor, and does contain numerous interlocks, both electrical and mechanical, to prevent some unwise procedures on the part of the operator and to reduce the possibility of unplanned reactor excursions. It is the purpose of this appendix to describe in some detail the philosophy used in this design. The control system design and philosophy are essentially identical with that used for the Spert III reactor and will have been tested by nearly six months operation under more trying conditions of temperature and pressure before its use in the Spert II system.

The control rod drive basically consists of a cylinder, a piston, a shock absorber, a drive screw and three-speed induction motor. No direct connection exists between the piston, which is joined to the control rod, and the drive screw and motor. Air pressure on the bottom side of the piston maintains the piston in contact with the drive screw, thus permitting positioning of the control rod with the drive motor and screw. Air at slightly less pressure, which is sufficient to drive the rod in against reactor vessel pressure, is maintained on the top side of the piston. Rapid insertion of the rods, or "scram", is accomplished by releasing the air on the bottom of the piston. An air reservoir of sufficient capacity to drive in and hold the rod in the down, or safe, position is built into the upper cylinder.

Limit switches are provided which stop the drive screw travel at the lower limit position and at 25 and 43 in. withdrawn ("upper limit" and "top limit" positions). Similar switches stop all the drives below the 1-in. withdrawn position unless the drive motors are in slow speed. This restriction is provided to protect the drive against over-running the lower limit switch.

Spring-locked latches are provided to hold down the control rod pistons once they are seated. A key-switch must be operated to unlatch the control rods and permit withdrawal. The transient rod drive is similar to the control rod drives except that in this case a latch is provided to secure the rod piston, and hence the rod itself, to the rod drive screw until unlocked by a key-switch. Inasmuch as the transient rod is dropped to insert reactivity and initiate an excursion, transient rod top air is applied only in preparation for a test.

In order to remove the top plug of the reactor pressure vessel on which the drive system is mounted, the rod piston assemblies are uncoupled from the actual rods by handling tools and the drive screws and pistons are withdrawn to their top limit position in order to provide maximum clearance when transporting the top plug to its dry dock location.

Provisions are made for test operation of the drive system in dry dock.

II. SAFETY FEATURES

Probably one of the least apparent and most dangerous of accidents which could conceivably occur is the possibility of the control elements being forced out of the reactor core by vessel pressure, without operator action. This is normally prevented by the procedural restriction that the rod drive screws be down and in contact with the seated rods before any pressure may be applied to the system. The rod drive brakes are designed to provide adequate protection against any motion of the rods under these conditions. In addition a "No-Back" clutch is provided in the shaft between the gear motor and the drive screw which prevents overhauling loads from driving the motor.

Additional protection is provided in the form of the mechanical latches which require the operation of a key-switch before the rods themselves may be raised from the seat position. These latches have been designed to hold the control rods in their seated position against full vessel pressure. When the reactor is in operation, control circuitry provides that loss of contact between any control rod drive screw and its rod piston will initiate full speed insertion of all drives. In addition, the action of the latches is such as to latch down any dropped rod immediately, thus providing hold-down until the drive has re-established contact with the piston. Finally, the scram air reservoir on the control rods is of sufficient capacity to hold the rods in against vessel pressure.

A regulating system is provided to maintain sufficient scram air on the rod pistons for any vessel pressure. Failure of this system causes automatic full speed insertion of the control rod drives. Since successful scram depends upon rapid exhaust of the hold air, four normally open exhaust valves are provided to vent the hold air. The action of any one valve is sufficient for scram and each of the four valves will be tested individually prior to every operation of the reactor. Failure of any scram component or fast run-down circuit to operate satisfactorily when tested will require that no operation be permitted until repairs have been made. Normal fail-safe practices have been followed in all circuit and plant design.

A standby gasoline-electric power source for the control system will be in operation "on the line" at all times when the reactor is in operation. A bus transfer unit provides switchover to the standby source in the event of main power failure. Failure of either this standby power or the main power source will initiate a scram and full speed rundown of the control rod drives. In the improbable event of simultaneous failure of both power sources, the lock-down latches and the scram air supply are each sufficient to hold down the scrammed control rods. The rod drives would, of course, remain withdrawn until restoration of power. Terminals are provided at the control console to permit a continuity check of the rod seat switches in the absence of power. Scram air would probably remain adequate to hold down the rods for a period of the order of hours without power to the compressor. In the event that commercial power is not restored in a few minutes, an attempt will be made to repair the standby source. However, restoration of commercial power is necessary before the control power is provided to shut down the plant in the normal manner.

Since an accidental drop of the withdrawn transient rod could initiate an unplanned power excursion, a mechanical latch mechanism secures the rod piston to the drive screw. This mechanism is controlled by a key-switch and must be latched in order to perform any operations upon the system other than scram, control rod insertion, or programmed initiation of a transient. As an added precaution, no top air may be applied except as a part of the arming sequence.

III. CONTROL SYSTEM DESIGN PHILOSOPHY

The following outline presents the principal features of the control system design philosophy.

A. Requirements for Control Power On

1. Commercial main power on.
2. All reactor-drive power plugs in place. (Loss of this interlock after control power is on provides an alarm signal to the operator but does not turn the power off.)
3. Main power key-switch on.

B. Requirements for Scram Reset

1. All control rod drives at their lower limit positions.
2. No scram signal.
3. Operation of scram reset button.

C. Requirements for Control Rod Withdrawal

1. All control rod drives in contact with the rod pistons.
2. Control rod scram air okay (indicated by no alarm from scram air pressure switch).
3. Standby power supply in operation.
4. Transient rod drive in contact with transient rod piston.
5. Arming relay not energized.
6. Transient rod piston latched to drive (indicated by a switch on latch mechanism).
7. Transient rod top air off (indicated by pressure switch in top air line).
8. No control rod insertion signal (control rod insert relay not energized).

9. No control rod fast rundown signal.

10. Operation of spring-return control rod withdrawal switch.

11. Control rod shock absorbers extended on each rod which is withdrawn more than 6-in.

12. Individual rod drives must also satisfy the following conditions in order for that drive to withdraw.

    a. Drive selected by operator (each rod drive has an independent on-off switch).

    b. No control rod insertion signal (individual control rod insert motor contactor not energized).

    c. Drive not at upper limit position (about 25 in. withdrawn).

    d. Drive not at top limit position (about 43 in. withdrawn).

    e. Emergency stop switch not actuated by operator.

Note I: Requirements 1, 2, 3, 9, 11 and 12 may be bypassed by the upper limit bypass key-switch during special drive testing and vessel top plug removal.

Note II: Control rod lock-down latches are provided which automatically lock the control rod pistons in their seated position. In order to begin withdrawal from this position all drives must be in contact with the pistons and the operator must operate the spring return control rod latch key-switch until the selected drives are clear of the latch.

D. Requirements for Control Rod Manual Insertion

1. Operation of control rod insert switch.

2. Each rod drive must satisfy the following conditions in order for that drive to insert.

    a. Drive selected by operator.

    b. Drive motor speed selector must be in slow speed position if drive is at or below the 1-in. withdrawn position (speed selector controls all control rod drive motors as a group).

    c. Drive not at lower limit.

    d. Emergency stop switch not actuated by operator.

E. Control Rod Fast Rundown

Fast rundown overrides manual withdrawal and insertion. It is an automatic fast speed insertion of all control rod drives until they reach the 1-in. withdrawn positions.

1. Fast rundown is initiated by any of the following conditions:

   a. Operation of the fast rundown button.

   b. Reactor scram.

   c. Alarm from control rod scram air pressure switch (low or high pressure).

   d. Loss of contact between any control rod drive and its rod piston.

   e. Failure of standby power supply.

   f. Failure of main power supply (indicated by transfer of power bus to standby power supply).

2. When all the drives have reached the 1-in. withdrawn position, the operator must select slow speed and manually insert the drives to contact with the rod pistons which resets the fast rundown signal.

3. A spring-return emergency stop switch may be operated on any rod drive motor to stop its insertion in the case of a malfunction.

Note I: The fast rundown may be prevented during vessel top plug removal by actuation of the upper limit bypass key-switch.

F. Scram

1. The reactor is scrammed by the operation of variously located manual scram buttons in the reactor control room or reactor building.

2. A reactor scram may also be programmed into the transient sequence timer if so desired.

3. Power failure will result in a scram by nature of the fail-safe circuitry.

G. Requirements for Transient Rod Withdrawal

1. All control rod drives in contact with rod pistons.

2. Control rod scram air okay.

3. Transient rod arming relay not energized.

4. Transient rod holding air on (indicated by a pressure switch in the hold air line).

5. Transient rod drive in contact with the rod piston.

6. Transient rod top air off (indicated by a pressure switch in the top air line).

7. Transient rod piston latched to drive (indicated by a switch on the latch mechanism).

8. No transient rod fast rundown in progress.

9. No transient rod insertion signal (transient rod insert relay not energized and insert motor contactor not energized).

10. Drive not at upper limit position (about 43 in. withdrawn).

11. Operation of the transient rod withdraw switch (spring-return).

12. Transient rod shock absorber extended if drive is above 6 in. withdrawn.

H. Requirements for Transient Rod Manual Insertion

1. All control rod drives in contact with the rod pistons.

2. Control rod scram air okay.

3. Transient rod holding air on.

4. Transient rod arming relay not energized.

5. Drive not at lower limit positions.

6. Operation of transient rod insert switch (spring-return).

7. Transient rod drive motor speed selector must be in slow speed if drive is at or below the 1-in. withdrawn position.

Note I: Requirement one is bypassed if all control rod pistons are in the seated positions.

Note II: Requirement two is bypassed if all control rod drives are in the lower limit position.

Note III: Requirement three is bypassed if all control rod drives are in the lower limit position, or if the transient rod drive is below 1-in. withdrawn.

I. Transient Rod Fast Rundown

Automatic fast speed insertion of the transient rod drive to the 1-in. withdrawn position following a drop of the transient rod.

1. Transient rod fast rundown is initiated by loss of contact between the drive and the transient rod piston with the requirement that the transient rod piston is in the seated position.

2. When the drive has reached the 1-in. withdrawn position, the operator must select slow speed and manually insert the drive to contact with the piston which will reset the rundown signal. The operator must insert the drive to the lower limit position in order to complete the firing sequence.

3. Transient rod fast rundown may be stopped in the event of malfunction by operation of the transient rod emergency stop switch.

J. Arming Sequence

1. The following conditions must be satisfied:

   a. Control rod scram air okay;

   b. Standby power supply in operation;

   c. All control rod drives in contact with the rod pistons;

   d. No control rod fast rundown in progress;

   e. Transient rod hold air on (as indicated by a pressure switch in air line);

   f. Transient rod shock absorber extended;

   g. Transient rod drive in contact with the rod piston.

2. The operator actuates the arming key-switch (spring-return) which causes the following:

   a. The arming relay energizes and seals on through the normally closed disarm switch as long as the conditions in one are maintained;

   b. The transient rod top air turns on (indicated by a pressure switch in air line).

3. The operator now operates the unlock key-switch (spring-return) which, if the top air is on and arming relay is energized, causes the transient rod latch control relay to unlatch the piston.

4. The three conditions energizing the armed relay which turns on the armed signal light are:

   a. Transient rod hold air on;

   b. Top air on;

   c. Piston unlatched.

5. Any of these three conditions cause the disarmed signal light to go out:

   a. Arming relay energized;

   b. Top air on;

   c. Piston latch unlocked.

6. The sequence may be stopped and reversed to the disarmed conditions at any time by operation of the disarm push button.

K. Transient Rod Firing Procedure

1. In the armed condition (both arming relay and armed relay energized) the operator selects and sets up the particular transient sequence timer which is to be used for the test and turns on the timer fire switch and the timer switches for any other programmed functions desired (recording oscillographs, scram, rundown, etc.).

2. To initiate the experiment, the operator actuates the timer start key-switch (spring-return).

3. At a pre-set time later, the timer fire contact energizes the fire relay which seals on until the transient rod drive reaches its lower limit position. The fire relay initiates the following sequence of action:

   a. The transient rod hold air fill valve closes and the hold air exhaust valves open.

   b. The top air control relay is sealed on by the fire relay, thus maintaining top air until the drive reaches lower limit.

   c. The latch control relay seals on, keeping the latch mechanism in the unlocked position until contact is regained with the piston and the drive reaches lower limit position.

4. The loss of contact between the transient rod drive and the rod piston causes the following events to happen:

   a. Along with the loss of transient rod holding air it causes the arming relay and the armed relay to de-energize, which removes the armed light indication.

b. The top air control relay and hence the transient rod top air seals on until contact is regained, independently of the action of the fire relay in 3b above.

c. The transient rod fast rundown is initiated as soon as the rod is seated after loss of contact and continues until the 1-in. withdrawn position is reached.

5. Arrival of the transient rod piston at its seated position energizes the fired light.

6. When the transient rod fast rundown stops the drive at the 1-in. withdrawn position the operator selects slow speed and inserts the drive to obtain contact with the piston and to reach the drive lower limit position.

a. The latch control relay is de-energized which latches the piston to the drive.

b. The fire relay is de-energized which causes the following:

(1) The transient rod hold air exhaust valve closes and the hold air fill valve opens;

(2) The top air control relay de-energizes, closing the top air fill valve and opening the top air exhaust valve;

(3) The fired light is extinguished, indicating the end of the fire sequence.

Note I: The operator may stop the fire sequence at any point prior to the actual firing by operating the timer-stop push button.

M. "Reactor On" Alarms

A set of warning lights at the reactor building are lighted and a set of warning horns at the reactor building sounds for 30 sec whenever any control rod or the transient rod drive is raised above the lower limit position. These alarms are bypassed whenever the upper limit bypass key-switch is operated.

When the reactor control drive units are in dry dock, a similar but separate set of alarms in the vicinity of the dry dock is actuated under the above conditions.

N. Plant Operating Requirements

No electrical interlocks are provided between the plant functions and the nuclear control system. Administrative control will be exercised to insure consideration of the nuclear implications of process plant action (see Appendix B).

O. Test Provisions

1. Control rod hold air exhaust valves.

    a. Individual test switches are provided which allow the operator to independently operate the hold air exhaust valves. The only restrictions are that the shock absorbers must be extended or the control rods seated.

2. Transient rod hold air exhaust valves.

    a. A test fire switch is provided which allows the operator to energize the fire relay, provided that the transient rod drive is in the lower limit position and that the transient rod top air is off. This relay operates the two transient rod hold air exhaust valves.

P. Key Bypasses Provided

1. The upper limit bypass key-switch will permit the operations necessary for the removal of the reactor vessel top plug and drive assembly. Under certain conditions it will also be useful in drive testing.

2. The main power bypass key-switch will permit limited operation of the system under emergency conditions with only the standby power supply as a power source.

3. The plug interlock bypass key-switch will permit control power to be applied for testing purposes with one or more of the drive motor power plugs disconnected.

APPENDIX B

OPERATING RULES AND PROCEDURES

I. INTRODUCTION

In view of the objectives of the Spert II facility, any attempt at a description of "normal" operation would be difficult. Each nuclear operation of the plant is considered to be an experiment investigating more or less unknown behavior. Every nuclear operation of the plant (and many non-nuclear ones) are therefore treated as potentially destructive tests.

The general operating procedures and administrative rules which will apply to all operations are outlined in this appendix.

II. GLOSSARY OF TERMS

A. Critical Experiment - Type I is the process of assembling an arrangement of fissionable material in the reactor under circumstances when the criticality conditions are grossly unknown. This would include the initial loading of new fuel assembly designs or core geometries, the loading of previously used materials under new conditions, or the achievement of critical loading after some alteration in fuel assemblies, structural materials or in-pile transducer materials has made uncertain any previous knowledge of the general point at which criticality is to be expected.

B. Critical Experiment - Type II is defined as the physical assembly of the reactor core with its structural components and in-pile transducers under conditions when the criticality and excess reactivity of the system are well-known from previous critical experiments.

C. Responsible Supervisor is defined as the supervisor designated by the Manager of the Reactor Projects Branch (Spert) to be responsible for the operation of the Spert II reactor in all its aspects and for the execution of experiments performed therewith.

D. Reactor Operator is defined as the person designated by the responsible supervisor to operate the nuclear console of the reactor in the performance of static and kinetic experiments. Such operators will have been previously trained in the operation of reactors and particularly in the performance of, and the hazards attendent to, kinetic tests by a suitable training period using the Spert I reactor.

E. Plant Operator is defined as the person designated by the responsible supervisor to operate the plant control panel in the performance of static and kinetic experiments.

F. Spert Senior Staff is defined as the group composed of the Spert supervisory personnel and such other persons as may from time to time be designated by the Spert Manager. At present this group consists of the following personnel:

(1) Manager, Reactor Projects Branch (Spert)

(2) Deputy Manager, Reactor Projects Branch

(3) Chief, Spert I Section

(4) Chief, Spert II Section

(5) Chief, Spert III Section

(6) Chief, Spert Engineering Section

(7) Chief, Spert Data Reduction and Interpretation Section

(8) Chief, Spert Instrumentation Section

This staff will formulate the program of experimentation to be carried out, and will review detailed procedures for the operation of reactor plants and for the performance of experiments. It will act as an advisory body to the Manager concerning his approval of these procedures, or modifications thereof.

III. PROCESS PLANT PROCEDURES

Significant hazards exist in the operation of the Spert II process plant itself, in the absence of any nuclear operation. The pressures and temperatures involved present the possibility of plant damage and injury to personnel in the event of accidental operator errors or component failures.

The probability of operator error will be minimized by extensive training of operators, and by the preparation of detailed operating procedures and checklists. Any mode of plant operation or set of operating conditions not covered in these procedures will be initiated only with the express approval of the responsible supervisor. Inasmuch as most such cases will involve questions as to the safety of the physical plant under conditions of unusual stress, this supervisor will consult with the Chief of the Spert Engineering Section and the Spert Senior Staff prior to giving his approval. A detailed procedure for plant maintenance and inspection will minimize component failures.

At any time that the reactor is in nuclear operation, or that the top plug of the reactor vessel is removed and fissionable material is present in the vessel, all actions concerning the process system must be approved by the responsible supervisor. This procedure will be necessary in view of the possible interaction between system conditions and the reactor neutronics.

Under no condition will the system be pressurized or the primary coolant pumps started without the permission of the responsible supervisor and the presence of a reactor operator on duty at the nuclear console, to assure that the control rods are properly latched down.

The above mentioned procedures and check-lists will of necessity be prepared during the non-nuclear check-out phase of the Spert II start-up since they will depend to a large extent upon actual details of the system design and performance.

IV. OPERATIONAL PROCEDURES FOR CRITICAL EXPERIMENTS

A. Pre-Experiment Procedure

Any manipulations of fuel or other items within the reactor vessel with the top head (and hence the control rod drive system) removed from the vessel, will be under the direct control of the responsible supervisor or a person expressly designated by him. Such operations will be categorized by him as either Type I or Type II critical experiments.

Before any related set of Type I critical experiments is begun, a program write-up will be prepared outlining the purpose of the experiments, detailed description of the procedures to be followed in the assembly, data to be taken, special precautions to be taken, and personnel to participate. This write-up will be approved by the Spert Senior Staff and all assigned personnel will be familiarized with it prior to the initiation of the experiments.

Before any Type II critical experiment is begun a detailed description of the procedures to be followed in the assembly will be prepared and all assigned personnel will be familiarized with it.

B. General Experiment Procedures

The following general procedures will be followed in all critical experiments of either type:

1. Supervision

The responsible supervisor or a person expressly designated by him shall be present at the reactor vessel to supervise the loading. This supervisor shall be aware of all persons in the reactor building during the experiment. A reactor operator will be on duty at the control console during such experiments.

2. Source

A neutron source of sufficient strength shall be present in the system to permit the reactor operator to observe the neutron level of the system during all loading operations.

3. Communication

This operator shall be in direct communication with the crew at the reactor at all times.

4. Records

A running log will be kept by the reactor operator during each experiment. This log will contain notations of each change made in the reactor core, disposition of fuel assemblies, and all other information thought pertinent.

V. GENERAL PROCEDURES FOR REACTOR OPERATION

Whenever the reactor is to be operated, the administrative procedures which have been used successfully with Spert I for the last three years will be in effect. These procedures, suitably modified for Spert II are outlined as follows:

A. Classification of Tests

Each series of tests to be performed with Spert II will be categorized by the Spert Senior Staff into one of the three categories of hazard which have been used for the Spert I tests. These are "slight", "moderate", or "high" hazard probability. The terms are of necessity loosely defined.

1. "Slight" Hazard Test

"Slight" hazard probability is used to describe test series for which sufficient previous data and experience exist to permit confident predictions of the results of a test.

2. "Moderate" Hazard Test

The category of "moderate" hazard will be used to categorize any series of tests, the results of which cannot be reasonably predicted by extrapolation or interpretation of previous test results. Once such tests have been safely performed, repeated tests of the same type will then be classed in the "slight" category. An exception to this statement will be made under conditions when the reactor fuel contains high concentrations of fission products. Operation under these conditions will frequently be classed as having "moderate" hazard even though it constitutes a repetition of previous tests. Whenever tests are classed as having moderate hazard probability, the responsible supervisor will obtain higher management approval to conduct these tests which will be co-ordinated with the activities of the AEC-IDO Site Survey Branch from whom meteorological information and area protection are obtained.

3. "High" Hazard Test

The category of "high" hazard probability implies a planned approach to destruction of the reactor, and tests of this nature are not presently planned for Spert II.

B. Supervision

During the progress of a nuclear test, the responsible supervisor, through his trained operators, will be in direct and sole control over the progress of the test. The procedure to be followed for a given test will have been approved in advance by the Spert Senior Staff and the operating personnel will have been acquainted with the detailed plans for the test, and with possible alternatives which might arise due to unforeseen events.

C. Reactor Area Security

When the plant is in readiness for nuclear operation, the reactor building and area will be evacuated. The assigned health physicist will make an inspection of the reactor building and area to ascertain that no one remains in the area. He will then return to the control center, set up a road block at the entrance to the Spert II access road, alert the guard at the entrance to the Spert area that a test is underway in Spert II and report to the responsible supervisor that the area is clear.

D. Operating Personnel

Because of the limited space in the control room, and the necessity for communication between operating and supervisory personnel, access to the control room during operation of Spert II will be restricted to the following personnel:

(1) The responsible supervisor,

(2) The reactor operator,

(3) The plant operator,

(4) The assigned health physicist,

(5) The assigned electronics technician,

(6) Any other Spert personnel so designated by the responsible supervisor, not to exceed two persons.

Whenever the reactor is in operation, at least two persons shall be present in the control room. These shall be a trained reactor operator and the supervisor or someone expressly designated by him.

E. Start-up

1. Check-out Procedures

Prior to start-up, the reactor operator will run through a check list procedure to determine the condition of the plant system and the operability of all instruments and safety features. The operator will also announce his intent over the intercommunication system which communicates with all parts of the reactor building. He will then listen for response from the reactor building via any of the several live microphones in the reactor building. This procedure is not intended as means of evacuating the area, but as a backup on step C above.

2. Warning System

When any of the reactor control elements is raised above its seated position a 30 sec warning horn is sounded at the reactor building and a set of interior and exterior warning lights is turned on. Any personnel present in the building when the horn sounds will prevent operation of the reactor by means of manually operated scram buttons in the reactor buildings.

F. Shutdown

At the conclusion of operation the reactor operator will ascertain that all control rods and the transient rod are seated, that the rod drives are in their lower limit positions and that the control rods are latched down. He will then make the appropriate entries in the operational logs and turn off the control power to the reactor. The key to the reactor power switch will be removed and returned to its normal storage position.

G. Re-Entry To Reactor Area

Upon completion of the test and following nuclear shutdown of the plant, the assigned health physicist will remove the road block and accompany the first personnel to re-enter the area. He will determine the radiation levels in the building and will inform the supervisor of any restrictions on working time in, or access to, the area.

Normal radiation protection procedures will be followed at all times.

If operational problems require access to the reactor building at any time after the area has been cleared and the roadblock set up, the responsible supervisor may approve the entry of designated persons while the roadblock is in place. These shall include the health physicist who shall inform the supervisor of the status of the area as described in step C above following such re-entry and prior to any reactor operation. No such entry will be allowed if the warning lights are on.

H. Exclusion Area

During all reactor operation, an exclusion area one-half mile in radius will be maintained around the reactor.

VI. GENERAL PROCEDURE FOR THE REMOVAL OF THE REACTOR VESSEL TOP PLUG

The removal of the vessel top plug which contains the rod drive mechanisms involves a specialized operation of the control circuitry and some extra precautions. Therefore a general procedure will be outlined here. Detailed procedures will be prepared in the light of actual operating experience.

(1) It is first determined that the reactor is in a properly shut down condition with power off and that the reactor vessel is at atmospheric pressure.

(2) The plug removal operation will take place under the direct supervision of the responsible supervisor or someone expressly appointed by him. During the complete procedure a reactor operator must be on duty at the reactor console with neutron level instruments in operation.

(3) The access ports in the vessel head are opened and any instrumented assemblies which are in the reactor core and whose leads pass through these ports are removed from the vessel to storage.

(4) The four control rods and the transient rod are uncoupled from their respective drive pistons using remote handling tools through the access ports.

(5) The reactor operator applies power to the control system, resets the scram circuit and throws the upper limit bypass switch to the bypass condition. This switch is key-operated and the key is under the control of the responsible supervisor. The bypass switch has as one of its functions the invalidation of the reactor warning light circuitry.

(6) On instruction from the supervisor at the reactor, the operator withdraws the transient rod drive and rod piston to the completely withdrawn position. Personnel at the reactor visually check to make sure that the rod has remained in place in the core.

(7) On instruction from the supervisor at the reactor, the operator withdraws the control rod drives and rod pistons one at a time stopping each at the 6 in. withdrawn position to allow personnel at the reactor to visually check that each rod is decoupled and has remained in place in the core.

(8) All control rod drives and rod pistons are completely withdrawn.

(9) After making sure that the upper limit bypass switch is still in the bypass position in order to prevent a Fast Rundown, the reactor operator informs the supervisor that the scram air lines may be disconnected.

(10) The crew at the reactor blocks, vents and disconnects the reactor scram air lines; mechanically clamps the rod pistons to prevent their motion in the absence of power; and stands clear of the hold air exhaust lines.

(11) On instruction from supervisor the operator turns off the reactor control power.

(12) The crew at the reactor blocks, vents and disconnects the transient rod top air and hold air lines and the control rod hold air lines. The power plugs to the drive system are disconnected and the top plug is removed to dry dock.

Procedures which are approximately the reverse of the above are used to place the drive system in operation at the dry dock or to replace the vessel top plug.

APPENDIX C - INHALED DOSE CALCULATIONS

In order to calculate the inhaled dosages quoted in Section IV (E) of this report, the following procedure has been followed. The radioactive cloud volume as a function of distance from the origin is assumed to be

$$V = \left[\pi^{\frac{1}{2}} C X^{\frac{2-n}{2}}\right]^3, \qquad (1)$$

where
V = cloud volume (m^3),
X = distance from origin (m),
n = Sutton's stability parameter (dimensionless),
C = Sutton's diffusion coefficient ($m^{n/2}$).

The concentration of activity in the cloud is assumed to follow the following relation:

$$C(r) = \frac{A}{V} e^{-\left(\frac{r}{r_0}\right)^2}, \qquad (2)$$

where $C(r)$ = radioactive cloud concentration $\left(\frac{\text{curies}}{m^3}\right)$,
r = distance from center of cloud (m),
r_0 = cloud radius (m) $= C X^{\frac{2-n}{2}}$,
A = total curies in cloud.

The activity of a specific isotope inhaled by a person standing directly in the path of the cloud during its total passage is given by

$$\begin{aligned} I &= \int_0^{\infty} J\,C(r)\,dt \\ &= \int_{-\infty}^{\infty} J \frac{A}{V} e^{-\left(\frac{r}{r_0}\right)^2} \frac{dr}{v} \\ &= \frac{JA}{vV} r_0 \pi^{\frac{1}{2}} = \frac{JA}{v V^{2/3}}, \end{aligned} \qquad (3)$$

where
I = inhaled activity (millicuries),
v = cloud velocity (m/sec),
J = inhalation rate (liters/sec).

Assuming J = (17 liters/min) · (1.67×10^{-2} min/sec), then

$$I = 2.84 \times 10^1 \frac{A}{v V^{2/3}} \text{ millicuries.} \qquad (4)$$

For a specific isotope the cloud activity is calculated by

$$A = (8.4 \times 10^5)\ P\ y\,(1-e^{-\lambda t_o})\,e^{-\lambda(X/v)} \text{ curies,} \tag{5}$$

where
- P = reactor power (Mw),
- y = fractional fission yield,
- λ = radioactive decay constant for isotope (sec^{-1}),
- t_o = operating time prior to release of fission products (sec),
- X = distance from release (m),
- v = cloud velocity (m/sec).

The internal dose date at a time t following inhalation is given by

$$\frac{dD}{dt} = \frac{f_a I E}{f_c\, g}\ \frac{(3.7 \times 10^7)(1.6 \times 10^{-6})}{10^2}\ e^{-\lambda_e t} \text{ rads/sec,} \tag{6}$$

where
- f_a = fraction of inhaled activity deposited in critical organ,
- f_c = fraction of organ available (= 0.2 for bone, = 1 for thyroid),
- g = mass of critical organ (grams),
- E = effective energy of radiation (Mev/disintegration),
- λ_e = effective decay constant (sec^{-1}) = $\lambda + \lambda_b$,
- λ_b = biological elimination constant (sec^{-1}),
- t = time after inhalation (sec),
- I = inhaled activity (millicuries).

Therefore, the total dosage for an infinite time thereafter is

$$\begin{aligned} D &= \int_0^\infty \frac{dD}{dt}\, dt \\ &= \frac{f_a I\ E}{f_c g \lambda_e}\ (5.91 \times 10^1) \text{ rads,} \end{aligned} \tag{7}$$

and using equations (4) and (5) above

$$D = (1.41 \times 10^5)\, \frac{P}{v\, V^{2/3}} \left\{ \frac{f_a E Y}{f_c \lambda_e g}\, (1-e^{-\lambda t_o})\, e^{-\lambda(X/v)} \right\} \text{ rads} \tag{8}$$

for a specific isotope.

The constants used for the calculations shown in Figures 14 thru 19, of this report are shown in Table 2. The meteorological constants used are shown in Table 1, page 38.

TABLE 2

ISOTOPIC DATA FOR INHALED DOSE CALCULATIONS (SEE TEXT FOR DEFINITIONS OF SYMBOLS)

Isotope	Critical Organ	g (grams)	f_c	f_a	E(Mev)	y	λ (sec^{-1})	e (sec^{-1})
I^{131}	Thyroid	20	1	0.15	0.22	0.031	9.86×10^{-7}	1.03×10^{-6}
I^{132}	Thyroid	20	1	0.15	0.50	0.045	8.02×10^{-5}	8.02×10^{-5}
I^{133}	Thyroid	20	1	0.15	0.40	0.051	9.44×10^{-6}	9.44×10^{-6}
I^{134}	Thyroid	20	1	0.15	0.65	0.063	2.88×10^{-5}	2.88×10^{-5}
I^{135}	Thyroid	20	1	0.15	0.31	0.062	2.20×10^{-4}	2.20×10^{-4}
I^{136}	Thyroid	20	1	0.15	2.2	0.034	8.06×10^{-3}	8.06×10^{-3}
Sr^{89}	Bone	7×10^{3}	0.2	0.22	0.55	0.047	1.52×10^{-7}	1.54×10^{-7}
Sr^{90}-Y^{90}	Bone	7×10^{3}	0.2	0.22	1.00	0.052	8.80×10^{-10}	2.97×10^{-9}
Y^{91}	Bone	7×10^{3}	0.2	0.14	0.57	0.056	1.41×10^{-7}	1.57×10^{-7}
Zr^{95}-Nb^{95}	Bone	7×10^{3}	0.2	0.12	0.2	0.067	1.24×10^{-7}	1.70×10^{-7}
Ba^{140}-La^{140}	Bone	7×10^{3}	0.2	0.20	1.06	0.066	6.26×10^{-7}	6.69×10^{-7}
Ce^{141}	Bone	7×10^{3}	0.2	0.10	0.16	0.059	2.43×10^{-7}	2.58×10^{-7}
Pr^{143}	Bone	7×10^{3}	0.2	0.063	0.31	0.061	5.81×10^{-7}	7.30×10^{-7}
Ce^{144}-Pr^{144}	Bone	7×10^{3}	0.2	0.10	1.29	0.059	2.92×10^{-8}	4.46×10^{-8}
Eu^{154}	Bone	7×10^{3}	0.2	0.09	0.37	0.0085	4.07×10^{-9}	9.80×10^{-9}

RE
PHILLIPS
FEB 1 9 19
NRTS
TECHNICAL LIBRARY

Phillips 66