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SAFETY ANALYSIS REPORT - SNAPTRAN 2/10A-1 SAFETY TESTS

Edited by J. M. Waage

PHILLIPS
PETROLEUM
COMPANY

ATOMIC ENERGY DIVISION

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Edited By

J. M. Waage

Prepared By

Phillips Petroleum Company

F. L. Bentzen
G. F. Brockett
O. M. Hauge
N. R. Horton
D. T. Jones
G. B. Matheney
T. M. Quigley
J. M. Waage

Atomics International

R. L. Detterman
M. S. Dunenfeld
L. I. Moss
L. A. Rash
W. A. Rhoades

PHILLIPS
PETROLEUM
COMPANY

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

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SUMMARY

Nuclear safety testing of a reactor of the SNAP 2/10A type will be conducted at the Test Area North (TAN) of the National Reactor Testing Station by Phillips Petroleum Company. The safety investigations are a part of the Atomic Energy Commission's nuclear safety program. The broad objectives of the test program are to demonstrate and determine the consequences of nuclear accidents approximating the maximum credible and to provide information regarding the characteristics of the SNAP 2/10A reactor of importance to evaluating and predicting the behavior of the reactor with respect to nuclear safety.

The initial test series, designated as SNAPTRAN 2/10A-1, is aimed at providing a portion of the information needed to attain these objectives. This test series is directed at providing safety information on the beryllium reflected SNAP 2/10A reactor in an atmospheric environment. The flight reactor has been modified to permit rapid reactivity addition for the safety tests. The experimental program includes reactor kinetic tests aimed at measuring the dynamic coefficients of the reactor, static physics measurements essential to interpretation of kinetic test results, and a nuclear excursion approaching the maximum credible.

The potential hazards of conducting the experimental program have been evaluated and the results of the safety analysis are presented in this report. The report contains a brief description of the experimental program, a description of the reactor test package and control system, a description of the test site and facilities, a discussion of the operating philosophy and test procedures, and an evaluation of the potential hazards attendant to the experimental program. The areas of consideration in this evaluation include materials handling and control, possible operator error and system failure, radiological hazards associated with the static and non-destructive kinetics tests, and the consequences of the destructive test.

On the basis of the safety analysis studies conducted, it has been concluded that the test program can be conducted without hazard to operating personnel or the general public.

SAFETY ANALYSIS REPORT
SNAPTRAN 2/10A-1 SAFETY TESTS

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I. INTRODUCTION

The development and utilization of nuclear systems for space applications present unique problems with respect to nuclear safety. In recognition of the need for information of importance in evaluating the hazards of nuclear systems intended for auxiliary power use, the U. S. Atomic Energy Commission has initiated aerospace safety studies as part of its overall nuclear safety programs under the Nuclear Safety Engineering and Test Branch of the Division of Reactor Development. That portion of the aerospace safety program directed toward studying the kinetic behavior and certain of the consequences of nuclear accidents of SNAP type reactors has been designated as SNAPTRAN. Consideration has been given to the kinds of nuclear accidents which can conceivably occur and two general categories have been found. These are: (1) nuclear excursions occurring in an atmospheric environment during assembly and launch, and (2) nuclear excursions resulting from immersion of the reactor in water or wet earth. The SNAPTRAN 2/10A-1 test series is aimed at providing information regarding the beryllium reflected reactor in air which is of importance in assessing the hazards involved in launching a SNAP 10A reactor into space[1] and in aiding in the understanding of the self-limiting characteristics of the reactor. The experimental investigations will be carried out at the NRTS by Phillips Petroleum Company as part of the STEP Project.

The reactor system for performing the SNAPTRAN 2/10A-1 studies has been designed by Atomics International and incorporates flight system design where possible with modifications to allow rapid insertions of reactivity. The reactor will be fabricated, assembled, and acceptance tested by Atomics International at Canoga Park, California. The reactor will then be shipped to the NRTS where Phillips Petroleum Company will assume the operational responsibility for conduct of the tests. Technical direction of the test program for the initial critical experiment will be provided by Atomics International personnel at the NRTS. Technical direction of all other phases of the test program will be provided by Phillips Petroleum Company with technical consultation provided by Atomics International. Conduct of the critical experiment will rely on the experience and techniques developed by Atomics International in the performance of critical experiments with SNAP type reactors, while conduct of the transient tests will rely on the experience and techniques developed by Phillips Petroleum Company in the performance of reactor transient tests with the SPERT reactors.

In the preparation of this report, Phillips Petroleum Company has provided the site and facility description and the evaluation of the test program hazards, and Atomics International has provided the description of the reactor and the kinetics analysis.

II. EXPERIMENTAL PROGRAM

The SNAPTRAN 2/10A-1 nuclear safety testing program will be performed on a modified flight reactor. The SNAP 2/10A reactor is ZrH_x-U fueled, NaK cooled, and beryllium reflected. The core is a right circular cylinder approximately 9 in. in diameter and 12 in. long containing 37 cylindrical fuel rods in a close-packed array.

For the SNAPTRAN 2/10A-1 tests the control drum drives have been modified to permit rapid addition and removal of reactivity. The reactor test system includes a thermally insulated housing, equipped with electric heaters, surrounding the reactor core vessel and beryllium reflector. Nitrogen will be used to isothermally heat the reactor core and beryllium reflector. The NaK coolant will be replaced by nitrogen during all tests with the exception of the final test series.

The experimental program encompasses a series of nondestructive tests and a destructive test. The nondestructive tests include static tests and those kinetic tests in which minor damage to the reactor occurs. The destructive test, in which the reactor will be destroyed, is a modeling of the maximum credible incident.

A. Program Objectives

The objectives of the SNAPTRAN 2/10A-1 safety program are to obtain information regarding the safety aspects of the SNAP 2/10A reactors during assembly, launch, and ascent. Specific objectives of the experimental program are:

(1) to demonstrate the consequences of an accident approximating the maximum credible incident,

(2) to determine the dynamic characteristics of the SNAP 2/10A reactor which are of importance in evaluating the nuclear safety hazards associated with launch, and

(3) to provide physics and engineering information which will aid in the understanding of the self-limiting characteristics of the reactor and the evaluation of calculation techniques.

In order to fulfill these program objectives, it is necessary to perform both destructive and nondestructive tests. The destructive tests, that is, tests involving the addition of a sufficient amount of reactivity to destroy the reactor, are aimed at providing information of importance in evaluating the consequences of a reactor runaway. Information regarding the nuclear energy release and other information of importance to determining the reactor disassembly process will be obtained. The nondestructive tests are directed toward providing information of importance in establishing the dynamic characteristics of the nuclear system. The information obtained is expected to aid in understanding the shutdown mechanisms and in the development of analytical models for predicting the kinetic behavior of the reactor.

B. SNAPTRAN 2/10A-1 Nondestructive Test Program

The nondestructive portion of the program will include loading the reactor to the desired excess reactivity, performance of static physics measurements, and performance of kinetic tests initiated by step and impulse reactivity insertions. These tests will be performed with no NaK in the core and with the reactor at various temperatures up to 1400°F.

The measurements to be made will include a determination of the peak power, the nuclear energy release, the internal fuel temperature and pressure generated by release of hydrogen, the direct radiation, and, in the case of limited fuel rod damage, the amount and distribution of fission products released. These and other data to be obtained are expected to provide information regarding the self-shutdown mechanisms effective in the nondestructive region. However, the nondestructive tests can not be expected to provide data needed to evaluate the consequences of an extreme power excursion. The purpose of the destructive program is to provide such information.

1. Mechanical Checkout

The mechanical performance of the control drums, interlock circuits, etc., will be checked for proper operation following installation of the reactor test package on the test dolly in the IET facility prior to loading fuel. The tests will include determination of the tooth clutch drop-out times for various scram spring loads, position-time data for scramming and firing the control drums, and the effects of pneumatic drive pressure variation on drum rotation time.

2. Static Tests

a. Ambient Temperature

The nuclear testing will begin with a loading to critical[2] using the subcritical multiplication rate technique to estimate the critical number of fuel rods. Loading will then continue until the full fuel configuration is reached. The desired excess reactivity will be achieved by the addition of beryllium shims to the control drums and permanent reflector.

Measurements of importance to reactor operation and to the interpretation of the reactor transient behavior will be made. These measurements will include the following:

(1) detailed control drum calibrations, including drum interactions,

(2) determination of reduced prompt neutron lifetime (ℓ/β_{eff}),

(3) measurement of the worth of reflector, fuel, and moderator,

(4) measurement of flux distribution, and

(5) power calibration.

b. Elevated Temperature

Isothermal temperature reactivity defect measurements and control drum calibrations will be made as a function of temperature from ambient to approximately 1400°F.

3. Transient Tests

a. No Fuel Rod Damage

(1) Ambient Temperature. Transient testing at ambient temperature will begin with a series of step reactivity insertion tests

starting with small reactivity additions which result in reactor periods of several seconds. Initially, several of these tests will be terminated by programmed scramming of the reactor after a preset time has elapsed, that is, before peak power is reached. These step tests will proceed to larger reactivity insertions and extended operating times; however during the tests, the maximum internal fuel rod temperature will not be permitted to exceed approximately 1400°F.

Self-limiting step tests will then follow using the same fuel temperature criterion to determine the maximum reactivity to be inserted.

Further study of the kinetic behavior of the reactor will be made by a reactivity impulse technique in which reactivity is inserted and withdrawn rapidly by means of high speed drum rotation. The impulse tests will be initiated starting with peak reactivity insertions less than prompt critical and proceeding to the maximum available or until the fuel rod temperature approaches 1400°F, whichever occurs first.

b. Partial Fuel Rod Damage

The testing will proceed at ambient temperature from nondamaging reactivity insertions to insertions which will result in internal pressures that rupture fuel rods but which are not expected to damage the core vessel, the adjacent control drums, the beryllium reflector, or other components. The number of these tests will be limited by the fuel available.

C. SNAPTRAN 2/10A-1 Destructive Test Program

The kinetics testing of the SNAPTRAN 2/10A-1 reactor package will include a modeling of a maximum credible incident in which the reactor will undergo destruction. The destructive test is expected to provide information which will be of importance in evaluating the consequences of such an incident.

In addition to providing the information described in the non-destructive test program, the nature and extent of core disassembly will be determined, and detailed fission product release data will be obtained by means of extensive downwind measurements. The monitoring and collection of data from the downwind grid is the joint responsibility of Phillips Petroleum Company and ID Health Physics personnel.

The reactor to be used for the destructive test will, insofar as possible, duplicate the nuclear characteristics of the flight reactor. NaK coolant will be added to the vessel and the thermal housing will be removed. Removal of the housing will facilitate instrumentation of the reactor and will allow visual and infrared photographs to be obtained.

Several transient tests will be performed in a region well below fuel damage in order to determine the extent to which NaK in the vessel has influenced the kinetic behavior, to check the mechanical operation of the drum drives, and to verify that the instrumentation is working correctly. The destructive test will be performed under strict meteorological conditions to minimize the potential hazards and to assure a proper wind direction for fission product cloud sampling.

D. Post-Test Program

Upon completion of the destructive test an area survey will be made to determine the extent of contamination spread and the distribution of reactor structural components. Fuel and reactor structural component fragments will be examined in the examination area to determine thermal and blast effects. The core remains will be dismantled and examined to determine the nature and extent of mechanical destruction.

III. TEST PACKAGE DESCRIPTION

A. General

The SNAPTRAN 2/10A-1 test package (Fig. 1), a modified SNAP 2/10A flight machine, has been designed to provide the capability of large step and impulse reactivity insertions. Wherever possible, the design of the SNAP 2/10A flight machine is used; however, certain deviations from flight hardware are necessary. These include modifications to the vessel and core to provide for the circulation of nitrogen through the core and to permit internal instrumentation leads to be brought out, modifications to the control drums and fixed reflector which permit the use of additional Be shims to increase the reactivity of the system, and a redesign of the control drum drive system to provide for scram and rapid drum movement for transient testing in addition to the normal drum drives for static experiments. The test package also has provisions for heating the core and reflector to 1500°F with an external heating system. Fig. 2 shows the test package model at IET with the test cell building partially removed.

Since the four control drums will be used for both static and transient tests, each drum is capable of two modes of reactivity insertion as well as being provided with scram capability. For normal reactor control the drums are driven, individually or grouped, by continuously variable motor drives capable of rotating the drums at rates from 0.05°/sec to 1°/sec. The impulse drums can be driven through an arc of 450°, that is, 225° on either side of the maximum reactivity position (0°). To simplify the description, all drum movements to the left, or counterclockwise, of 0°, assuming 0° to be the maximum reactivity position, are said to be in the positive sector and drum movements to the right, or clockwise, of 0° are in the negative sector. One pair of diametrically opposed drums has been equipped for step reactivity insertions and the other pair for impulse reactivity insertions. Since the stresses developed in the drive system during step and impulse reactivity insertions are greater than those normal for the flight machine, it was necessary to strengthen the control drum mounting brackets and shafts and to add counterweights to the shafts opposite

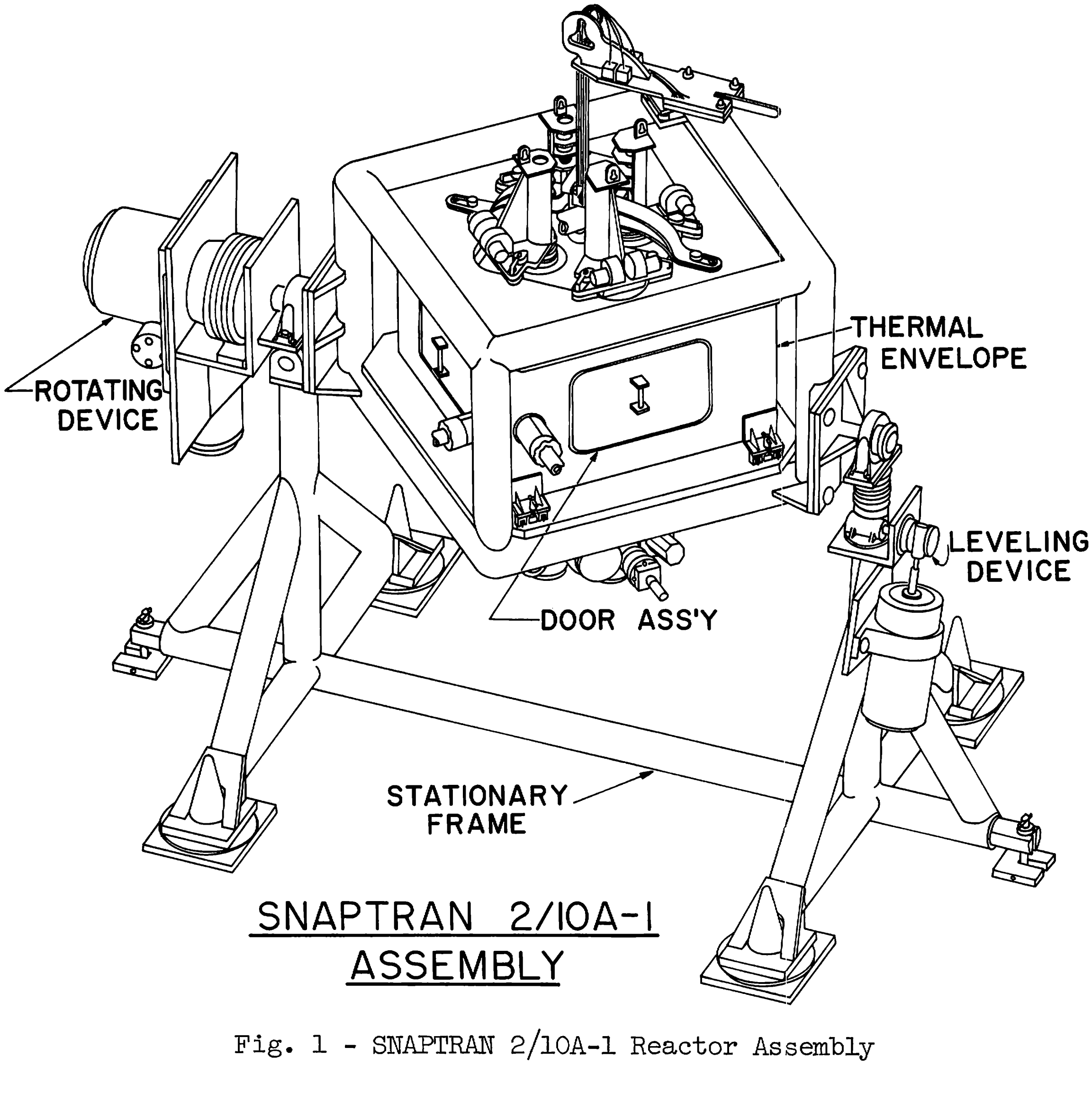


Fig. 1 - SNAPTRAN 2/10A-1 Reactor Assembly

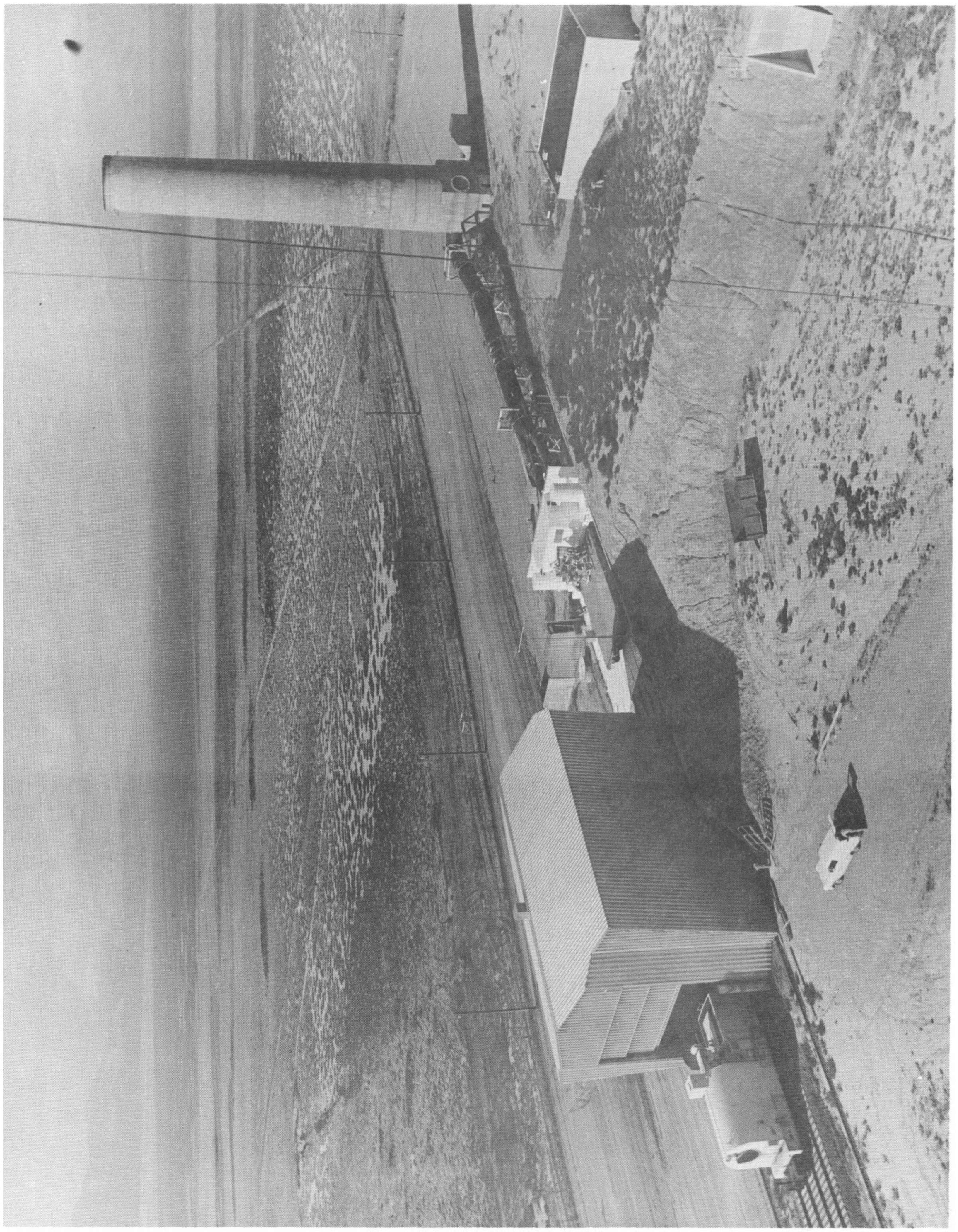


Fig. 2 - Test Package Model in IET

The step drums operate only in the positive sector and step reactivity inputs will be accomplished by driving one or two of the drums to 0° from a position 45° out. The step drums have mechanical stops that prevent drum rotation beyond 0°. Each of the step drums is driven by a pneumatic-mechanical drive system capable of rotating the drums 45° in 0.050 second. In a two drum step insertion, the two drives are linked together with a chain and sprocket assembly to assure synchronous movement. The scram mechanisms for the step drums are mechanical and are designed to rotate the drums from 0° to 135° in 0.150 second.

An impulse test will be accomplished by rotating one or two drums through the full 450° of travel; however, only for ~ 270° (135° on each side of 0°) will the drums be in the active region. The system is designed to rotate the drums through this 270° arc in 0.030 second by a pneumatic-mechanical drive system. During two drum impulse tests, the drums are linked together with a chain and sprocket system to assure a synchronous movement. In addition to initiating impulse tests, the impulse drums will be used to initiate the destructive test. The drums will be driven from -225° to 0° with the active region, -135° to 0°, being traversed in 0.010 second. Mechanical stops will be added prior to the destructive test to prevent drum rotation beyond 0°. The impulse scram is a pneumatic-mechanical unit and is designed to rotate the drums from 0° to +135° in 0.150 second and from 0° to -135° in a somewhat longer time.

Following the nondestructive test series, the test package will be moved to the Examination Area and disassembled, inspected, and modified for the destructive test. The modifications include the addition of stops on the impulse drums, replacement of the core vessel by one with a welded head, charging of the core vessel with NaK, and removal of the thermal envelope.

B. Structure

The structural framework for the test package is made from 5 in. OD tubular aluminum and includes the mounting structure and the rotating framework. The mounting framework provides vibration and shock mounting to the flatcar, package leveling, and support for the rotational framework. The rotational framework supports the reactor assembly, control drum assemblies, control drum drives, and the thermal envelope.

The mounting structure is positioned on the flatcar by four guides welded to the deck and held down with two mounting pins. The mounting pins are designed to prevent the package from turning over during transit or maintenance operation, and to prevent vibrational loads or shock loads from the flatcar being transmitted to the package.

The rotational framework is attached to the mounting framework with trunions and bearings. A motor driven jackscrew, which levels the rotational framework in the transverse direction, is located at one end of the mounting framework. On the other end of the mounting framework is the rotational motor and brake assembly. This is used to level the package in the rotational direction. The package level positions are indicated on the console and controlled from it. The rotational motor-brake assembly is used to invert the framework in order to permit removal or replacement of the control drum drive assemblies. Package rotation is also used to make up the couplings between the N_2 piping on the package and the piping on the flatcar. The motor drive unit is such that the brake is set whenever power is off the motor. This permits the reactor package to be held in any position required for maintenance, and prevents the unit from swinging back in the event power is lost to the motor.

A thermal envelope surrounds the reactor and is supported on the rotational framework. The thermal envelope serves to insulate the reactor from the atmosphere during the tests above ambient temperature and contains the heaters used to heat the Be reflector. The outside of the envelope is stainless steel and the inside is Super X insulation. There are four doors capable of remote removal, one on each side of the envelope. Mounted on these doors are the 12.5 kw electrical heaters which are used to heat the Be reflector and to maintain reactor temperature during the isothermal tests.

For the tests above ambient temperature, the reactor temperature will be raised by a heated stream of N_2 gas. Up to 0.2 lb/sec of N_2 at 1500°F will be required for the high temperature tests. The N_2 supply system (see Fig. 3) has been designed to fulfill the heating and cooling requirements of the test package.

The equipment associated with the N_2 system includes three 450-gallon storage tanks for liquid N_2, an ambient air vaporizer, a gaseous N_2 receiver, flow measuring equipment, an N_2 heater, and the associated piping, instrumentation, and controllers.

The liquid N_2 for the process will be trucked in and stored in three liquid N_2 storage tanks until it is required for a test. The tanks are pressurized by air heating in a recirculation line. The tanks are protected against overpressure by a rupture disc and relief valve.

The liquid N_2 flows from storage to the vaporizer where it is vaporized and heated to about 70°F and is then stored in the N_2 receiver. The receiver is also protected against overpressure by a relief valve.

From the N_2 receiver, the gas flows through the flow orifices and the pressure controller to the N_2 heater. The wide variation in flow rates required for the different temperatures dictated the use of five orifice runs. The flow is switched through the proper orifice by the use of solenoid valves, and the input signals are switched into the recorder by solenoid valves, also.

The N_2 heater is controlled by a saturable reactor and serves to heat the gas to the required test temperature.

After leaving the N_2 heater, the gas flow is divided: one stream bypasses the package and is used to assist in flow control at low rates; one stream is used to heat the thermal box and external reflector; and the third stream is used to heat the reactor. Flow in each stream is controlled by flow control valves. After one pass through the system, the gas is exhausted to the atmosphere.

C. Reactor (see Fig. 4)

1. Core

A cross section of the reactor is shown in Fig. 5. The core is a right cylinder which consists of 37 fuel rods arranged in a triangular array on 1.260 in. centers to form a hexagon 8 in. across the flats. The fuel rods are held in this array by upper and lower grid plates and Be filler pieces are used to adapt the hexagonal core to the cylindrical vessel described below. The fuel-moderator for the reactor is an alloy of zirconium hydride and 10 wt% of 93% enriched uranium. The fuel-moderator density is 6.08 g/cm^3 and its volume is 520 in^3 (8540 cm^3). The reactor core contains 4.75 kg of U^{235} and 464 gram-moles of H_2. The fuel-moderator is clad with Hastelloy N having a wall thickness of 0.015 inch. The outside diameter of each fuel element is 1.250 in. and its length is 12.25 inches. A gap of approximately 0.001 in. exists between the cladding and the fuel-moderator. The ultimate strength of the fuel-moderator alloy is about 15,000 psi, and the bursting pressure of the Hastelloy N cladding is about 1500 psi. The ends of the fuel rods have caps and grid plate indexing pins welded to each end. The upper and lower grid plates which are fabricated from Hastelloy C provide precise positioning of the fuel rods with respect to the vessel and to each other.

2. Vessel (see Fig. 6)

The reactor vessel is cylindrical in shape, 8.94 in. OD, with a 0.032 in. wall thickness. It is fabricated from type 316 stainless steel. Two vessels will be provided for the test series, one for the initial static and transient tests, and one for the destructive test. The vessel used for the initial static and transient tests will have a clamped head, while the vessel used for the destructive test will have a welded head. Because of the high temperatures involved, a metal-to-metal seal is used between the vessel and the head on the bolted head vessel. Although this joint may not be leak-tight during operation, any leakage gases will be carried off by the thermal envelope exhaust system. During the initial loading and critical tests, the vessel head

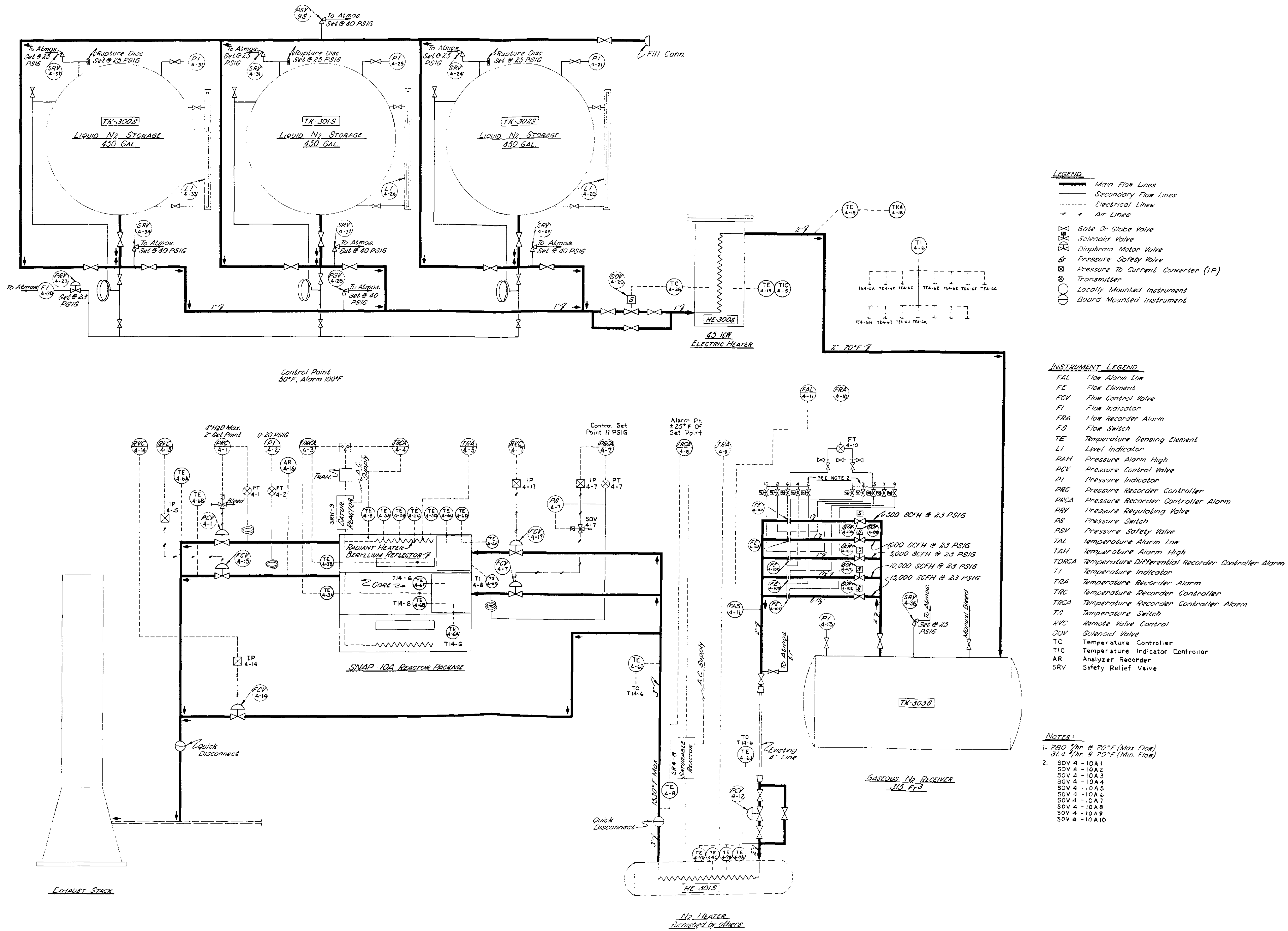


Fig. 3 - Nitrogen Supply System

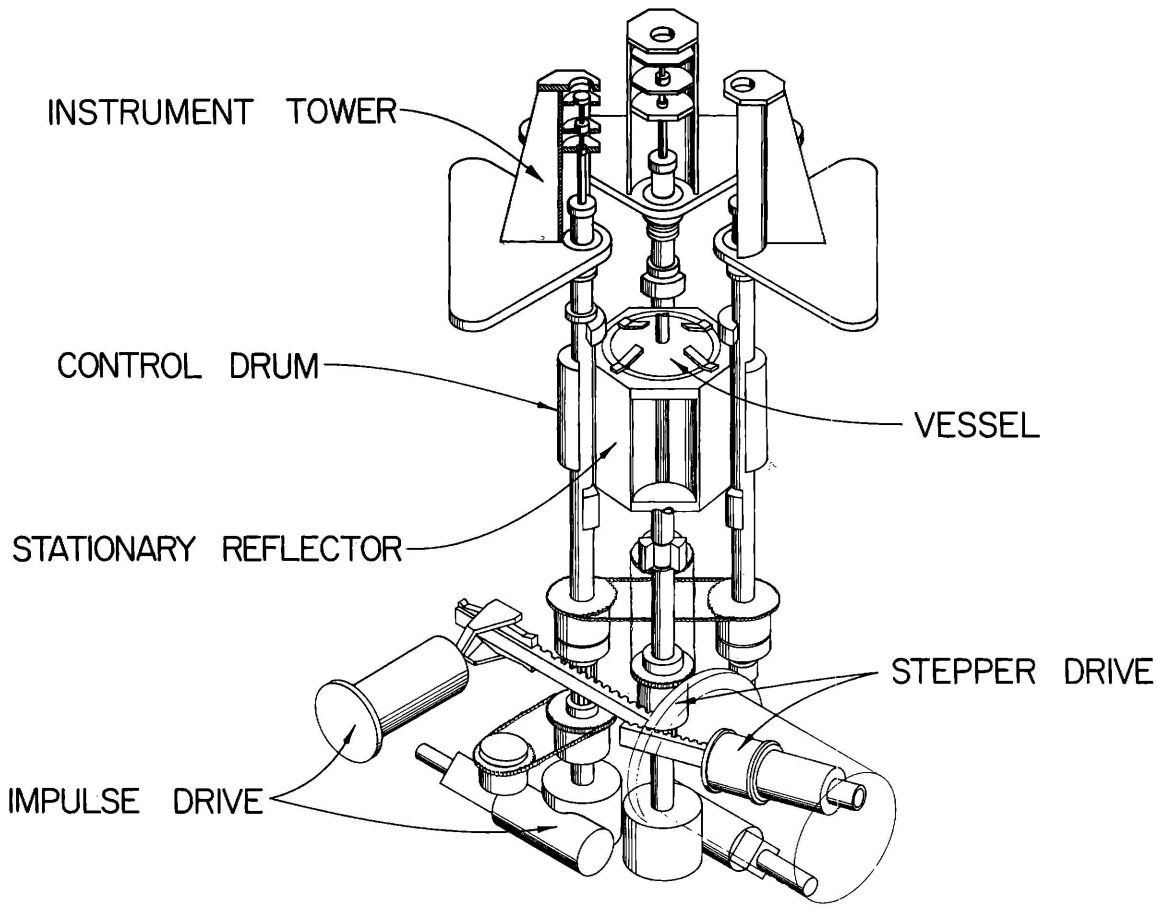


Fig. 4 - Reactor Pictorial

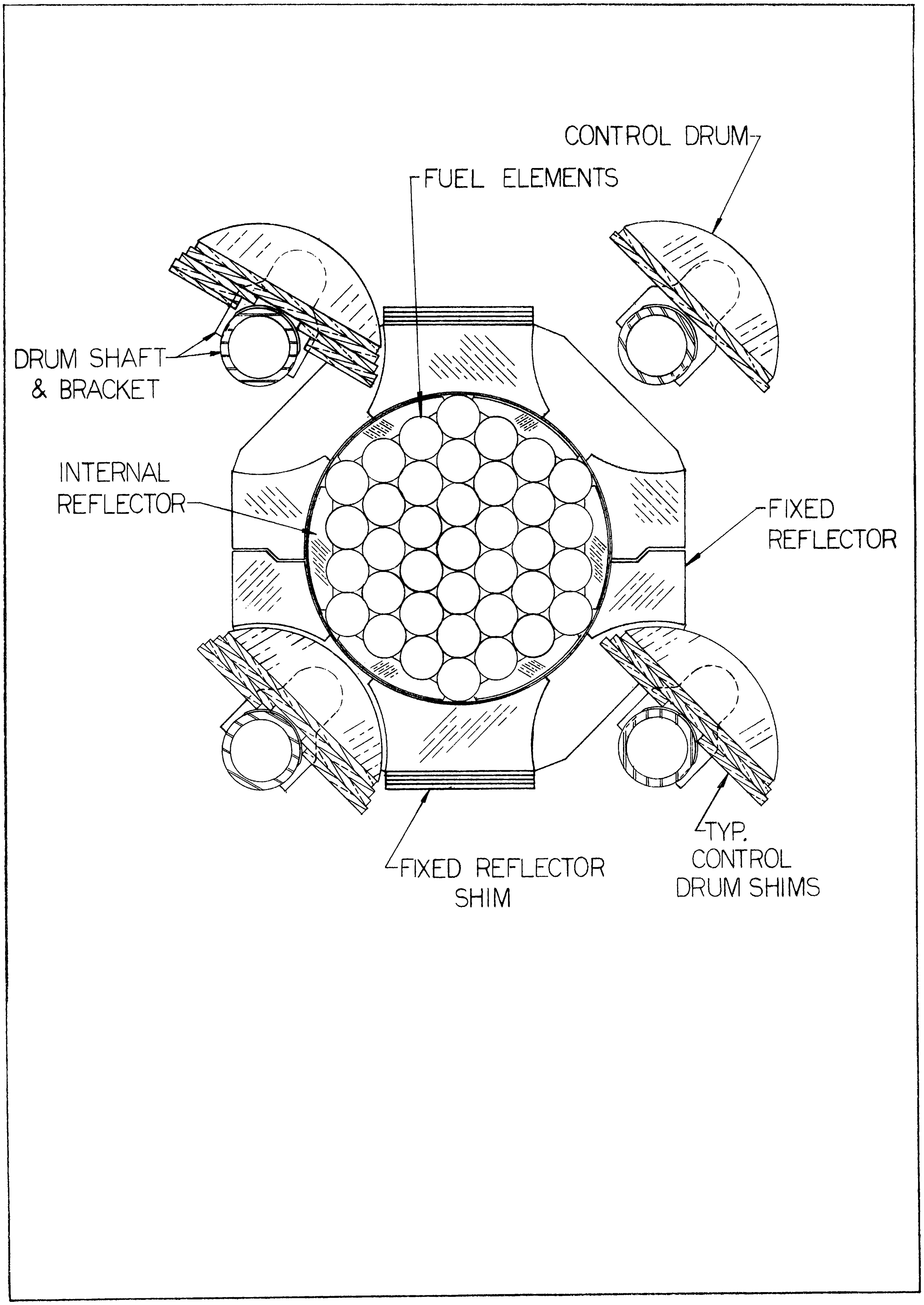


Fig. 5 - Reactor Cross Section

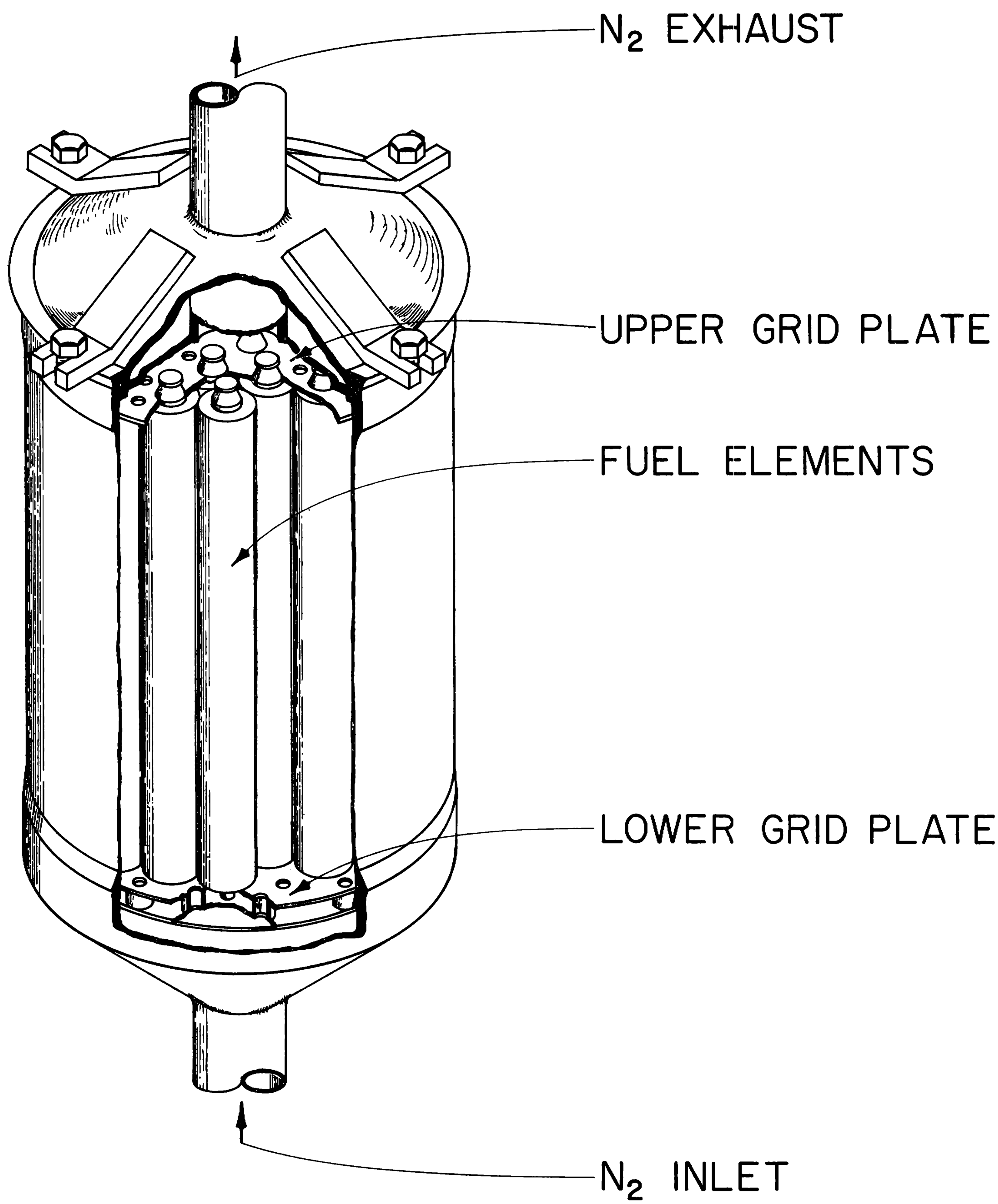


Fig. 6 - Vessel Pictorial

will not be on. For the static and transient tests, the void areas in the vessel will be filled with nitrogen, and for the destructive test the reactor will be charged with NaK.

3. Reflector

A fixed Be reflector surrounds the reactor. Four rounded slots, 90° apart, have been provided in this reflector to permit rotation of the Be drums for reactor control. The nominal thickness of the reflector is 1.875 in.; however, 0.125 in. shims can be added to increase the thickness to 2.5 inches.

4. Control Drums and Drives

The control drums are the same as those used on the SNAP 2/10A flight machine. A drum is comprised of a sector of a right circular cylinder of 3-1/2 in. radius and has a maximum nominal thickness of 1-3/4 in. (see Fig. 5) and can be shimmed to 3-5/8 inches. Each control drum is bolted to extension brackets fixed to a stainless steel drive shaft. The lower end of the shaft is splined with one tooth blank, and this spline fits into one of the couplings on the cross assembly. The cross assembly serves as the lower support bearing for the drum shafts. The upper drum shaft bearing is attached to the upper plate of the structure.

Stainless steel cooling fins are welded to the drum shaft below the upper bearing and to the lower coupling. Blowers force air across these fins to protect the bearings from the high temperature. Tungsten counterweights are used above and below the drums to minimize the unbalanced loads on the bearings when the drums are being rotated at high speeds.

Normal positioning of the drums will be accomplished by rotating the drums to their proper positions with an electro-mechanical system consisting of a geared drive motor, a gear reducer, an electromagnetic tooth clutch, and the drive shaft (see Fig. 7). The motor is a permanent magnet, dc motor having speed control through supply voltage control. Armature speed at a drum speed of 1°/sec is 20,000 rpm. Armature disassembly occurs at approximately 30,000 rpm due to

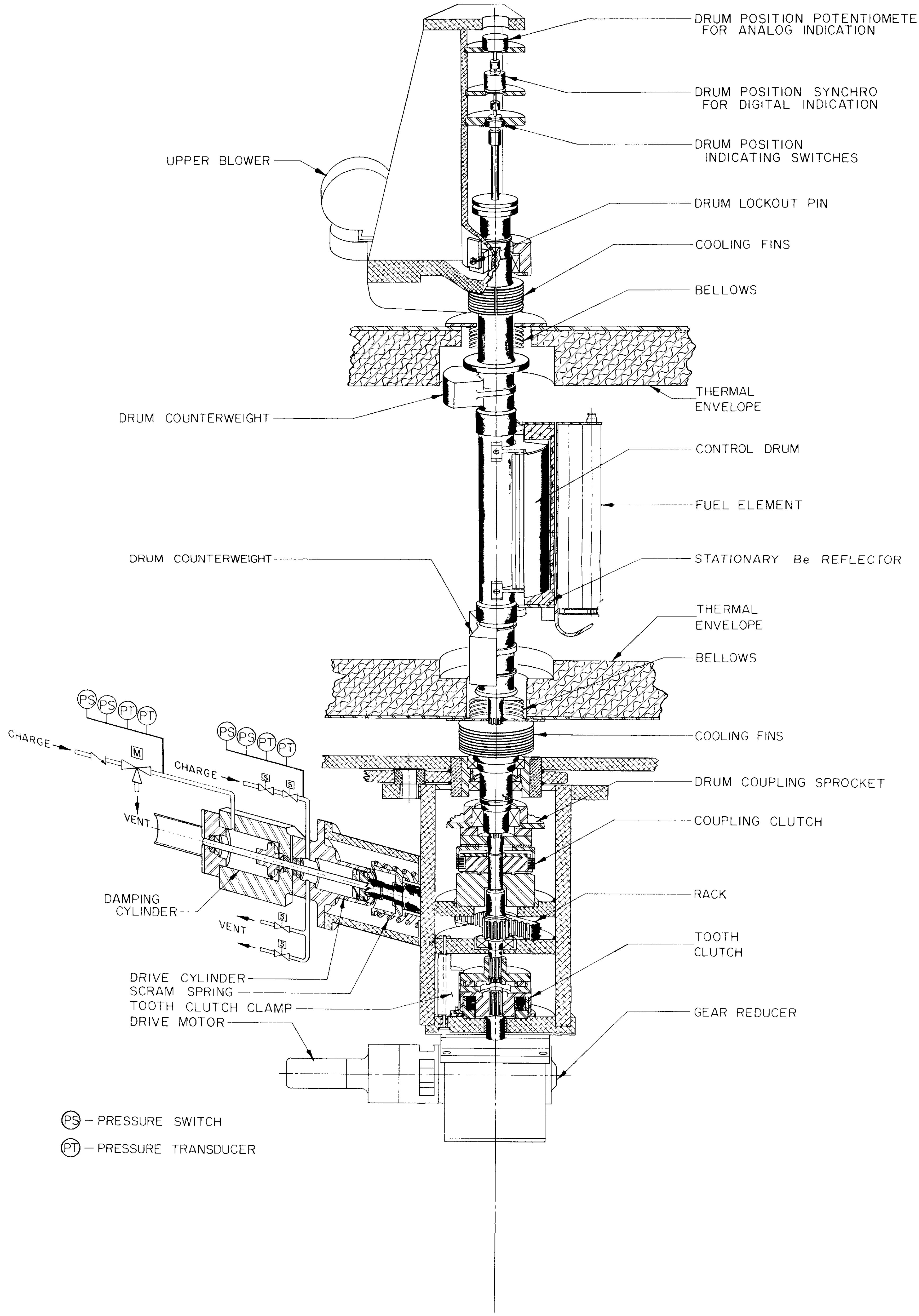


Fig. 7 - Stepper Drum and Drive Assembly

centrifugal forces, so the maximum drum insertion rate is limited to 1.5°/sec/drum. Armature speed is reduced by a 2000:1 ratio to the output shaft by a gear-head integral with the motor. The motor is connected to a right-angle worm-gear reducer with a 60:1 ratio which gives a total reduction of 120,000:1 through the motor and gear reducer. The gear reducer cannot be driven backward by applying a load to its output shaft.

An electromagnetic tooth clutch is used to couple the drum drive shaft to the gear reducer output shaft. This tooth clutch is used to disengage the drum drive shaft from the motor-gear reducer so that the drum can be scrammed and inserted rapidly for step and impulse inputs. A tooth clutch is used to provide positive transmission of torque when the clutch is engaged (power on) and a complete disengagement with power off. The clutches used for the two step reactivity input drums are rated for 300 ft-lbs of torque, and those used for the two impulse reactivity input drums are rated for 1200 ft-lbs of torque. The wide difference in torque rating is due to the large difference in force required to accelerate the step and impulse drums. A one-inch diameter shaft is used between the tooth clutch and the coupling to the drum shaft on the two step drums, and a 1-1/2 inch diameter shaft is used on the impulse drums. Both ends of the shafts are splined to fit into the tooth clutches and the couplings.

The equipment required to perform a step input includes the drive cylinder and piston, rack and pinion gear train, damping cylinder and piston, electromagnetic tooth clutch, tooth clutch clamp, and the valves, pressure switches and pressure transducers necessary to charge the cylinders with the proper pressure.

The drive piston and cylinder, and the damping cylinder and piston are designed as integral units. A common shaft connects both pistons and a common housing serves both cylinders. The cylinder diameter for the drive piston is 2 in., and the cylinder diameter for the damping piston is 2-3/4 inches. One end of the drive cylinder is open to the atmosphere since it is only driven in one direction, but both ends of the damping cylinder are closed since the damping piston is double acting. On one end of the damping cylinder the energy of the driving piston and drum is absorbed, and on the other end of the cylinder the energy of the scram spring and drum is absorbed. The damping medium in the damping cylinder is compressed N_2 which is also used to move the driving piston.

A rack is connected to the driving piston and mates to a pinion gear which is machined integral with the drum drive shaft. The rack and pinion transform the linear motion of the piston into the rotary motion of the drum drive shaft. Guide rollers on the back side of the rack keep the rack and pinion engaged at all times.

The tooth clutch is used during step inputs to resist the force of the charged driving piston and cylinder. The tooth clutch is released to insert the drum to the maximum reactivity position. A tooth clutch clamp is used to prevent an inadvertent clutch release if a power failure should occur during the time the driving cylinder is being charged with N_2. The tooth clutch clamp surrounds the tooth clutch with a ring which can be rotated approximately 5° by two arms 180° apart attached to solenoids. Three cams, anchored and pivoted to the frame and connected to a toggle link which is mounted on the ring, have a lip of about 1/8 in. that extends over the tooth clutch in the clamped position. To release the clamp, the solenoid is energized and rotates the ring about 5°. The toggle link, rotating with the ring, pivots the cam about its anchor point and the protruding lip of the cam swings away from the tooth clutch. Power is required to engage or release the tooth clutch clamp, and a position switch on the clamp is used as an interlock to prevent pressurizing the driving cylinder if the clamp is in the released position. A 0.001 in. clearance between the tooth clutch and the lip of the clamp is used to insure that the tooth clutch is fully meshed before the clamp is engaged. When the driving piston is fully charged, the torque on the tooth clutch is about 80 ft-lbs.

The driving cylinder is equipped with two solenoid valves in series on the inlet pressurizing line, and two solenoid valves in parallel on the vent line to prevent a malfunction due to a valve failure. Two pressure switches are used in the interlock circuits to prevent inserting the drum if the cylinder pressure is too high or too low, and two pressure transducers are used to indicate the cylinder pressure on the control console.

In the damper cylinder pressurizing line, a motor driven three-way valve is used to provide for pressurizing or bleeding the cylinder after the reactor has scrammed. This valve has position indication on the console. During reactor operation, the valve will be positioned such that the regulated bottle reservoir is connected to the cylinder. Two pressure switches are used in the interlock circuits to prevent any drum insertion unless the damper cylinder is pressurized, since machine damage could occur if a step drum is scrammed without the damper cylinder being pressurized. The driving cylinder and damping cylinder are rated for 2200 psi. Driving pressure will be 500 psi and the pressure in the damping cylinder is 1000 psi.

Three switches are used to indicate the rack position. These positions are rack out (180°), rack in (0°), and rack at the 45° arming position. Both step drums are limited to 180° of rotation, from full-out (180°) to full-in (0°), by the length of the rack and by the head end and crank end of the damper cylinder.

Both step drum drives are identical except for the couplings to the drum shafts. One of the couplings has an electromagnetic disc clutch between it and a sprocket and the other coupling has only a sprocket. The two drum drives are connected by a chain on the sprockets. The purpose of the clutch and chain is to synchronize the timing of the drums when a two drum step insertion is performed. The clutch is disengaged during one drum steps.

The equipment required to drive an impulse drum (Fig. 8) includes the driving and damping cylinder and piston assembly, the rack and pinion gear train, the electromagnetic tooth clutch, the tooth clutch clamp, and the valves, switches, and transducers used for pressurizing and venting the cylinder. Of this equipment, the rack and pinion is similar to that described for the step drum drives, and differs only in size. The electromagnetic tooth clutch has been described earlier, and the tooth clutch clamp is the same as that described for the step drive, differing only in size.

The normal direction of drum insertion for step transients and operational drum positioning is clockwise in the positive sector with travel limited to the arc from 180° to 0°. Of this 180° of drum rotation, only 135° (+135° to 0°) of the drum has significant worth. This is true in the negative sector also (-135° to 0°). For impulse tests the drums are driven counterclockwise through an arc of 450° (225° on both sides of 0°) with the 90° dead-band of the drum between +135° and -135° used to accelerate the drum and the dead-band between +135° and -135° used to stop the drum and absorb the energy of rotation.

The impulse driving piston and cylinder assembly combines the driving and damping function into one double-acting piston and cylinder 5-1/2 in. in diameter. The piston is driven to one end of the cylinder by the motor through the rack and pinion. An annular reservoir around the cylinder is charged to the proper pressure and this pressure is applied to the driving side of the piston. When the tooth clutch is released, the piston starts to accelerate and transmits a rotary motion to the drum through the rack and pinion. At a point in the cylinder corresponding to 90° of drum rotation, N_2 begins to be bypassed through ports in the cylinder to the damping end of the cylinder. The bypass rates are such that during the time the piston is traveling through the ported area, the piston travels at essentially constant velocity. As the piston continues down the cylinder, it closes the ports and traps a volume of N_2 ahead of it. It is the compression of this trapped N_2 that cushions and stops the piston at the opposite end of the cylinder. A motor operated valve in the exhaust line is used to control the pressure release rates after the piston has closed the bypass ports. A small orifice in the piston relieves the pressure when the impulse is over.

The cylinder pressurizing line has two solenoid valves in series used for pressurizing, and four solenoid valves on two lines in a series-parallel arrangement for venting the cylinder. Two pressure switches with high and low pressure limits are used in the interlock circuits to prevent inserting the impulse too fast with high pressure, or too slow with low pressure. A pressure transducer is used to indicate the cylinder pressure on the console.

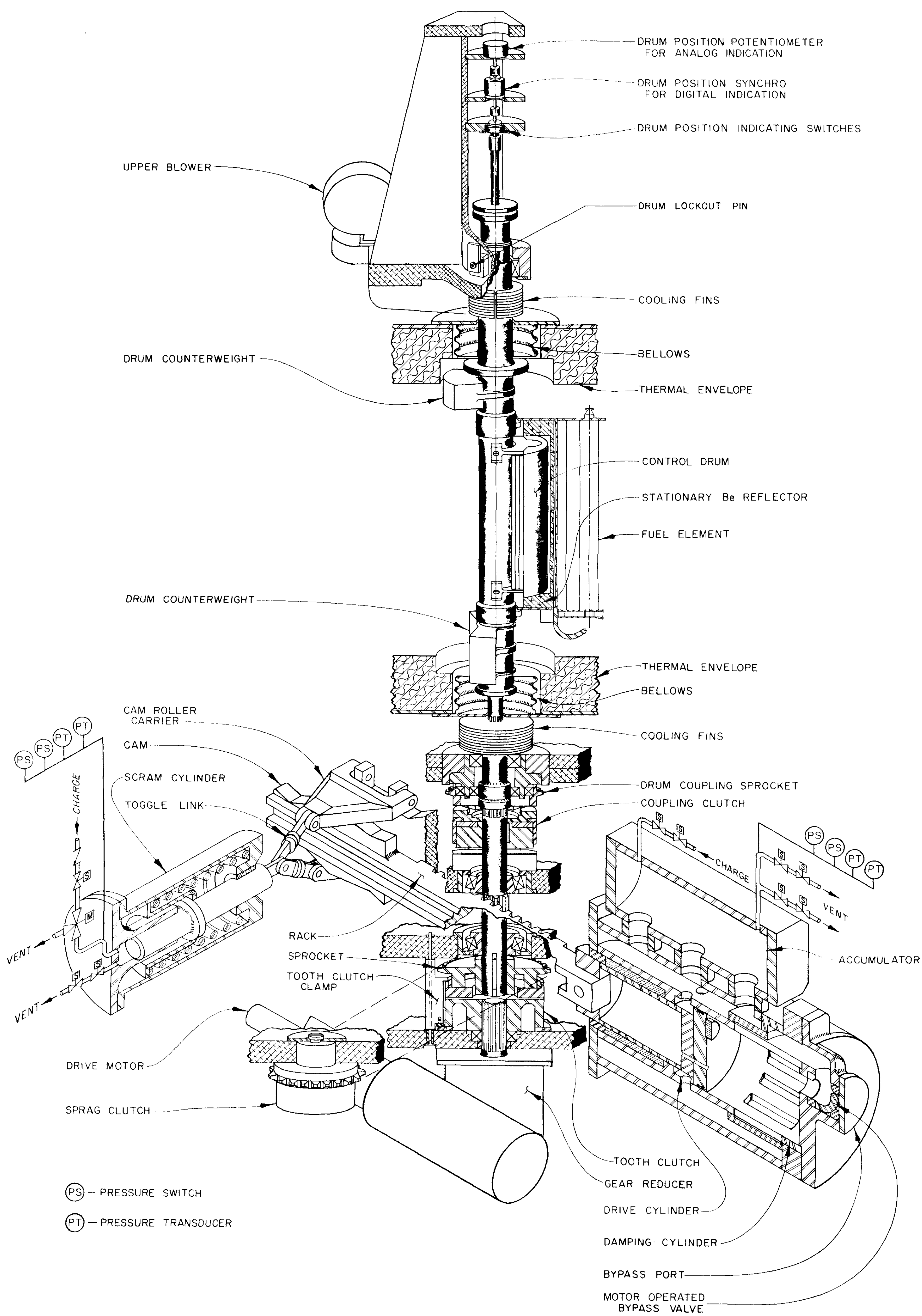


Fig. 8 - Impulse Drum and Drive Assembly

The pressure used to drive the impulse piston is 1250 psi. The acceleration on the drum due to the 10,000 lb driving force is 8220 rad/sec^2. The maximum torque on the tooth clutch is 833 ft-lbs, and the clutch is rated for 1200 ft-lbs.

In the event a two drum impulse input is to be performed, the coupling of the two drums is the same as for the step drums. With the exception of larger sized components, the operation is the same. As the drum reaches its limit of travel following an impulse test, it is possible for the energy stored in the damping cylinder to drive the drum back into the reactor. To prevent this, a Sprag clutch is coupled to the drum shaft with a chain and sprocket and is used to prevent drum rotation in a direction opposite the impulse insertion rotation. The Sprag clutch must be disengaged when the impulse drums are being inserted by the motors and requires power to change it from engaged to disengaged and vice versa.

The scram for the two step drums is accomplished by a spring which is connected to the rack. The released position of this spring corresponds to the full-out position of the drum (180°). The spring is compressed any time the drum is inserted. To scram the drums during normal operations or impulse tests, the tooth clutch is released, and the spring drives the rack to return the drums to the full-out position. To scram the drums after a step input, the driving cylinder pressure is dumped in addition to releasing the tooth clutch. The step drums can only be driven in the positive sector and always scram in the same direction (counterclockwise).

The scram mechanism for the impulse drums is considerably more complex because it must operate in both sectors, and must not scram the drum back through the maximum reactivity position from either sector. The mechanism used to accomplish this includes two linear cams integral with the end of the rack, cam rollers on links which are anchored on one end and connected to a toggle link mechanism on the other end, a connecting rod connected at one end to the toggle link and to a piston on the other, a cylinder, and a spring. The cams are on both the upper and lower surface of the rack and have a slope of about 23° on the side that corresponds to the positive sector and a slope of about 6° on the side of the cam that corresponds to the negative sector. The horizontal component of the force applied to the cam by the roller through the toggle link, connecting rod, piston and spring gives the movement to the rack required to scram the drum. The scram mechanism is only used during positioning operations and step input tests. As the drum is positioned during these tests, the pinion drives the rack, the rollers ride higher on the cam which in turn compresses the spring. This keeps a force on the cam and, when the tooth clutch is released, the drum is scrammed.

There are two limitations associated with this mechanism--the cam dead spot and the need for invalidating the scram during the impulse input test. At the peak of the cam there is no horizontal component on the roller and if the drum is stopped at this point, no scram is possible. Since the normal working sector for the impulse drums is the positive sector, the dead spot has been located at about 10° beyond the maximum reactivity position in the negative sector. This minimizes the possibility of the drum being stopped on the dead spot. Invalidating the scram during an impulse test is necessary because the driving force is too high if the scram mechanism is engaged during the insertion. The scram lockout is accomplished by pressurizing the scram lockout cylinder to drive the piston and connecting rod. This motion causes the rollers to be lifted off the cams. A switch is actuated by the connecting rod when the scram has been invalidated. Two pressure switches are used in the interlock circuits to prevent charging the driving cylinder if the scram is not invalidated. A motor-driven, position-indicating, three-way valve is used to vent or pressurize the cylinder. The pressure required to lift the rollers off the cams is 500 psi.

5. Position Indication

The position indicating devices are mounted on the upper portion of the drum shaft. Three types of position indication are used: a digital position indication, an analog position indication, and a limit indication.

Digital indication is obtained from a servo-synchro system which has an overall accuracy of ±0.1°. The readout from this system is located on the control console.

The analog indication is obtained from a potentiometer mounted on the drum shaft. This system has an overall accuracy of ±1°. The signal from the potentiometers is recorded on a panel mounted adjacent to the console.

The drum position is also indicated by cam operated switches, and on all drums the maximum reactivity position (0°) and the minimum reactivity position (180°) are indicated on the console by lights. On the impulse drums, an indication is given to show whether the drum is in the positive or negative sector.

6. Reactor Characteristics

The reactor characteristics are summarized in Table III-1.

D. Control System Description

1. General

The reactor control system for the SNAPTRAN 2/10A-1 test is designed to provide flexible control capabilities for static measurements as well as control capabilities for rapid reactivity insertions required in transient tests.

Since both the static and transient reactivity control is provided by each drum, the control system features a high degree of functional layout, system diagnostic indication and interlock control to allow the operator to place maximum attention on the overall testing operation. In addition, the control system provides for administrative checks and surveillance through keyswitch interlocks. Finally, automatic programing equipment is used to minimize the time during which the reactor is held in an armed state. The details of the reactor control system are discussed below.

2. Control Console

The console is designed to enable the operator to maintain visual surveillance over all control room operations. The console has a sloping front design and accommodates standard 19-inch panels. Three pair of rectangular sections are separated by two pair of 45° sections which serve as corners, so that the overall appearance is that of three

TABLE III-1

SNAPTRAN 2/10A-1 REACTOR CHARACTERISTICS

Reactor Design	
Fuel Elements	
Number	37
Fuel diameter, inches	1.212
Cladding material	Hastelloy N
Cladding diameter, OD, inches	1.25
Cladding thickness, inches	0.015
Fuel alloy, w/o U in Zr	10
Degree of hydriding, $N_H(10^{22}$ a/cc)	6.5
Active fuel length, inches	12.25
Internal Reflectors	
Composition	Be
Cladding	None
Lower Grid Plate	
Material	Hastelloy C
Thickness of each sheet, inches	0.062
Overall thickness (including spacers), inches	0.500
Upper Grid Plate	
Material	Hastelloy C
Thickness, inches	0.125
Core Vessel	
Material	316 SS
Internal diameter, inches	8.875
Thickness, inches	0.032
Reflector Control Elements	
Number	4
Material	Be
Nominal thickness, inches	1.875 minimum 3.500 maximum
Nuclear Parameters	
Uranium loading, Kg U-235	4.75
Mean fission energy, ev	0.18
Average isothermal temperature coefficient, ¢/°F	-0.22
Prompt (fuel) temperature coefficient, ¢/°F	-0.05
Mean prompt neutron lifetime, 10^{-6} sec	6.5
Initial cold excess reactivity, $	
2.250 in. reflector thickness	~ 6.00
Total control drum worth, $	~ 9.00
Effective delayed neutron fraction	0.008

sides of a square. The two center sections, as shown in Fig. 9, contain the operational controls. The left center panel, which is the precision positioning panel, accommodates motor drive controls, drum position indicating meters and lights, count-rate meters for reactor startup, and the scram reset and console scram buttons. Switches and pressure indicators used for controlling the pneumatic drives in initiating transients are in the right center panel, or transient panel. The corner sections shown to the left contain controls for console power, indicators for drive motor currents, indicators and control for drive motor speeds, reactor leveling switches, and keyswitches for bypassing certain control interlocks. Switches for starting and resetting the timer program sequence and lights which indicate the program sequence operation are located in the 45° section to the right of the transient panel. The timer, with the interval setting controls, is in the first rectangular section to the right of the corner. Controls for communication between the control room and the reactor area and other parts of the facility are located in the right half of the program sequence panel.

The end rectangular section, which is adjacent to the timer, contains the receiver for a closed circuit television monitor which is used to survey the Test Cell. The rectangular section adjacent to the left corner contains a special purpose pressurized ink strip-chart recorder with ten channels, eight of which are capable of 1% accuracy and 4 ms full-scale rise time, the other two providing time reference markers. This "events recorder" has 12 chart speeds ranging from 3 mm/min to 20 cm/sec. Four channels will be used for recording drum position as indicated by the shaft-mounted potentiometers; four more channels will be available for other pertinent data such as magnetic tooth clutch currents, solenoid valve currents, log power, etc. The events recorder will always be recording whenever the control system is in operation in order to provide a permanent record of all drum movement. Normally the chart speed will be appropriate for recording drum speeds of one degree per second or less, as used in normal motor drive operation. The chart speed will be increased to the maximum speed for effective recording of rapid drum motion during transient initiation, programed scrams, or normal operational scrams.

The left end rectangular section contains a strip-chart recorder for recording reactor temperature data or other process variables.

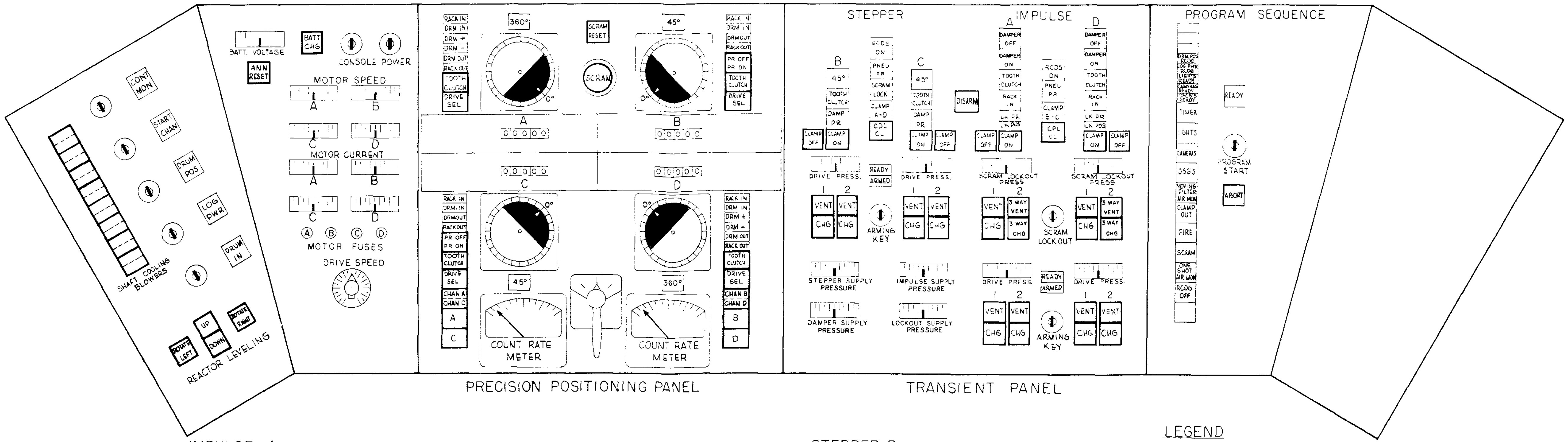
Logarithmic and linear operational amplifiers and chart recorders for nuclear power level are mounted on a rack directly above the two operational panels in the center of the console.

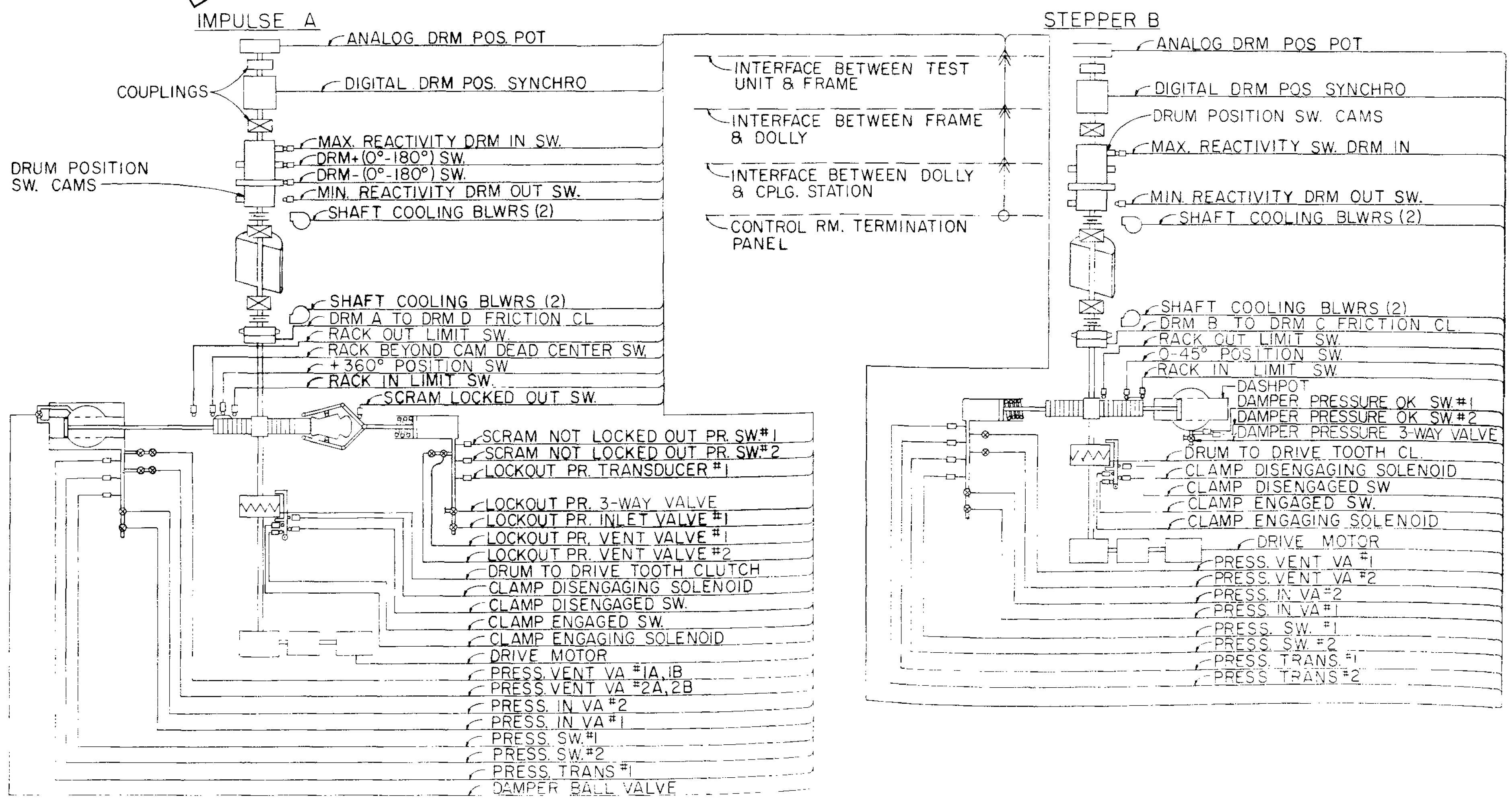
Key operated switches are provided on the console to control: (1) power to the console, (2) arming of the stepper drives, (3) lockout of scram on the impulse drives, (4) arming of impulse drives, (5) starting of program sequence timer, (6) driving of impulse drums past the fully inserted position into the negative region, and (7) bypassing of certain drum insertion interlocks such as the one requiring that specific data recorders be operating before drum manipulations.

All keyswitches are keyed differently, permitting separate administrative control of each function. Console power is controlled by two keyswitches in series, so that permission is required from health and safety supervision as well as nuclear operation supervision before the control system may be energized.

Except for the red mushroom head scram button and the keyswitches, all lights and switches on the control console are Minneapolis-Honeywell operator-indicator type, rectangular illuminated push buttons. These units combine the functions of control and indication and contribute to compactness of panel design and simplicity in operation. Where indication only is required, as in indicating the state of a limit switch on a drive mechanism, the pushbutton feature is left inoperative. The lighted "pushbutton" surface of each unit, whether used as a switch or only for indication, is engraved with a legend for identification of its function.

The switches are all multi-pole double-throw units. Two types of actuators are provided. One type of actuator changes a switch from one "throw", or state, to the other by means of a ratchet on successive operations, maintaining either state of a circuit. The other type of actuator is used for "momentary" circuits, in which one throw of the switch is maintained by the actuator in its normal position and the other throw is maintained only while the actuator is held in its depressed position by the operator.





LEGEND

CHAN	CHANNEL
RACK IN	RACK NORMAL INSERTION MECHANICAL LIMIT
DRM IN	DRUM A OR D AT MAX. REACTIVITY- DRUM B OR C AT STEPPER STOP LIMIT
DRM +	DRUM IN REGION CLOCKWISE FROM MIN. TO MAX. REACTIVITY
DRM -	DRUM IN REGION COUNTERCLOCKWISE FROM MIN. TO MAX. REACTIVITY
DRM OUT	DRUM AT MIN. REACTIVITY POSITION
RACK OUT	RACK NORMAL WITHDRAWAL MECHANICAL LIMIT
DRIVE SEL	DRIVE MOTOR SELECTOR
RCD ON	DRIVE POSITION AND NUCLEAR LOG POWER RECORDERS ON
VENT	ARM OR SCRAM LOCKOUT PRESSURE RELEASE VALVE SOLENOIDS
CHG	ARM OR SCRAM PRESSURE INLET VALVE SOLENOIDS
LK PR	SCRAM LOCKOUT PRESSURE
LK POS	SCRAM LOCKOUT POSITION
DRM POS RCDG	POSITION DRUM RECORDING
LOG PWR RCDG	LOG POWER RECORDING
OSCG'S	OSCILLOGRAPHIC RECORDERS
AIR MON	AIR CONTAMINATION MONITORS
PNEU PR	MANIFOLD PRESSURE CORRECT

NOTES

1. STEPPER C- SAME AS STEPPER B EXCEPT CPLG CLUTCH OMITTED
2. IMPULSE D- SAME AS IMPULSE A EXCEPT CPLG CLUTCH OMITTED
3. HEAVIER OUTLINED LIGHTS REPRESENT ILLUMINATED PUSHBUTTON SWITCHES; OTHERS ARE INDICATOR LIGHTS

☐ ILLUMINATED PUSHBUTTON SWITCH
☐ INDICATOR LIGHTS

Fig. 9 - Control Panel Layout and Control Schematic

The lights indicate closed or open circuits by being on or off except where continuous illumination is desired for easy identification, in which case change of state is indicated by color. Color indication is also used where more than two states exist such as in a timer sequence where a light will be off until a circuit is "selected", at which time it come on white, and while the circuit is "operative" during the program sequence the light turns to red.

Where both "throws" of the switches are used with on-off light indication, the illuminated surface is divided into separately lighted halves. Each unit may be equipped with as many as four bulbs, so that at least two bulbs in parallel are used for all indications. Thus, erroneous indication resulting from bulb burnout can occur only when a pair fails simultaneously. A single burnout is evident from the unbalanced appearance of the indicator, and must be remedied immediately to prevent the hazard of its becoming a double burnout.

Indicator lights are arranged in vertical rows on the console panels to show logical sequence of interlocks which must be accounted for in the various operations in preparation for a transient test. For example, in order for a stepper drive to be armed, the "damper pressure", "tooth clutch", "45°" position, and tooth clutch "clamp on" lights must come on, indicating that the corresponding interlocks are satisfied in sequence. When all of these lights are on, the "ready" light comes on, indicating that operation of the "arm" keyswitch will pressureize the drum drive cylinders.

In four special cases, lights are used for identification of a switch. For example, when the system scrams, the scram reset switch is lighted calling the operator's attention to its location. The annunciator acknowledge switch, the disarm switch, and the timer sequence abort switch function similarly.

3. Console Power

The main power supply to the console (Fig. 10) is a storage battery consisting of 39, 50-amp-hr lead-acid cells connected in series-parallel to float on a 28 volt constant voltage charger. The charger is supplied from commercial ac power and may be shut off during operations where freedom from electrical noise is required.

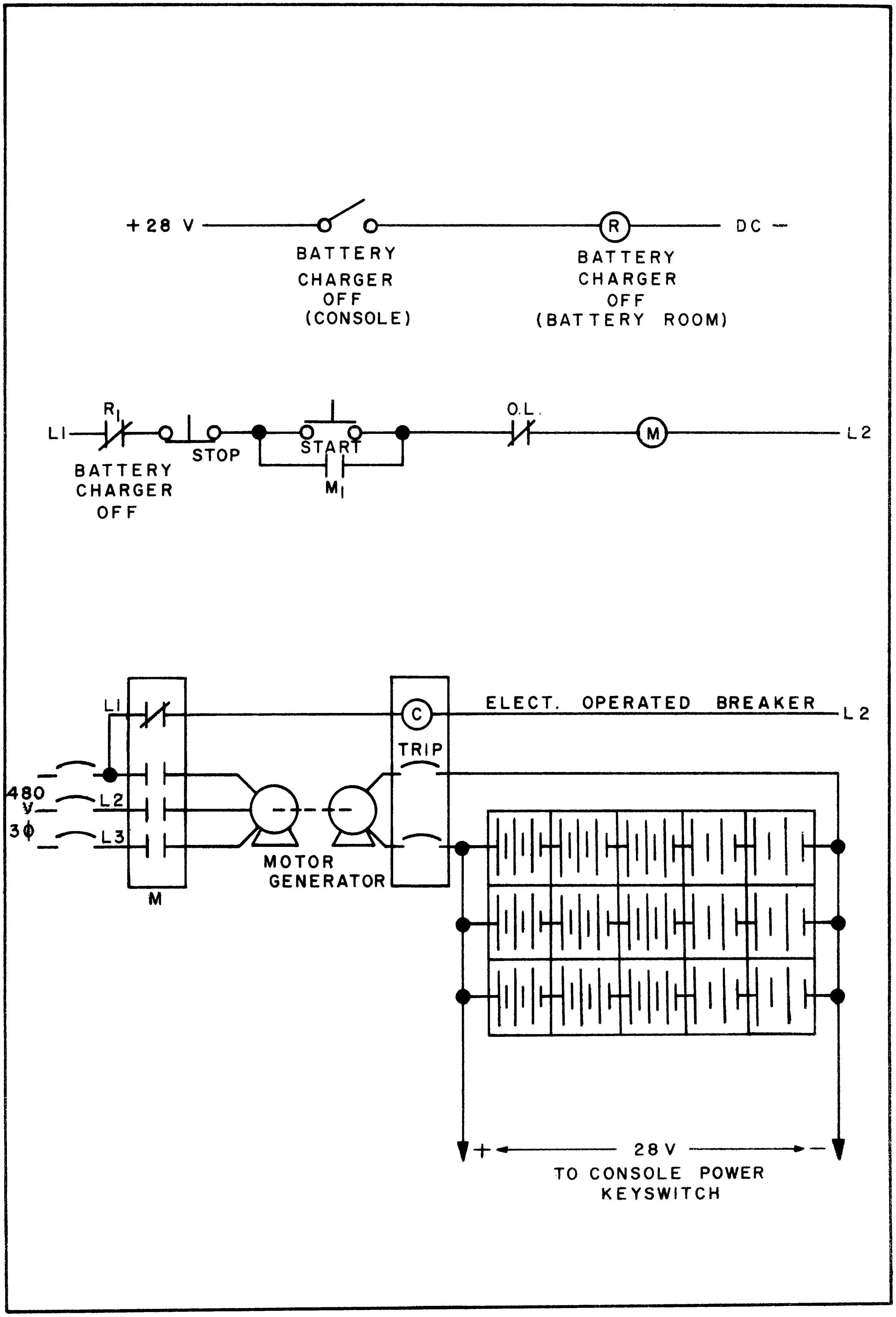


Fig. 10 - Console Power Supply

Satisfactory voltage must be maintained for the magnetic clutches which are rated at 24 volts. In the event that the battery voltage drops to 24 volts, an alarm sounds to alert the operator. The series-parallel arrangement makes this power supply virtually failure-free. Cells may be replaced in groups of two or three without interruption of service.

4. <u>Auxiliary Power</u>

Auxiliary 60 cps ac power will be provided by a gasoline engine driven alternator, with automatic transfer switch for selecting either auxiliary power or commercial power.

Commercial power outages, although rare, are seldom preceded by any warning, whereas an engine driven unit may be expected to show symptoms prior to a failure. For this reason, operational and experimental reactor instrumentation, reactor control drum shaft cooling blowers, and other vital apparatus requiring 60 cycle power will be powered by the engine driven unit.

5. <u>Scram Systems</u>

All drum insertion is opposed by heavy springs which provide scram when the tooth clutches coupling the drum shafts to the worm drives are disengaged.

Current for the tooth clutches is supplied by the "scram buss". The scram circuitry (Fig. 11) connecting the scram buss to console dc power, consists of a series of normally closed manual push buttons at various locations, an ac power failure interlock relay, and two master scram relays in series. The coils of the master scram relays are in parallel, connected to the scram buss through the normally closed contacts of a "sigma" scram-signal relay, thus forming a self-holding circuit. The scram circuit is reset by shunting the series contacts of the scram relays with a momentary normally open console push button labeled "scram reset".

One scram button is mounted on the console; several are mounted in the reactor area, and provisions are made for portable scram buttons in the control room. The reactor area scram buttons are of maintained position type which require deliberate reset after use, prohibiting the console operator from effecting scram reset without first investigating the cause. The sigma scram relays, which may be replaced directly by an amplifier when fast response is demanded, accepts scram signals from sources such as power level or period safety trip instrumentation or a program sequence timer.

A "Log N" period and level amplifier with appropriate trip circuits is provided for safety scrams. This will be set to scram the reactor if power level exceeds 1000 watts or if the period of e-fold power rise is shorter than ten seconds. These restrictions, effective during normal static operation, will be bypassed automatically by the program sequence timer used to initiate transient tests.

Each tooth clutch is controlled by a selector circuit containing a holding relay and a sigma relay similar to the master scram circuit. Tooth clutches are energized individually by momentary console push buttons which shunt the holding relay contacts. A tooth clutch can be dropped out by scramming, by turning off console power, or by a signal to the appropriate sigma relays.

No interlocks are provided to prevent operation of the drive motors when the tooth clutches are disengaged. Operating the motors unloaded presents no overspeed hazard since motor speed is primarily dependent upon armature voltage rather than upon output torque.

6. Drum Motor Drive Control

In order to effect drum motion from the motor drives the scram circuit must be made operative and the appropriate tooth clutches must be engaged. A single pistol grip reactivity increase-decrease switch on the lower portion of the precision positioning panel provides on-off control of forward and reverse voltage to all of the drive motors. "Drive selector" switches, located near the respective drum position indicators, connect motor selector relays to the reactivity increase-decrease switch. These motor selector relays, which have double-throw contacts, serve a dynamic braking function. The relays

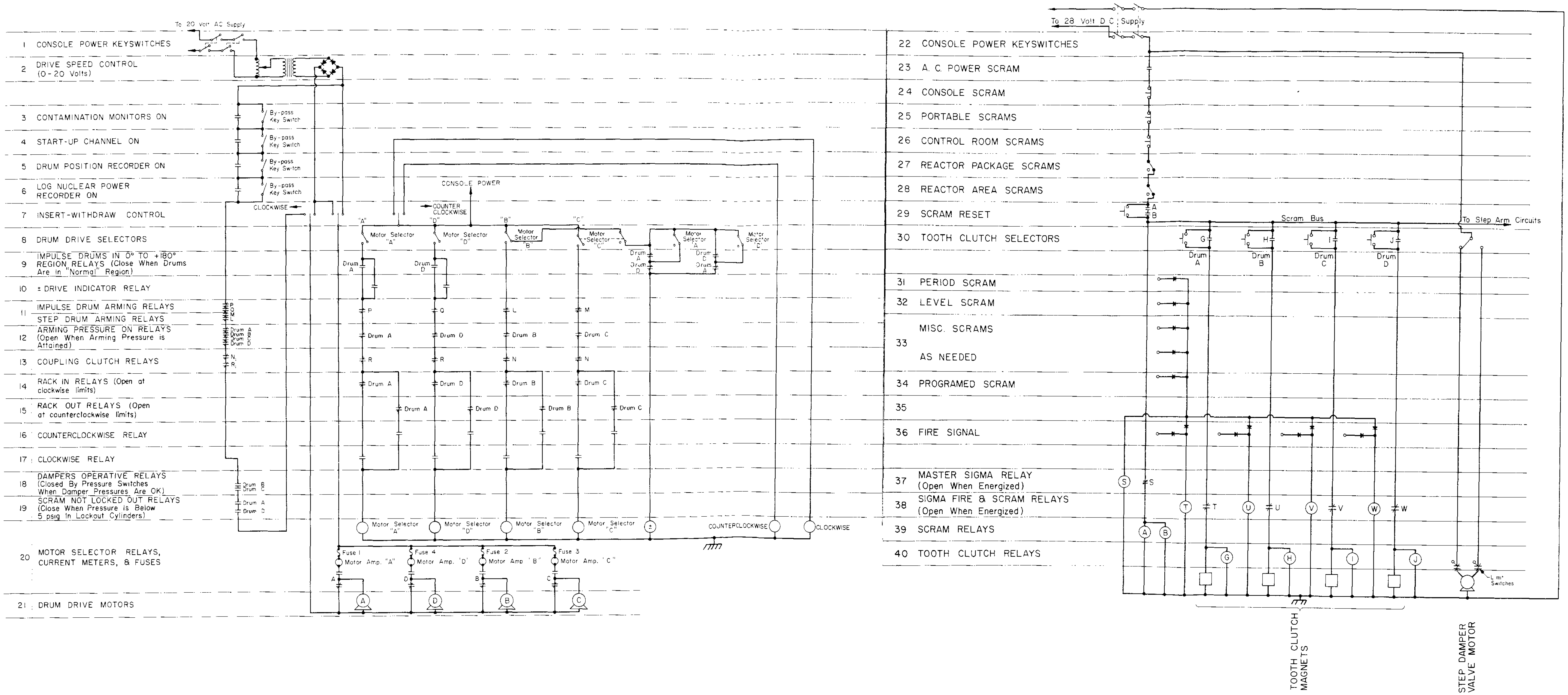


Fig. 11 - Precision Position and Scram-Fire Logic Control System Schematic

also permit individual or simultaneous operation of the drive motors. Dynamic braking, to minimize motor coasting, is accomplished by short-circuiting the motor armatures after drive voltage is removed. (static braking is inherent in the 120,000:1 self-locking worm gear speed reducers). The variable transformer, rectifier, reactivity increase-decrease switch, interlock and control relays and motor armature current ammeters comprise the basic motor drive circuitry.

The motor drive circuits contain a number of interlocks which are provided between the rectifier and the increase-decrease switch in the reactivity increase circuit. Since the impulse drums may be operated on both sides of the maximum reactivity position, the motor selector circuits are interlocked to prohibit insertion of any drum while another is being withdrawn. Interlocks which apply to all drums are placed directly in the motor circuit ahead of the selector relays. Interlocks which apply to specific drums operate indirectly through the control circuits of the appropriate motor selector relays. The reactivity increase circuit is interlocked to prevent operation if contamination monitors are inoperative, if startup channels are inoperative, if the drum position events recorder is inoperative, if logarithmic nuclear power recorder is inoperative, if any drum is armed, if either pair of drums is coupled, if either of the stepper drive damper pneumatic pressures is insufficient, if either of the impulse drive damper valves is not in the scram mode, or if any pneumatic pressure is indicated in the scram lockout cylinders.

Each motor selector relay circuit is interlocked to prevent motor operation if the drum is armed, coupled, approaching a mechanical limit, if the tooth clutch is clamped, or if the Sprag clutch is operative.

7. Pneumatic Drum Drive Control

a. Impulse Transients

It is necessary to constrain the scram springs on the impulse drums since the pneumatic drives for these drums are not capable of imparting the required accelerations with the scram springs acting. (See Section III-C-4). It is also necessary to ensure that nothing disarms, or partially disarms, the impulse drives during the firing of an impulse transient. Such an occurrence would result in prolonging the impulse, and damage might occur to the reactor. Hence, to prevent interference from the scram mechanism, a scram lockout pneumatic cylinder is incorporated to maintain the scram springs fully compressed and to maintain the associated mechanical linkage completely disengaged from the drums during impulse tests. Also, to prevent inadvertent disarming the impulse drive is charged, or "armed", in the "cocked" position, with nitrogen gas at 1250 psig. The valves are then closed and no further charging or venting can take place until completion of the transient.

Consistent with these requirements, all of the pneumatic control valves of the impulse drives are normally closed solenoid valves, requiring deliberate actuation both to arm and to disarm the drives. Control power is not required from either the console power buss or the scram buss to maintain the impulse drives in either the armed or disarmed state.

In order to set up for an impulse test, the impulse drum will be driven into the negative sector and positioned at the point where the reactor is intended to become critical for the predetermined impulse magnitude. The reactor will be brought to criticality by appropriate manipulation of the remaining drums in the positive sector. The impulse drum(s) will then be "cocked" (driven to the -225° position), "armed" [pneumatic drive(s) charged with nitrogen], and "fired" [tooth clutch(es) released].

Driving the impulse drums into the negative sector requires use of a "drum in bypass" keyswitch. In normal static test operation, "drum in" limit switches act as limits on drum insertion by motor drive. The withdrawal circuits ordinarily facilitate rotation back into the positive sector. For the case of the impulse drums, however, alternative withdrawal into the negative sector from the "drum in" limit is permitted by operation of the "drum in bypass" keyswitch enabling critical positions to be determined and the drums to be cocked and armed for impulse transients.

The "drum in bypass" operates only while the "drum in" switches are actuated and since only one drum can be in this position at any one time a single bypass keyswitch of momentary type, serves both impulse drums.

Except for bypassing the "drum in" limit, determining criticality and cocking of the impulse drums are essentially static operations governed by static test interlocks. However, additional conditions must be satisfied before arming of the drums is permitted, and appropriate interlocks are included in the arming circuitry to account for these conditions, as discussed below.

A relay must be energized by signals from the drum position and data recorders indicating a ready or operating condition. Pressure switches must indicate that nitrogen supply pressures to the impulse drives and the step dampers are adequate. Each tooth clutch clamp is equipped with switches for separate indication of engagement and disengagement. The "disengaged" switches must be actuated and the "engaged" switches must be not actuated. Tooth clutches must be energized on the drum(s) to be armed. Indication of tooth clutch engagement is obtained from relays operated from the clutch magnet circuits. The "rack in" limit switch(es) must be actuated on the impulse drum(s), indicating that the cocked position has been attained (-225°).

When the above interlocks have been satisfied, arming may proceed. (See Fig. 12). If a two-drum impulse is to be initiated, the drum-pair coupling clutch is energized. The tooth clutch clamps are then engaged. Interlocks prevent energizing the coupling clutch after either tooth clutch is clamped. Other interlocks prevent arming either impulse drum if the other impulse drum is clamped unless the coupling clutch is energized. With the drum(s) secured in the cocked position by clamped tooth clutch(es), the scram lockout cylinders may be charged, constraining the scram spring(s) and disengaging corresponding linkage from the cams on the drum drive rack(s).

To accomplish scram lockout, a spring return keyswitch must be held actuated, opening the inlet solenoid valve(s) and permitting the motor driven 3-way valve(s) to be driven to the charge position. The 3-way valve motors are controlled by console push buttons which must be actuated in conjunction with the keyswitch. When each lockout cylinder is fully charged, a switch on the corresponding scram linkage closes a relay, shunting the keyswitch, which may then be released by the operator thus completing the scram lockout interlock. The Sprag clutch activating solenoids operate in conjunction with the 3-way valve motors.

The impulse damper valve(s) is then opened by a switch on the console completing preparation for deceleration and locking of the impulse drum(s) at the end of impulse travel (+ 225°). Interlocks prevent this step until scram lockout is accomplished.

With the above interlocks satisfied, concurrent actuation of the impulse arming keyswitch and appropriate arming inlet valve push buttons causes admission of high pressure nitrogen to the impulse drive cylinder(s). Correct charging pressure is indicated by actuation of pressure switches on each pneumatic drive, which operate "armed" lights and a "ready" light on the console.

Operation of the sequence timer disengages the tooth clutch clamps, releases the tooth clutches which initiates the impulse transient, and subsequently terminates the resulting power excursion by scramming the remaining control drums.

b. Step Transient

The pneumatic stepper drives are capable of rotating the step drums against the force of the scram springs. It is therefore unnecessary either to constrain the stepper scram springs or to make any provision which restricts the disarming of the stepper drives. Any deviation from a prescribed step insertion can result only in a slower reactivity insertion, since all step insertions will consist of rotating a step drum to the maximum reactivity position. The magnitude of the power excursion is determined by the positions of the remaining drums. Hence, the stepper arming circuits have been designed to derive power from the scram buss, and control valves for the stepper pneumatic drives have been selected to disarm the drives automatically upon interruption of electrical power. Scramming always disarms armed stepper drives and if scram should occur after the tooth clutch clamps are removed for firing, all drums will scram.

In cocking the step drums (driving them to the 45° initial position) no special interlocks are required beyond those associated with static test operation. However, in order to arm, additional interlocks must be satisfied. The step arming logic is shown in Fig. 13. As in the case of the impulse test, the step drums must be in the cocked position

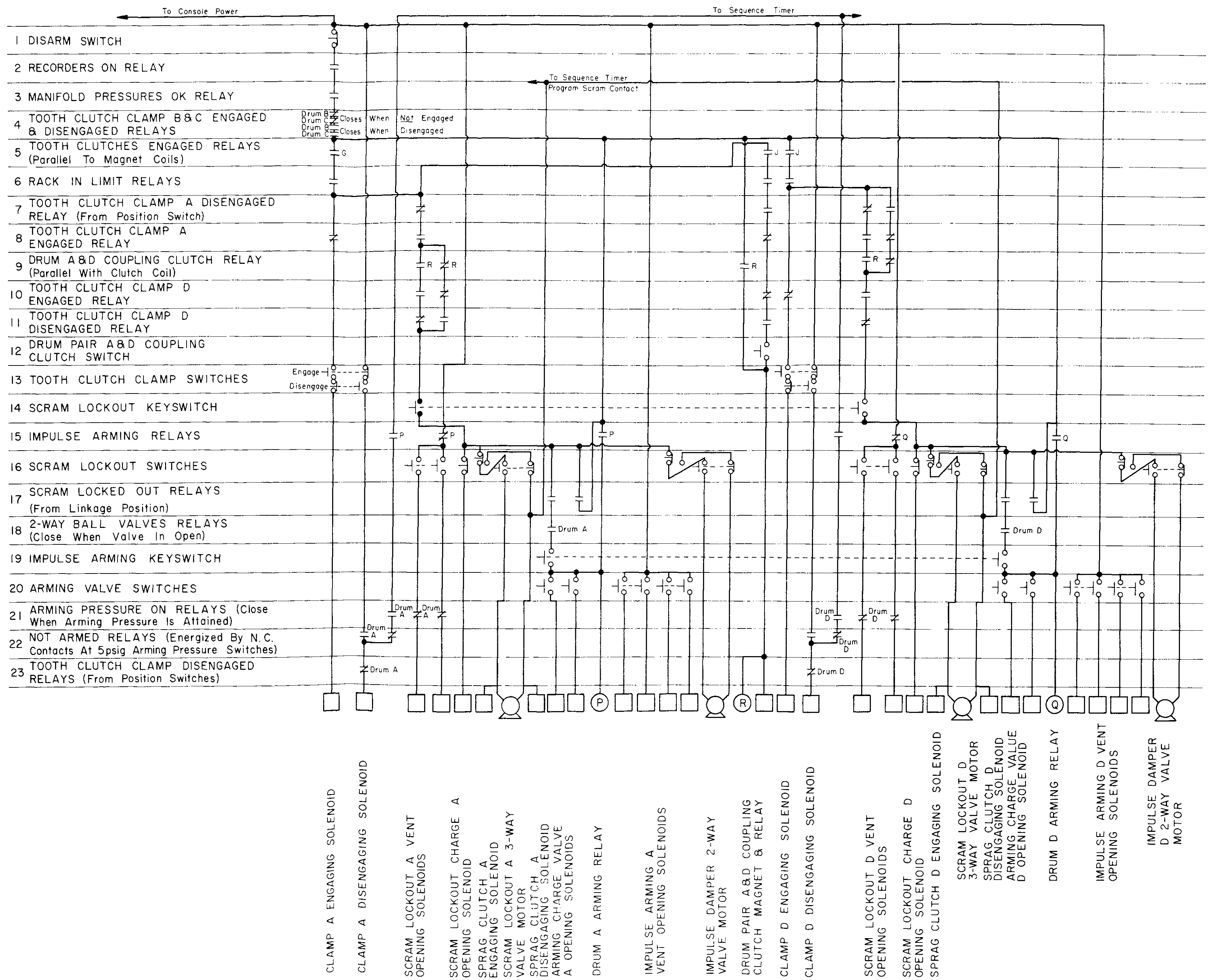


Fig. 12 - Impulse Arm Logic Control System Schematic

and relays must indicate that drum position and data recorders are operational and nitrogen pressures are satisfactory. Again, as in the impulse test, the choice of a one-drum or two-drum step is determined by the coupling clutch. Interlocks provide that neither drum can be armed if the tooth clutch clamp of the other drum is engaged unless the coupling clutch is also engaged. Further, the coupling clutch may not be engaged after any tooth clutch clamp is engaged. When the interlocks indicating that the tooth clutches are clamped and the drums coupled, actuation of the spring-return step arming keyswitch energizes the step arming relay and admits power to the step arming solenoid valves. The relay acts as a holding circuit shunting the keyswitch, thereby allowing the keyswitch to be released. The solenoid-operated normally open vent valves close when energized and the normally closed inlet valves open when energized, causing the pneumatic drive cylinders to be charged with compressed nitrogen. Pressure switches actuate when proper charging pressure is reached, lighting the "armed" and "ready" lights on the console and satisfying interlocks necessary for firing the transient by the sequence timer. Each solenoid valve is furnished with a normally closed test switch enabling its power to be interrupted to check its operation.

E. Operational Instrumentation

For effective surveillance of the nuclear status of the reactor, the operator will have control of and indication from: (1) four channels of neutron pulse counting equipment, covering in staggered overlapping ranges, source level to 100 w, (2) two channels of linear power recording equipment covering 1 w to 1 Mw, (3) one channel of log power recording equipment covering 1 w to 10 Mw. These systems are schematically represented in Fig. 14 where closed circuit television and aural communication surveillant equipment is also shown.

The detectors for the lower level startup and linear power channels will be placed on the dolly directly below and above the reactor where a variable shielding effect from rotating drum movement will be minimized. The threshold of sensitivity for these detectors will be 1 cps for 10^2 nv at the reactor for the start channel and 10^{-11} amps for 2×10^5 nv or 10^{-2} watts at the reactor for the power channel.

A 10 curie remotely retractable source placed directly under the reactor will be used for the loading experiments where the source radiation will be seen through the reactor by a startup channel. A one curie source will be used for reactor operation once the full core loading is achieved.

For transient tests the high level power channels will be placed under the dolly and at other distant locations to cover power levels in overlapping ranges up to 10^9 watts.

F. Experimental Instrumentation

The reactor will be instrumented to provide the maximum amount of information within the capabilities of existing instrumentation and within the limitations of the technology and space available for the installation of in-core detectors. The rigid requirements for pressure, temperature, and other detectors possessing adequate response characteristics as well as the techniques for attachment of these detectors to the SNAP reactor fuel pose a particularly difficult problem which must be recognized. In order to provide the maximum amount of information and still avoid compromising the integrity of the reactor fuel or other components during the destructive test, the instrumentation installed for the nondestructive test series in which nitrogen is substituted for the NaK coolant, a reactor vessel equipped with instrument lead access nozzles will be supplied. The measurements to be made during this test series are as follows:

(1) fast and thermal neutron flux
(2) gamma flux
(3) internal and external fuel rod temperatures
(4) longitudinal fuel rod strain
(5) internal fuel rod pressure
(6) upper and lower grid temperature
(7) internal vessel pressure
(8) external vessel temperature
(9) vessel strain; hoop and longitudinal
(10) permanent beryllium reflector temperature and strain

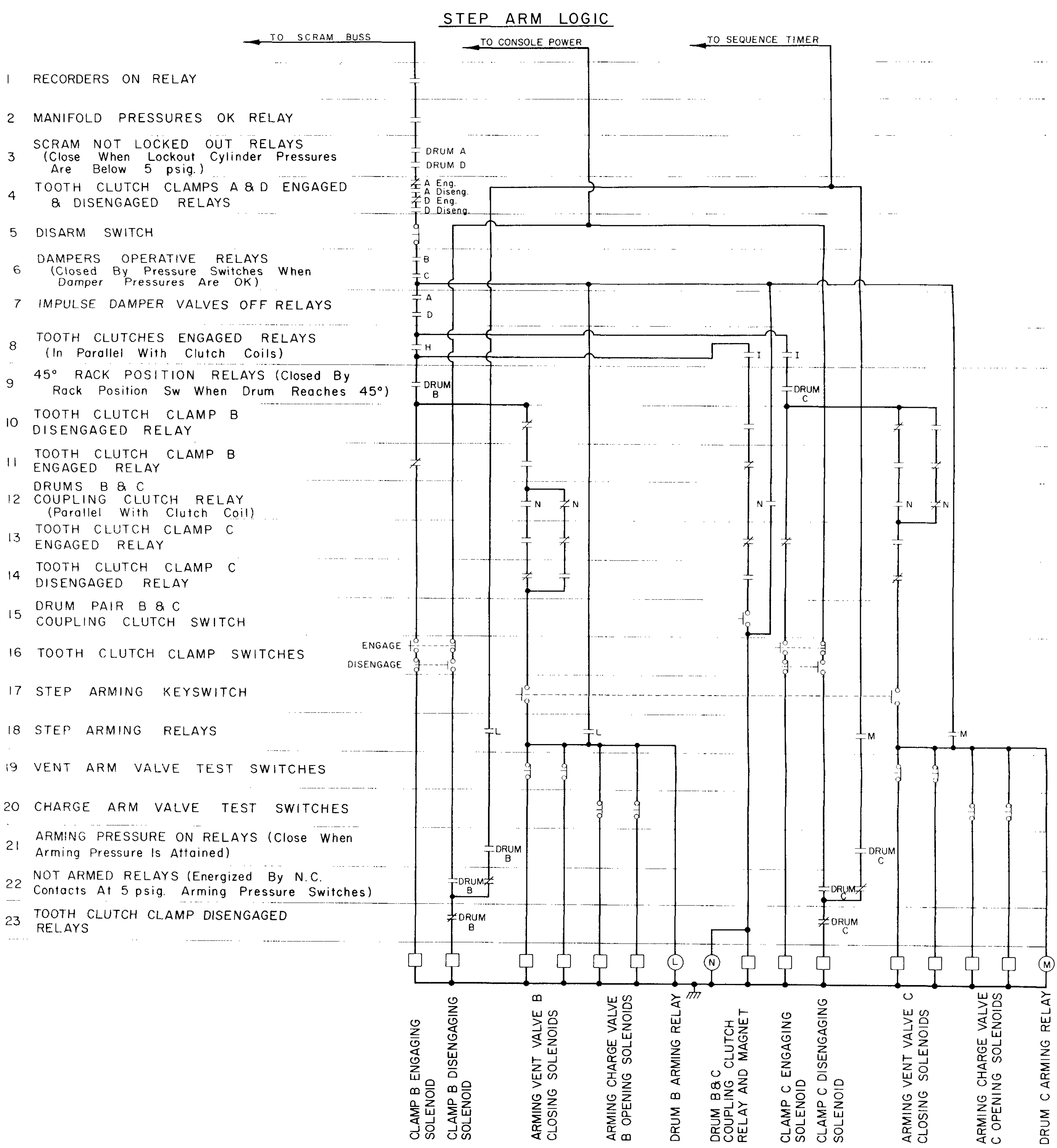


Fig. 13 - STEP Arm Logic Control System Schematic

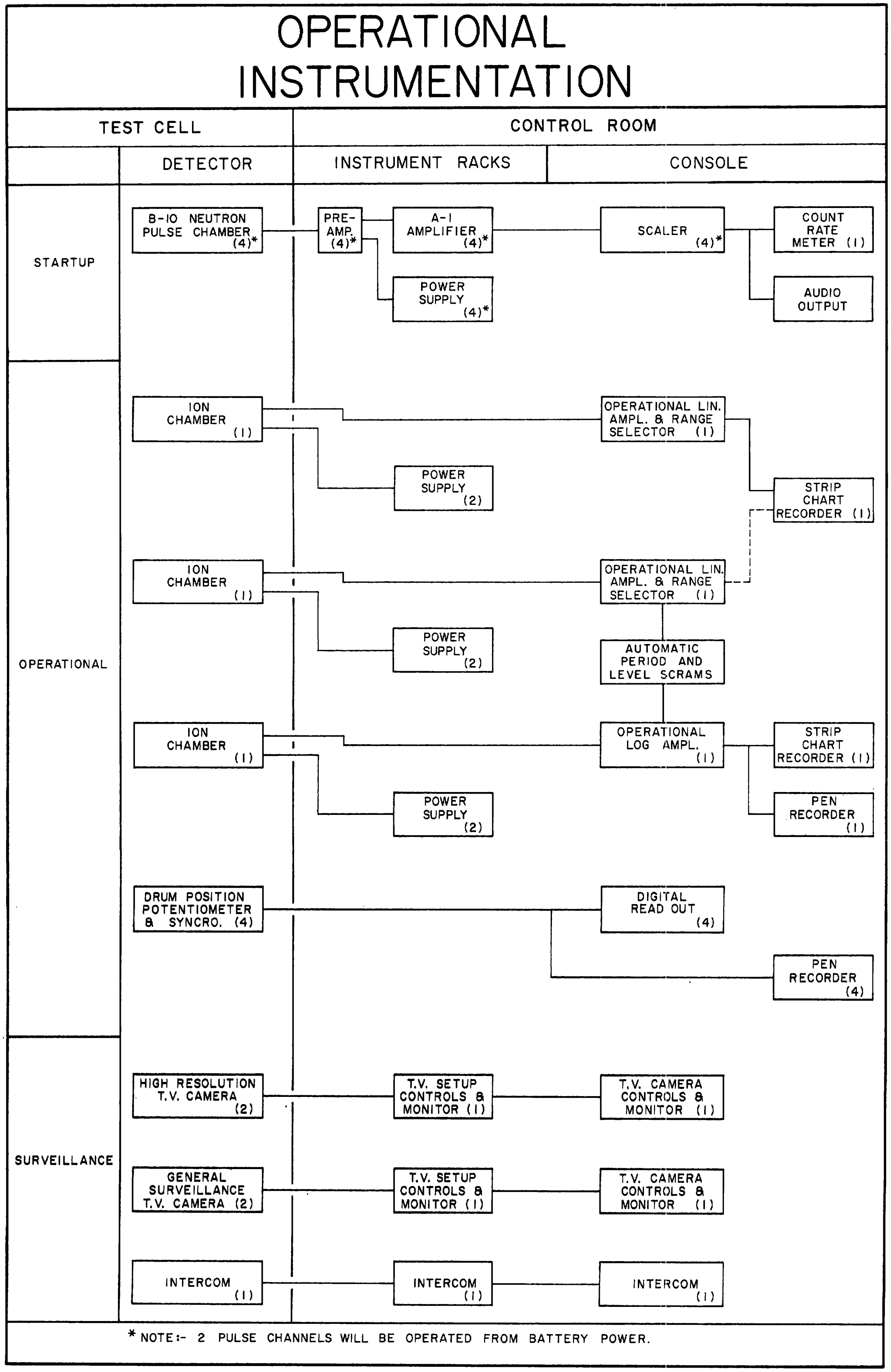


Fig. 14 - Operational Instrumentation

For the destructive tests the following parameters will be measured:

(1) external fuel rod temperature
(2) upper and lower grid temperatures
(3) internal vessel pressure
(4) external vessel temperature
(5) vessel strain; hoop, and longitudinal
(6) permanent beryllium reflector temperature and strain
(7) acceleration of beryllium reflector fragments
(8) acceleration of vessel fragments
(9) acceleration of mounting dolly
(10) fast and thermal neutron flux
(11) gamma flux
(12) blast pressure and temperature
(13) acoustic pressure
(14) fragment velocity
(15) infrared radiation

The recording instrumentation for the above measurements are schematically represented in Fig. 15.

One of the experimental objectives of the STEP program is to measure and evaluate the radiological hazards associated with the SNAPTRAN 2/10A-1 destructive test and if possible to correlate the theoretical and experimental results.

To completely evaluate the radiological hazards the proposed instrumentation distribution, as shown in Fig. 16, will be capable of measuring the following parameters:

(1) gamma ray dose rate from residual fission product decay as a function of distance from the reactor and time after the transient
(2) gamma ray dose from residual fission product decay as a function of distance from the reactor and time after the transient
(3) gamma ray dose from the transient plus that from residual fission product decay as a function of distance from the reactor

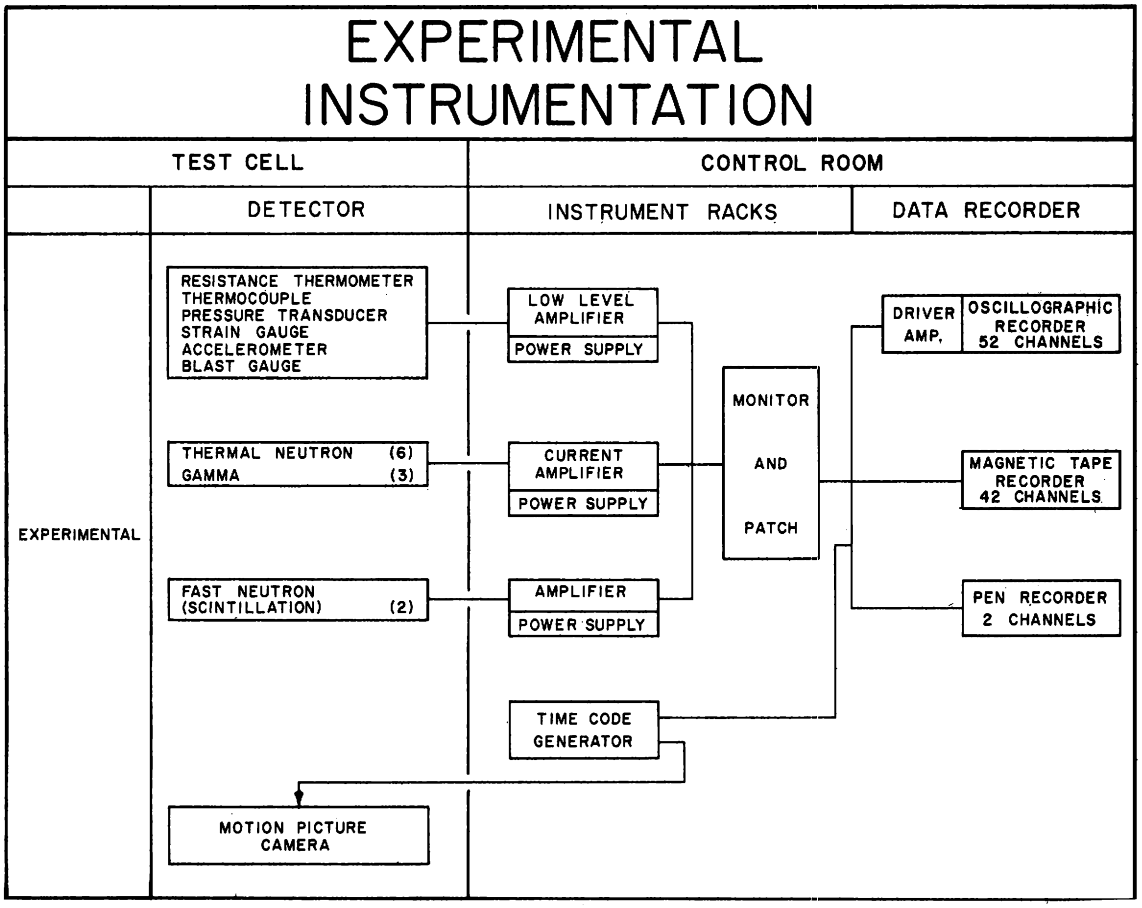


Fig. 15 - Experimental Instrumentation

(4) gamma ray dose from the transient as a function of distance from the reactor

(5) gamma ray dose from the transient plus that from residual fission product decay as a function of distance from the reactor and time after the beginning of the test

(6) total neutron dose rate from the transient as a function of distance from the reactor

(7) total neutron dose from the transient as a function of distance from the reactor

(8) gamma ray dose rate from the cloud as a function of position under the cloud at various times after the excursion

(9) total dose from the cloud as a function of position on the grid

(10) total dose from fallout as a function of position on the grid 24 hours after the excursion

(11) maximum gamma-ray dose-rate from fallout as a function of position on the grid and time after the excursion

(12) fallout distribution, concentration, and complete isotopic identification as a function of position on the grid

(13) particulate activity in the cloud and complete isotopic identification as a function of position on the grid

(14) noble fission gas concentrations in the cloud and their isotopic distribution

(15) cloud concentration and distribution of radioisotopes according to particle size

(16) measurement of beryllium release

As shown in Fig. 16 Phillips Petroleum personnel will be responsible for radiological measurements for distances up to ~ 2.3 miles downwind of IET. The AEC-ID will be responsible for measurements and evaluation of the radiological parameters for distances greater than 2.3 miles.

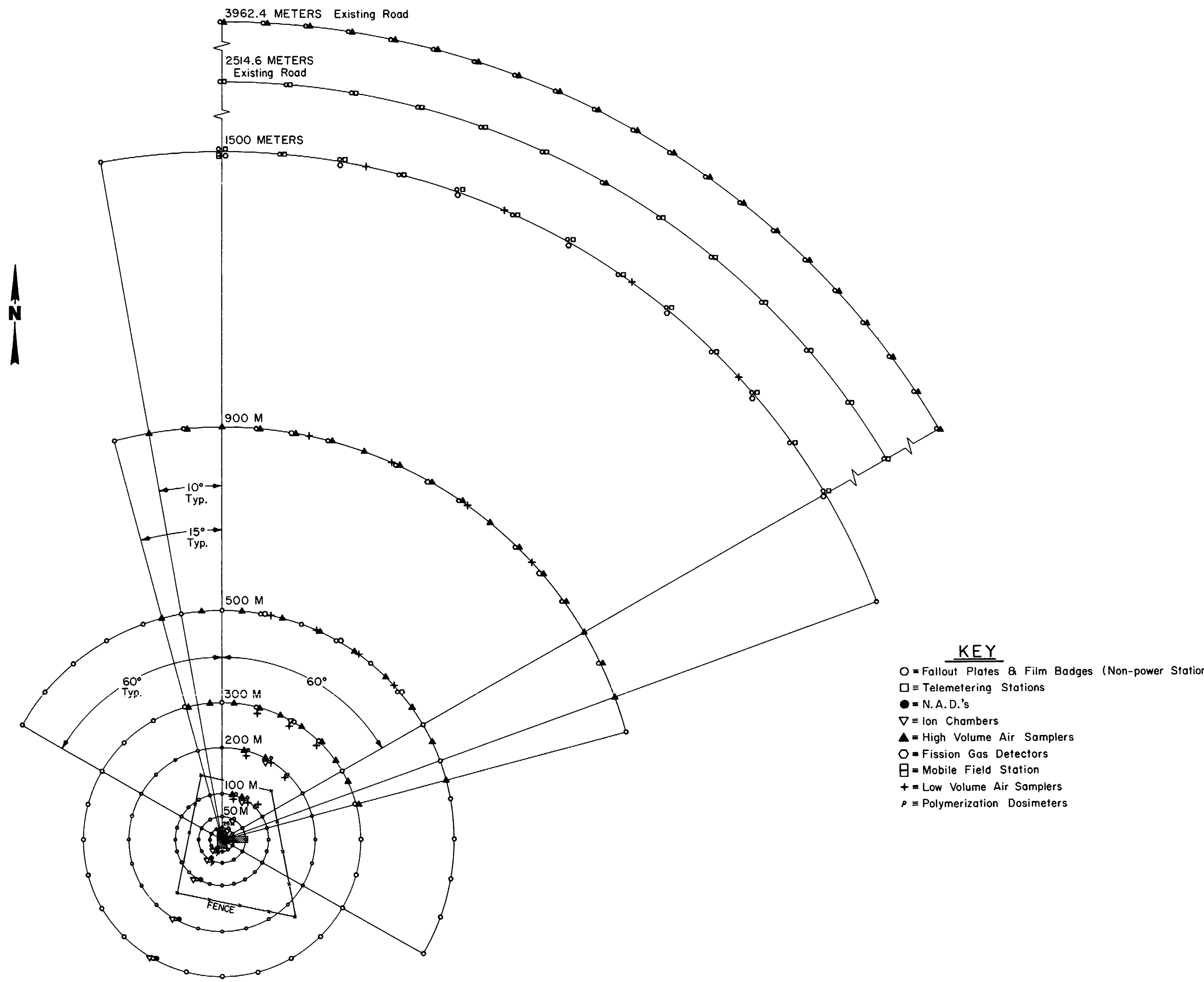


Fig. 16 - Monitoring Grid

IV. DESCRIPTION OF SITE AND FACILITIES

A. Site Description

1. General

The Test Area North (TAN) facilities are located within the National Reactor Testing Station (NRTS) boundary and are approximately 27 miles north-northeast of the Central Facilities area, as shown in Fig. 17. The TAN area, formerly known as the Aircraft Nuclear Propulsion (ANP) area, is located at a latitude of 43° 50' N, longitude 112° 41' W and at an elevation of 4790 ft[3]. The TAN area is composed of four different facilities (Table IV-1), one of which is the Initial Engineering Test (IET) facility, formerly identified as the Initial Engine Test facility. The SNAPTRAN 2/10A-1 reactor test program will be conducted in the IET facility.

TABLE IV-1

TAN FACILITIES

Location	Distance & Direction from IET		Personnel Distribution
Initial Engineering Test (IET)			20
Lithium Cooled Reactor Experiment (LCRE), formerly Flight Engine Test (FET)	1.5 miles	West-southwest	0
Low Power Test (LPT)	2.7 miles	South-southeast	20
Technical Support Facilities (TSF), formerly Administrative & Maintenance Area (AMA)	1.3 miles	Southwest	158

The projected daytime population of TAN is approximately 300 personnel. The 300 daytime and/or estimated 50 nighttime personnel will be distributed among the TSF, LPT, LCRE, and IET facilities.

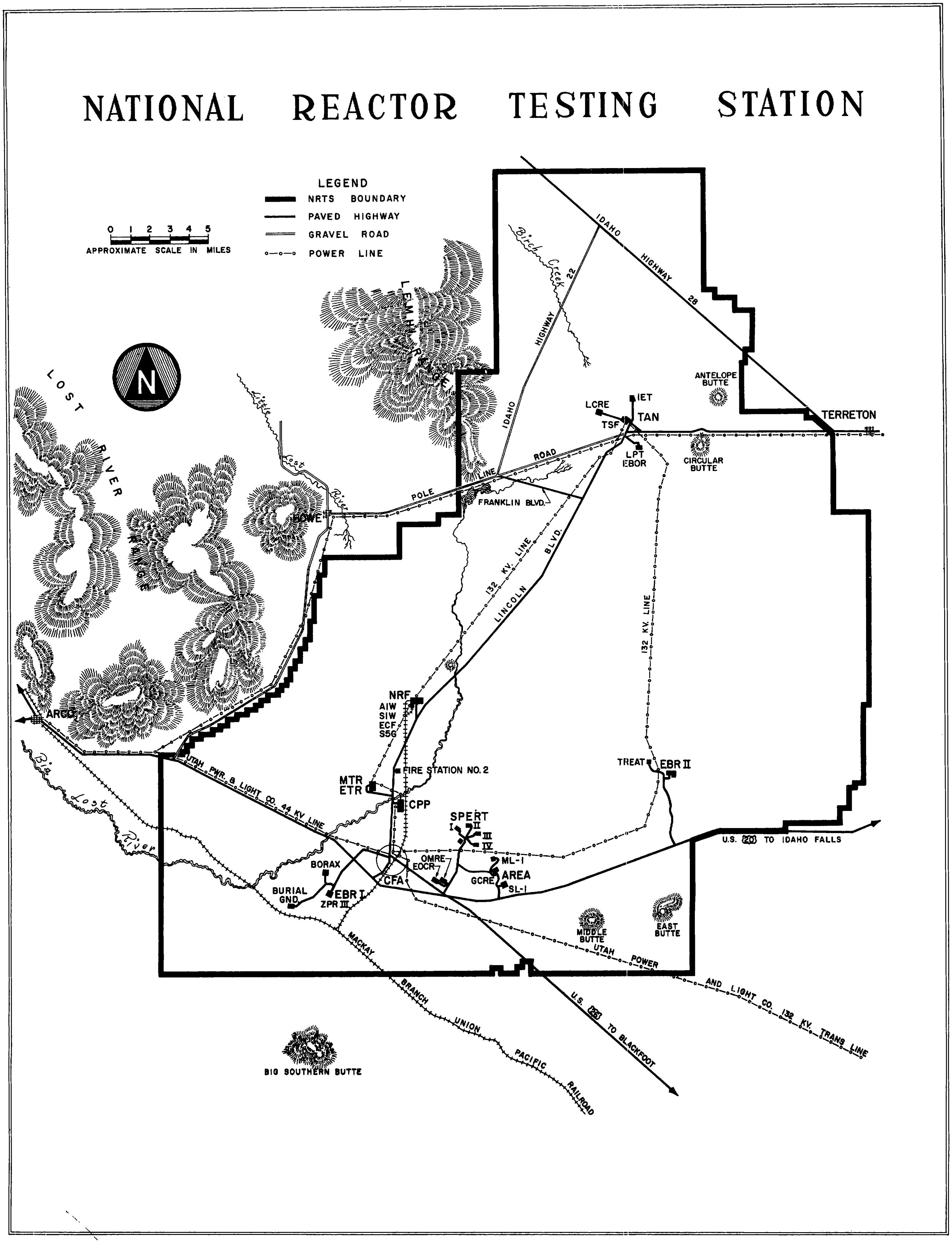


Fig. 17 - National Reactor Testing Station

The surface of the area is nearly flat with an elevation of 4800 ft at the IET. Surface drainage is good and studies have shown there are no important impermeable boundaries existing within the TAN area[4].

A meteorological tower located at IET is used to provide weather information for controlled disposal of radioactive materials to the atmosphere. The tower is 200 ft high and contains wind direction and temperature measuring equipment.

2. Geography

a. Topography

The NRTS is located in southeastern Idaho on the Snake River Plain at the foot of the Lemhi, Lost River, and Beaverhead Centennial mountain ranges (Fig. 18). From north to south it is 34 miles long, and from east to west it is 29 miles wide[5].

The site is at an average elevation of 5000 ft above sea level with a 1% slope from southwest to northeast. As seen in Fig. 19, which is a topographic cross section along the X',X' and Y',Y' axis of Fig. 18, the elevation is lower along the Snake River to the east except for a gentle 600 ft rise midway between the site and the river.

The mountains, seen in Fig. 18, encircle the Snake River Plain and rise to elevations of 10,000 and 11,000 ft above mean sea level. While the Centennial Mountains form an unbroken barrier at the northern end of the Snake River Plain, the ranges to the northwest and southeast, which form the side of the plain, are penetrated by deep valleys oriented northwest-southeast[5]. From the valleys between the mountain ranges to the northwest of the site, the courses of the Big Lost River, Little Lost River and Birch Creek run into playas or sinks, located in the northcentral section of the NRTS[6].

b. Population Distribution

(1) On Site. The working force at each of the NRTS facilities as shown in Table IV-2 is variable, depending on the construction work in progress. The closest facility to TAN is NRF which

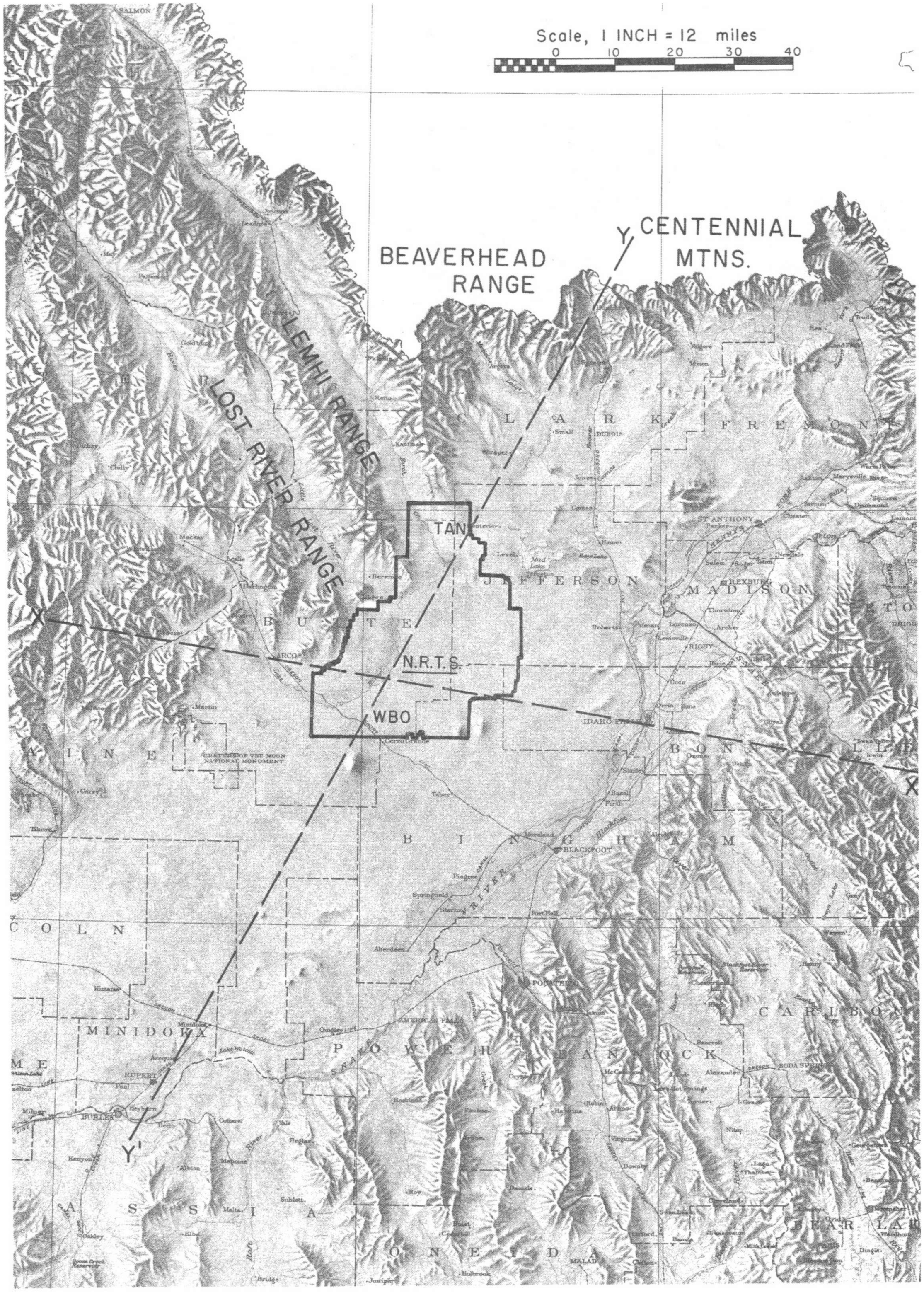


Fig. 18 - NRTS and Southeastern Idaho

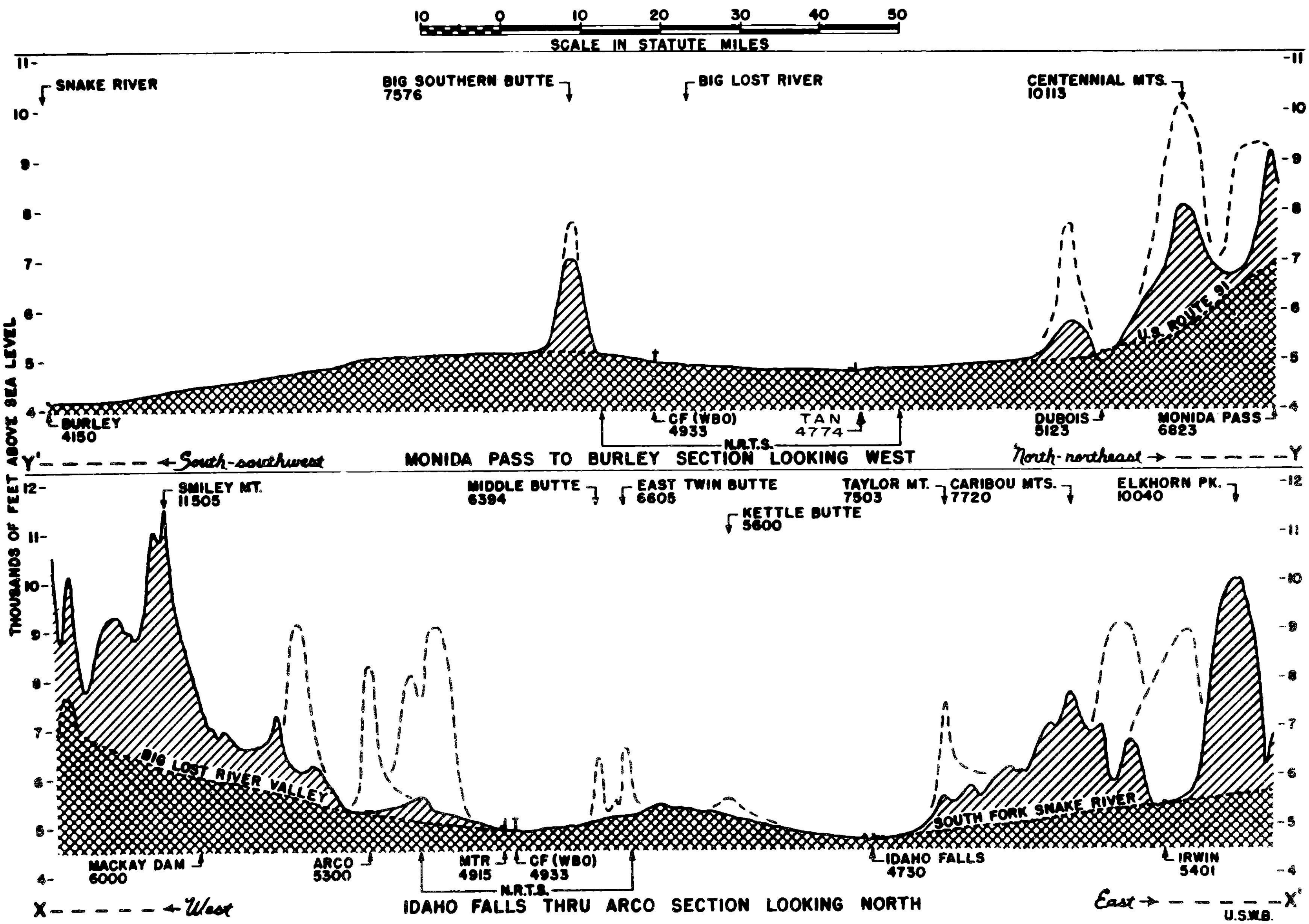


Fig. 19 - Topographic Cross Section Through NRTS

is approximately 22 miles southwest and has a daytime population of 1400 with a nighttime force of 200.

TABLE IV-2

POPULATION DISTRIBUTION OF NRTS[7]

Location	Day	Night	Distance from TAN in Miles	Direction from TAN
TAN Area	300	50		
NRF	1400	200	22	SW
MTR-ETR	700	150	25	SW
CPP	300	20	25	SW
EBR-I and BORAX 5	100	2	30	SW
Central Facilities	800	100	27	SSW
OMRE-EOCR	250	10	27	SSW
SPERT	140	5	26	SSW
Army Reactor Area	100	5	27	SSW
TREAT and EBR-II	200	5	22	S
Total	4290	547		

(2) Off Site. The towns with unlisted populations in Table IV-3 are unincorporated and no census figures are available. For location of towns see Fig. 20.

3. Meteorology

The meteorological conditions at NRTS are described in detail in previous reports[6,9]. These reports are used as a basis for the following data:

a. Temperature[10]

The maximum recorded temperature at the Weather Bureau Office (WBO) at NRTS was 101°F in July, 1960, and a minimum temperature

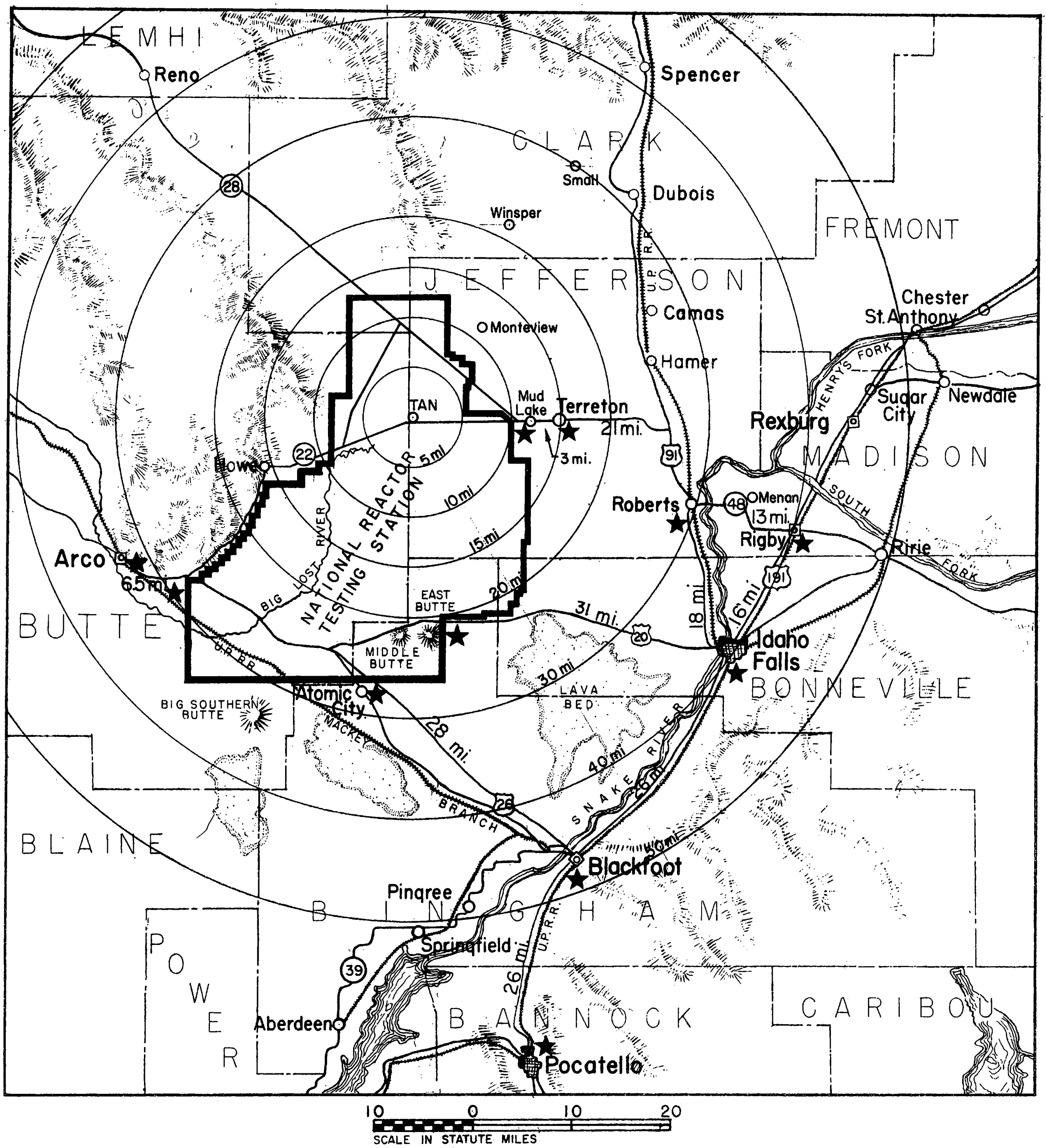


Fig. 20 - National Reactor Testing Station and Vicinity

TABLE IV-3

OFF SITE POPULATION DISTRIBUTION[8]

Location	Distance from TAN in Miles	Population	Direction from TAN
Mud Lake	10-15	187	East
Monteview	10-15	*	NE
Berenice	10-15		West
Terreton	10-15	*	East
Howe	15-20	25	WSW
Hamer	20-30	144	East
Winsper	20-30		NNE
Roberts	20-30	422	ESE
Small	20-30	10	NNE
Dubois	30-40	447	NE
Menan	30-40	496	ESE
Lewisville	30-40	385	ESE
Idaho Falls	30-40	33,161	SE
Arco	30-40	1,562	WSW
Moore	30-40	358	WSW
Darlington	30-40	10	West
Atomic City	30-40	141	South
Humphrey	40-50	25	NNE
Spencer	40-50	100	NE
St. Anthony	40-50	2,700	ENE
Parker	40-50	284	ENE
Plano	40-50		East
Salem	40-50		East
Sugar City	40-50	584	East
Rexburg	40-50	4,767	East
Lorenzo	40-50	100	ESE
Sunny Dell	40-50		ESE
Rigby	40-50	2,281	ESE

TABLE IV-3 - Continued

Location	Distance from TAN in Miles	Population	Direction from TAN
Ucon	40-50	532	ESE
Iona	40-50	702	SE
Ammon	40-50	1,882	SE
Shelley	40-50	612	SE
Basalt	40-50	275	SSE
Firth	40-50	322	SSE
Goshen	40-50		SSE
Blackfoot	40-50	7,378	SSE
Moreland	40-50	320	SSE
Riverside	40-50		SSE
Rockford	40-50		South
Martin	40-50		WSW
Grouse	40-50	58	West
Mackay	40-50	560	ESE

* The rural population in the area surrounding Mud Lake, Monteview and Terreton is approximately 1,000.

recorded was -40°F in January, 1962. For the TAN area, a maximum temperature of 103°F was recorded in July, 1960, while the minimum temperature recorded was -43°F in January, 1960.

b. Precipitation

Precipitation for the NRTS in general averages 7.69 in. and appears in the form of rain, snow, sleet, and hail. As to the rain-snow distribution, snow has been experienced in every month except July, and rain can be expected at any time during the year. The greatest 24-hour fall of rain was 1.73 in. in June, 1954, and that of snow, 8.5 in. in January, 1957. For the TAN area in particular, the annual amount of moisture is 7.35 in., with a maximum rainfall per 24 hours of 1.33 in. recorded in July, 1953.

c. Adiabatic, Lapse, and Inversion Conditions

Normal weather conditions at NRTS develop lapse conditions during daylight hours with inversion conditions readily forming around sunset and continuing until after sunrise. During the day, especially clear summer days, thermal convection and the accompanying turbulence mix the surface layers of air with those above so as to bring both to a nearly common speed, which is greater than the undisturbed or night surface speed. Should the surface winds maintain a speed greater than 15 miles per hour through the night, they will frequently prevent the formation of an inversion. The season with the least number of inversions is spring with inversions occurring on 92% of the days and lasting for one hour or longer. During the winter season, inversions are of a longer duration than those during the summer months due to the longer nights in the winter. From Fig. 30 of IDO-12015[9], inversions may be expected 92% to 98% of the nights of the year with a duration of one hour or longer, and an inversion of at least ten hours' duration may be expected on more than 61% of the nights of the year.

When the inversions break up, there is a period of fumigation. All measurements of the fumigation period indicate that it is of short duration.

Measurements of the thermal gradient at NRTS have been underway since 1950, and the data in IDO-12015[9] lead to the following conclusions:

(1) When long durations of inversions are considered (greater than 15 hours), Fig. 30 shows that during the summer months the duration of such an inversion has only a 0.2% probability of occurring. However, during the spring, fall and winter seasons the probability of an inversion of such duration increases considerably.

(2) Lapse conditions where $n = 0.25$ or lower exist over 50% of the time.

d. Winds

The NRTS is in the belt of prevailing westerly winds which are channeled upon entering the valley. A southwest wind predominates at the south end of the site while south-southwest winds occur most frequently at the north end. The strongest channeled winds at the north end of the site generally come from the northwest out of the Birch Creek Valley. On occasion, these winds have been observed for greater than 60 consecutive hours.

The winds at NRTS show a seasonal variation with the principal contrast being between winter and summer as shown in Figs. 21 and 22. Particularly noticeable in the winter is the absence of the southwest wind at the north end of the site in the vicinity of TAN. The prevailing wind during this time is from the northeast. A correlation between precipitation and wind direction indicated that a precipitation wind rose did not vary significantly from a surface wind rose.

For the purpose of establishing the upper limit of environmental hazard resulting from an activity release at IET, strong inversion conditions with Sutton's stability constant $n = 0.5$ are used. Lapse conditions, represented by the Sutton constant $n = 0.20$, are used as the most probable average condition existing at the site.

A discrepancy of about 17% was noted between the value for the diffusion coefficient, C, as listed in the HTRE hazards report(4), and the values listed in IDO-12015(9). The values listed in IDO-12015, and as shown in Tables 30 and 31, are used in Sutton's diffusion equations since these values were found by the site Weather Bureau Office to be more accurate than those listed in the HTRE report.

4. Geology

The NRTS is at the central northern edge of the semi-arid Snake River Plain in southern Idaho, adjacent to the southern foothills of the Lemhi and Lost River ranges. The plain extends in a great arc about 350 miles across Idaho, from the Oregon boundary west of Boise to near Ashton in eastern Idaho. The surface of much of the plain is

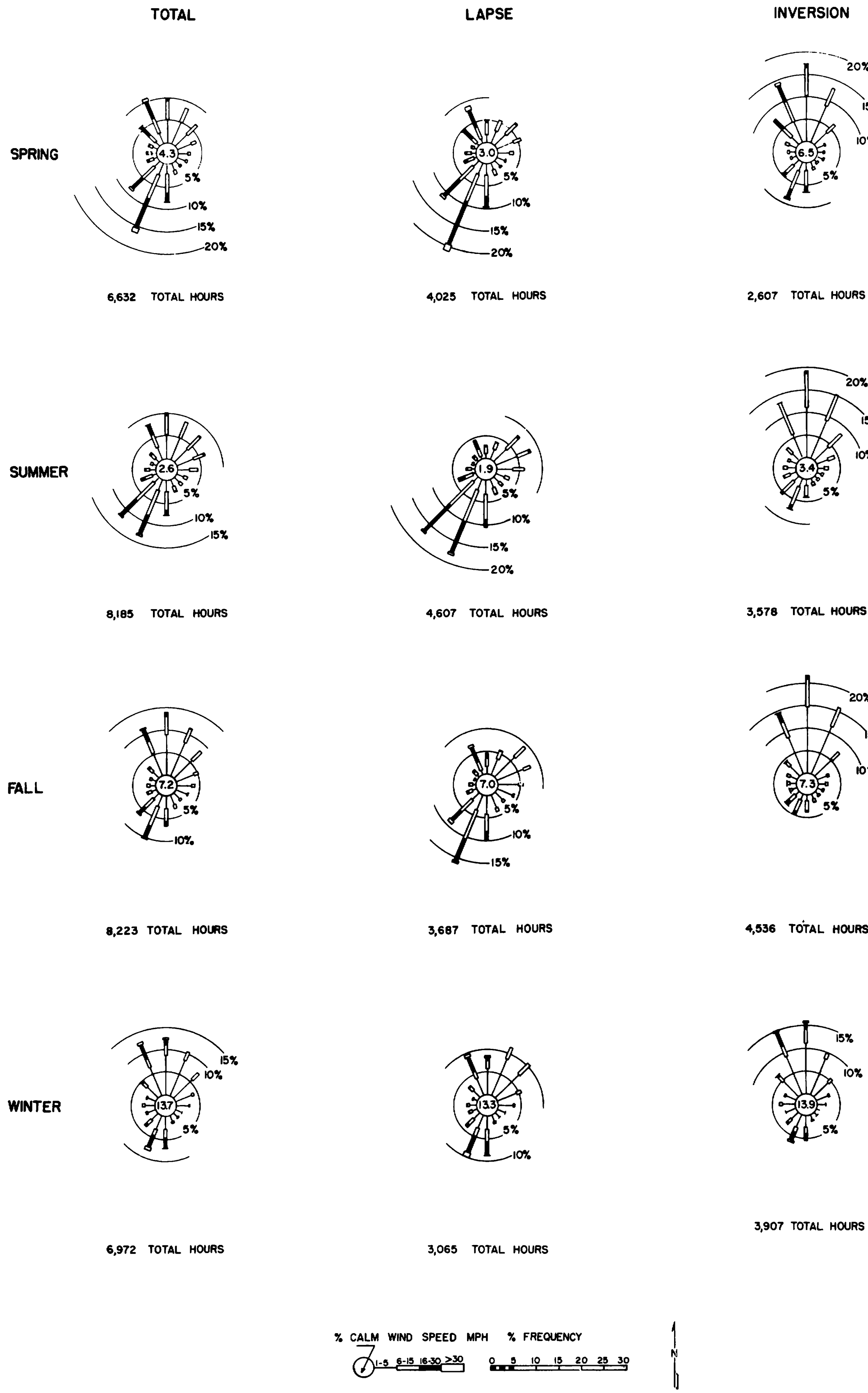


Fig. 21 - TAN 20 Foot Level Wind Roses 1953 Through 1956

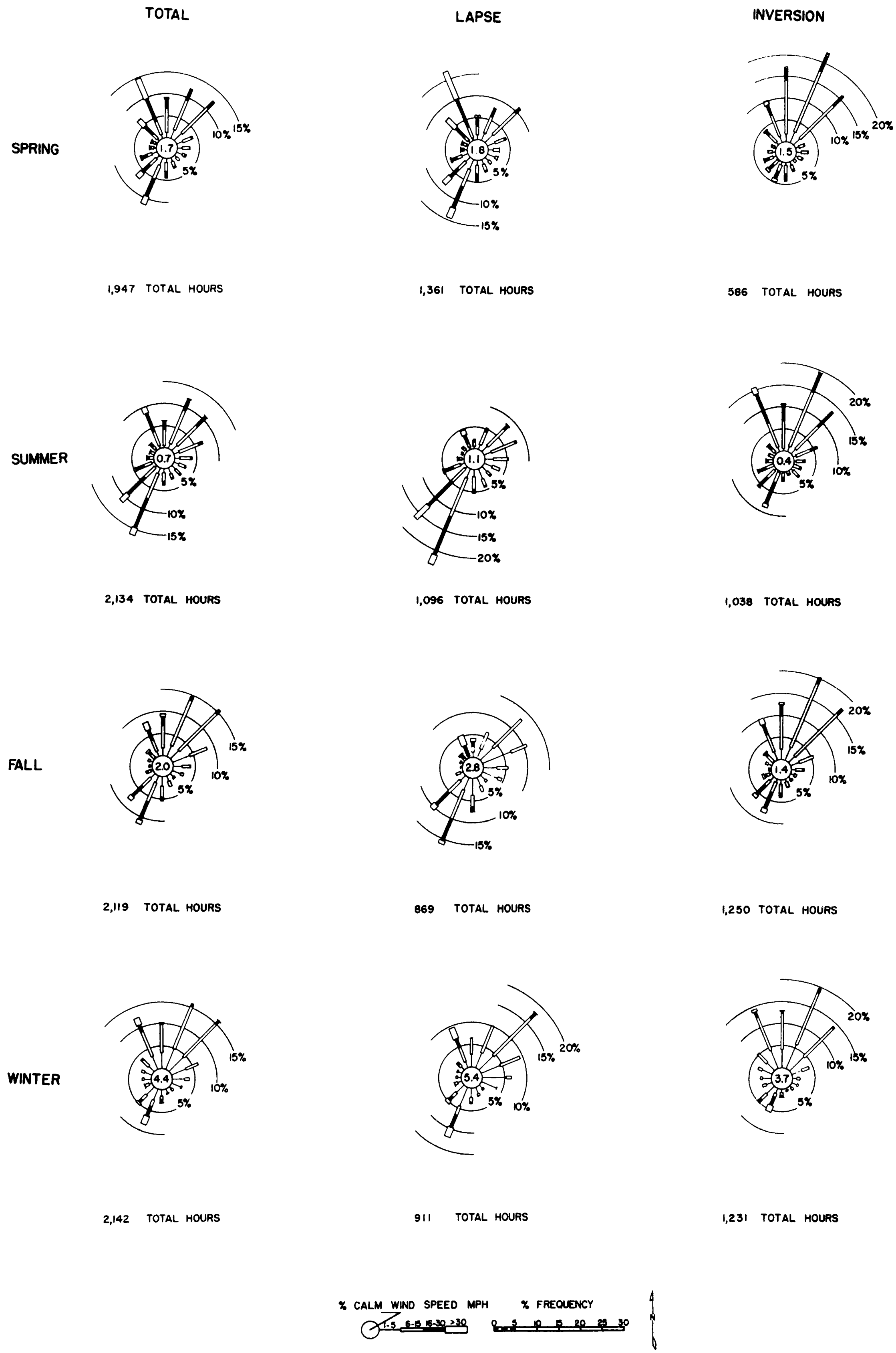


Fig. 22 - TAN 150 Foot Level Wind Roses May 1956 Through April 1957

covered by waterborne and windborne topsoil, under which there is a considerable depth of gravel ranging in size from fine sand to 3 in. aggregate(5).

The NRTS has no well-defined, integrated, surface-water drainage system, and it is not crossed by perennial streams. However, the NRTS overlays a natural underground reservoir of water, having an estimated lateral flow of not less than 500 cu ft/sec, or about 323 million gal/day(11).

The main sources of water for this reservoir are the streams that originate in the mountains to the north and disappear into the porous soils of the NRTS area. These sources of water include the Big Lost River, Little Lost River, Birch Creek, and also an additional source from Mud Lake Basin(11).

The altitude of the water table ranges from 4580 ft above sea level in the northern part of the station to about 4400 ft near the southwest corner. The water table in the TAN area is at a depth of 200 ft and is seemingly very flat and at places may slope less than one foot per mile. Immediately beneath the central TAN area, the general direction of underflow appears to be south and southwest(4).

5. Seismology

The NRTS site is located in a region which "The Pacific Coast Uniform Building Code", (1949), designated as a zone-two area(12). Although many recorded earthquakes have been felt in Idaho, none were of sufficient intensity to cause more than minor damage to buildings. Of the 14 recorded earthquakes with an epicenter within the state, seven had their epicenters within 100 miles of the IET site, four to the southeast and three to the west. One of these of unrecorded intensity was at Arco, Idaho. In spite of the fact that a zone-three area exists both north and south of the Arco area, the distances are so great that a zone-two designation has been considered completely safe(5).

Although the lava plain of the Snake River is geologically young, the surrounding mountains are mostly of great age(5). Some recent geological faults appear to cross the plain beneath the lava beds, although their traces are not evident on the surface. None of these show indication of recent historical movement outside the lava plain. It may be expected that earthquake shocks will continue to be felt in the site area, but a prediction as to their intensity cannot be made with assurance.

B. Facilities Description

The facilities used in the SNAPTRAN 2/10A-1 tests are all located in the TAN complex. The Initial Engineering Test (IET) facility will be used for all nuclear tests and for initial assembly of the package. The Examination Area (Building 607), which is located in the Technical Support Facilities (TSF) area, contains complete facilities for handling radioactive power plants. This facility will be used for package disassembly and reassembly after the fission product inventory is sufficient to prohibit direct contact. All metallurgical examinations will also be done in this facility. Movement of the reactor package between these facilities is provided by mounting the package on a special four-rail dolly. A shielded locomotive provides power for moving the dolly along the rail system connecting the facilities. The facilities are described in the following section.

1. IET Building

a. General

The Initial Engineering Test (IET) facility is located ~ 1.3 miles north of the TSF within a barbed-wire exclusion area. The exclusion area, or test area (Fig. 23), restricts access to within approximately one mile of the facility. The immediate IET area (Fig. 24) is surrounded by a security fence with guard house and consists of a Test Cell, Control and Equipment Building, and several other auxiliary facilities. Included are fuel pumping, storage tank, exhaust stack sampling, and chlorination buildings. The storage tank building is used as a liquid nitrogen unloading and vaporizing station. Access to the area is provided by means of a four-rail track system and an automobile roadway from TSF.

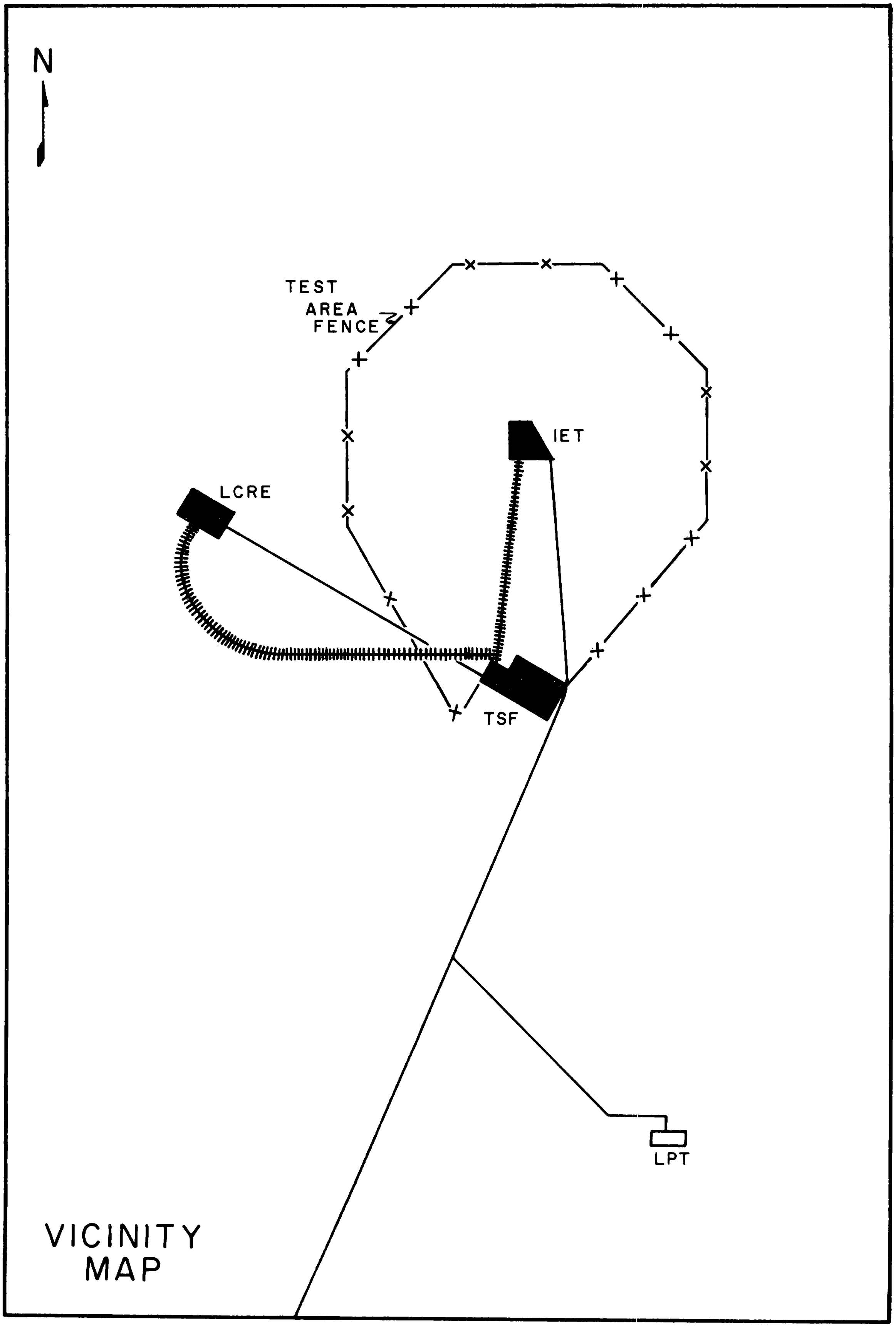


Fig. 23 - Vicinity Map of TAN Area

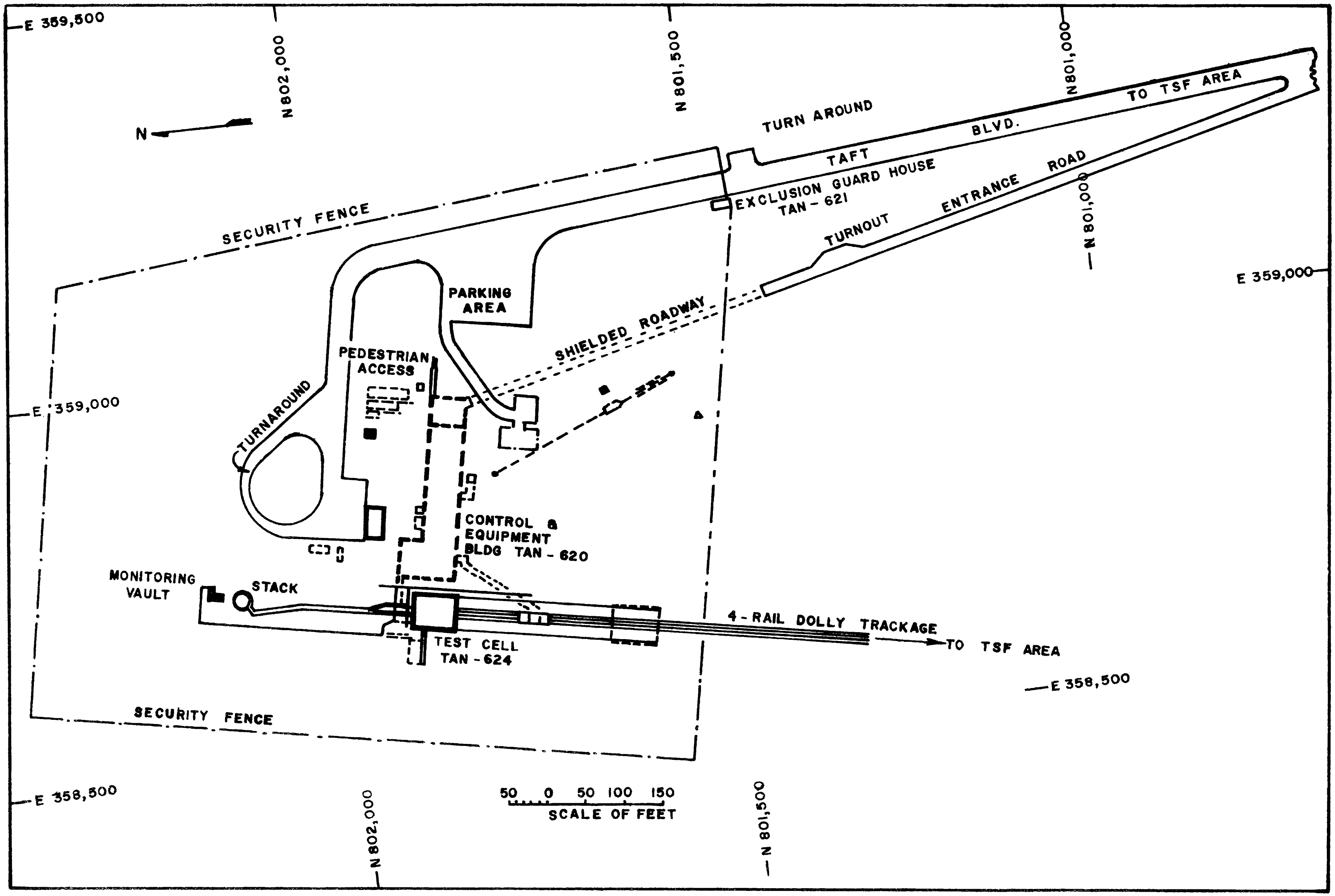


Fig. 24 - Plot Plan of IET Area

b. Control and Equipment Building

A plan view of the Control and Equipment Building is given in Fig. 25. The building is a poured concrete structure approximately 50 ft wide x 200 ft long, having 2 ft thick walls and a 3 ft thick ceiling. The ceiling is 2 ft below grade and is covered by mounded earth approximately 14 ft deep. The Control and Equipment Building and the Test Cell are separated by 15 ft of earth at the closest distance.

The west end of the building, which is nearest to the Test Cell, houses the control room, instrument shop, data analysis offices, mechanical equipment room, three administrative offices, and a health physics room. The control room is 24 ft x 38 ft and houses the reactor console, the process instrumentation, the test data readout instrumentation, and the remote viewing equipment. A detailed plan view of this room showing the location of various equipment is given in Fig. 26.

The large room, directly east of the control room, measures 28 ft by 44 ft and houses the instrument repair shop and data analysis offices. The health physics room and the three offices are adjacent to the instrument repair shop north of the control room. To the north of these lies the mechanical equipment room, 14 ft x 31 ft, which contains heating and ventilating equipment for the connecting tunnel to the Test Cell, for the coupling station, and for the change room. In addition, one section is utilized for a storeroom. The tunnel leading from this area to the Test Cell is arranged in a U configuration with boron treated floor and boron-gunite treated walls to provide personnel shielding.

The east end of the Control and Equipment Building contains the equipment room and the weather and radiation instrument area, boiler room, air conditioning equipment, power distribution room, diesel room, photo laboratory, and rest rooms. The largest room is the 21 ft x 106 ft equipment room which houses air compressors, freon refrigerating units, air washers, and water treating equipment. The 21 ft x 20 ft weather and radiation instrument area also contains valves and manifolding for the fire protection distribution system. The weather instrumentation and remote air sampling recorders are detailed in Table IV-4. The boiler room houses two 4.2×10^6 Btu/hr fire-tube

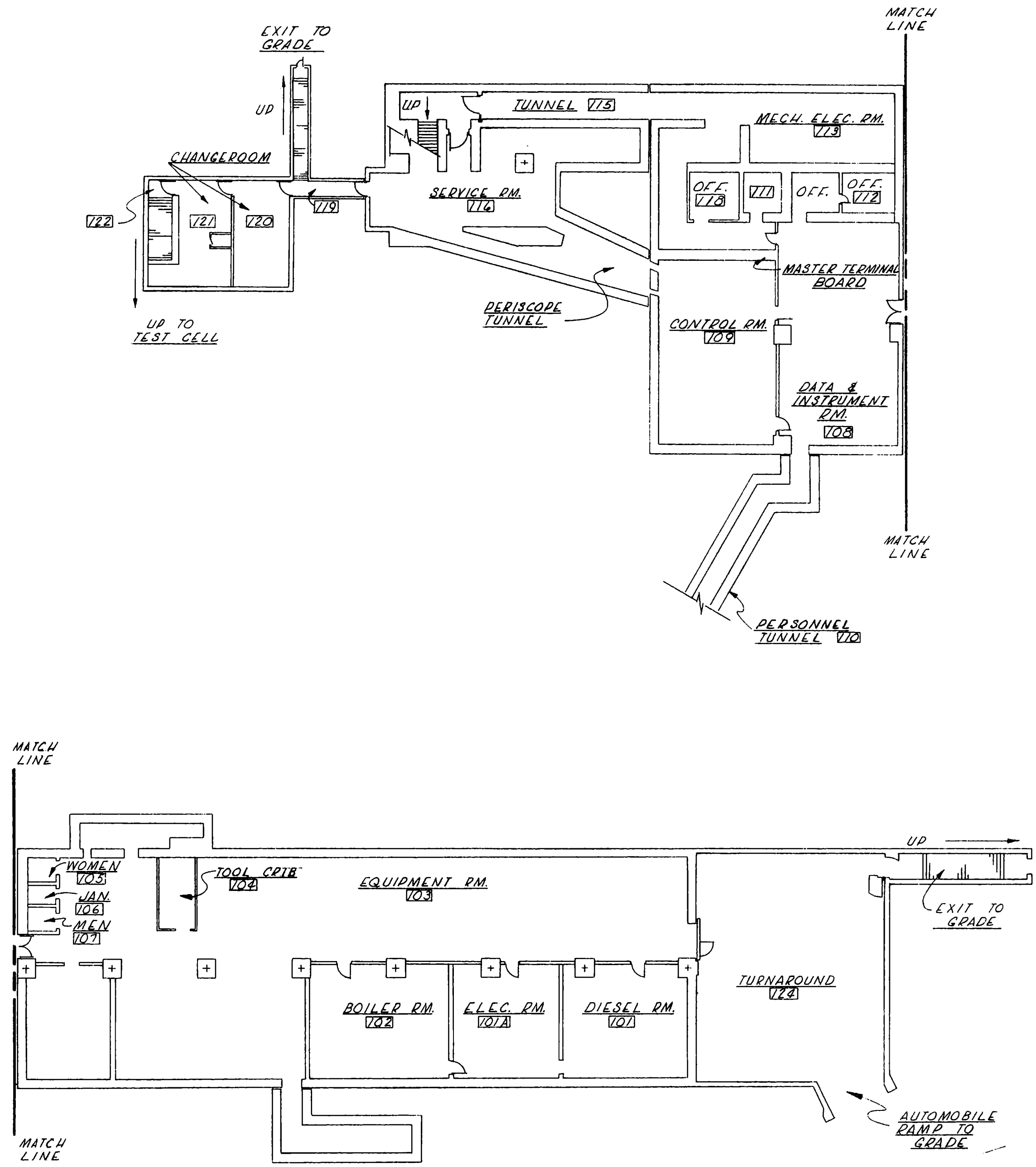


Fig. 25 - IET Control and Equipment Building

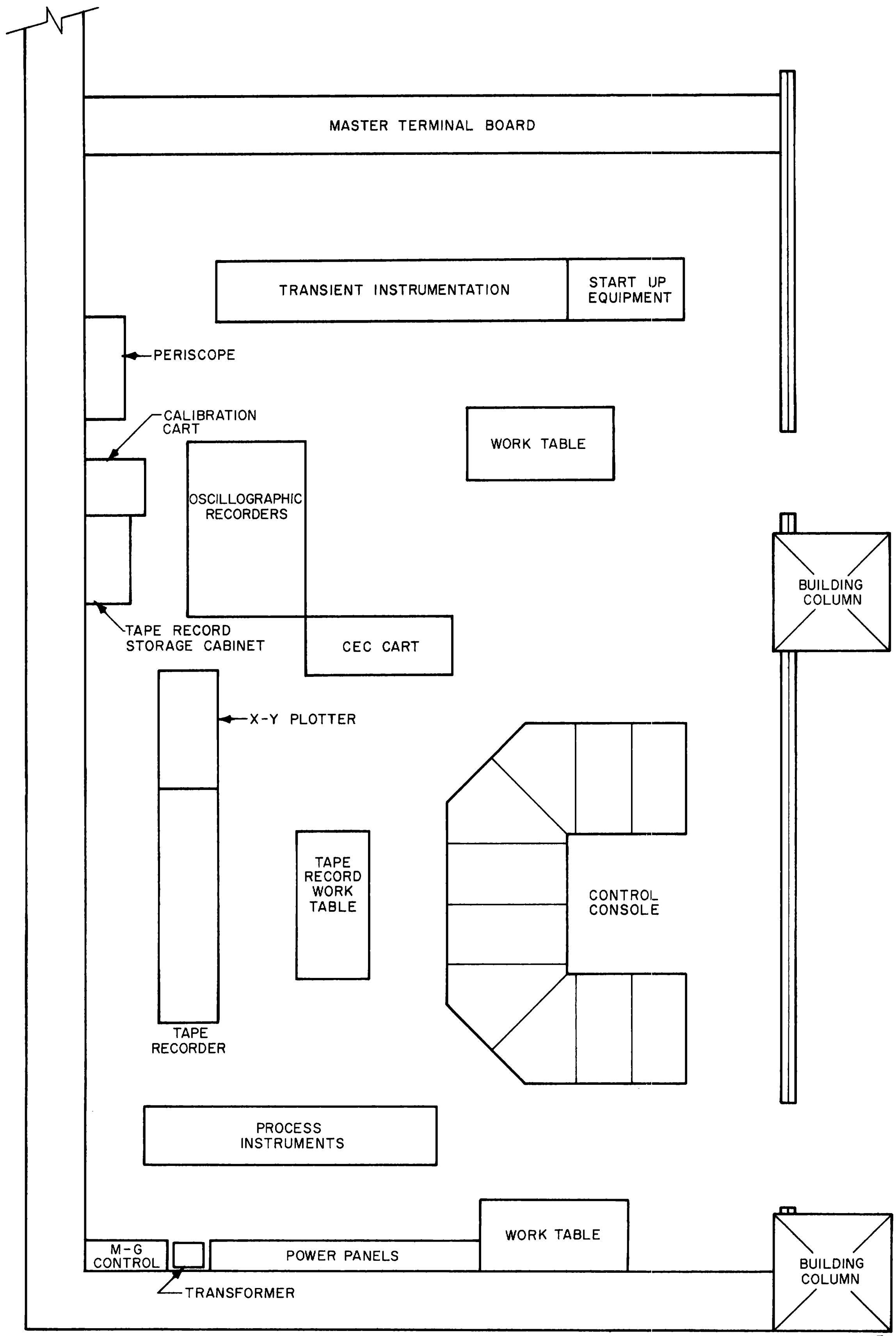


Fig. 26 - IET Control Room Layout

TABLE IV-4

WEATHER AND REMOTE AIR SAMPLING INSTRUMENTATION

Parameter Measured	Location of Measuring Device
Wind speed and direction	IET weather tower at 20, 75, 180, and 200 foot levels
Temperature	IET weather tower at 5, 20, 50, 60, 140, 150, 160, and 200 foot levels
Barometric pressures	Control room
Dew point and atmospheric air temperature	Weather tower
Stack air flow	IET exhaust stack
Stack activity - beta and gamma	Building 713
Remote area monitoring	Test Cell high and low, personnel entrance, covered roadway entrance
Activity monitor	Control room, data room, equipment room, service room, coupling station, access tunnel

boilers and all auxiliary equipment necessary to supply low pressure steam for area heating. Supply air filters and ducting occupy the areas between the instrument repair room and the boiler room. The 23 ft x 24 ft electrical switchgear room contains the protective equipment and breakers for commercial and emergency power distribution. The 29 ft x 24 ft diesel room houses a 600 kw emergency power diesel generator.

A 4 ft wide tunnel leads from the data analysis office area to a hatchway which opens between the center tracks of the four-rail system. By this route, shielded access to and from the building is provided to the shielded locomotive. Automobile access is also possible using the shielded auto ramp at the east end of the building with underground parking provided at the Control and Equipment Building level. An 8 ft wide door between the building and the parking area allows small truck access for repair and replacement of equipment.

c. Test Cell Design

The Test Cell was originally designed for weather protection only, but has been modified to allow for limited containment. A plan view and sections are shown in Fig. 27. The cell is constructed of 16 gauge aluminum siding and roof decking on an aluminum framework. The inner surface of the walls and roof girders have been covered with a 10-gauge carbon steel shell for ease in decontamination of the facility. A rubber skirt between the building and the ground decreases the air exchange between the building and the surrounding atmosphere.

The building is supported on double flange track wheels and can be moved by locomotive a distance of 260 ft from the test pad. When in place over the pad, about 43 ft x 60 ft of floor space is provided with a clear height of about 35 feet. The test pad is constructed of boron concrete to provide neutron shielding for the Control and Equipment Building.

At the north end of the Test Cell is located the coupling station which accepts the facility plug mounted on the reactor dolly. All instrument and utility connections to the reactor package from the control room, etc., are made in this station. The floor level of the coupling station access room is located vertically about half-way between the Control and Equipment Building level and that of the Test Cell floor. The room is 10 ft long x 22 ft wide with an 8 ft ceiling. The floor is boron treated and the walls are boron-gunite coated. The ceiling is constructed of high-density concrete and the stairs leading to the access tunnel are boron-concrete. The access opening to the reactor dolly is 19 ft long x 1 ft high and can be closed by means of lead shielding doors. As the coupler on the rail dolly is only about 8 ft wide, the remaining opening will be shielded by the doors during the test.

The 1,000,000 Btu/hr oil-fired building heater and the 12,000 cfm exhaust blower are located immediately above the coupling station. The exhaust blower is coupled to the stack system. This equipment is housed in a 10-gauge carbon steel enclosure for protection during the test program.

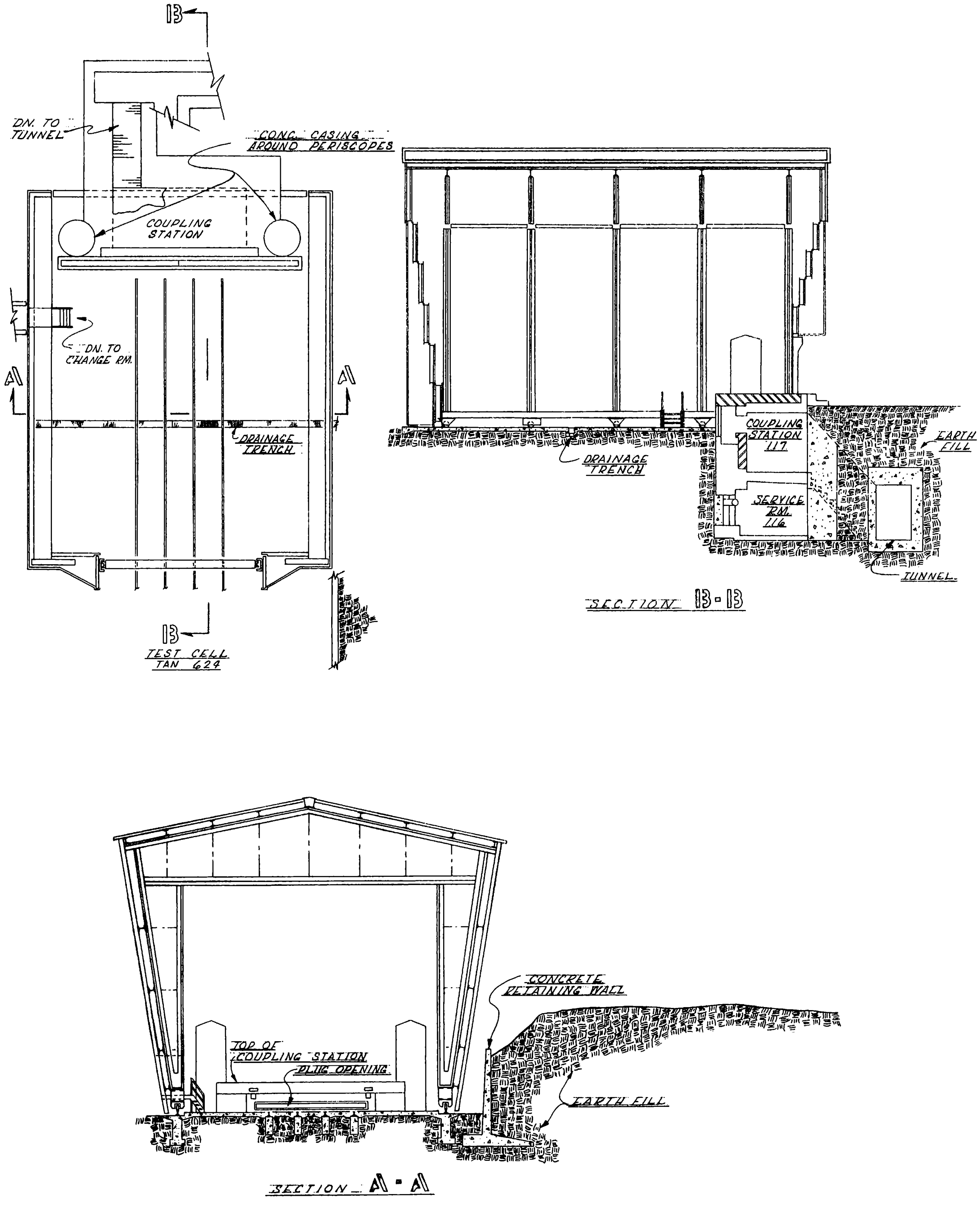


Fig. 27 - IET Test Cell

The periscopes for remote viewing from the control room are located on either side of the coupling station and are housed in a 10 in. pipe surrounded by 7 in. thick lead-shielding rings which are in turn covered by boron-concrete, giving a total diameter to the periscope column of 5-1/2 feet.

Normal access from the Control and Equipment Building to the Test Cell is by means of a connecting tunnel which passes under the coupling station through a decontamination room to the Test Cell stairway.

The area under the coupling station serves as a gallery for routing all instrument, electrical and utility leads and piping into the coupling station. The Test Cell air sampling collection system is also located in this area.

d. Utilities

(1) Power. The IET obtains 3-phase power at 13.8 kv from the Technical Support Facilities (TSF) area where it is transformed from the 132 kv NRTS loop system via a 7500 kva transformer. A 1000 kva transformer at IET reduces the voltage from 13.8 kv to 480 volts. A switchgear room located in the IET Control and Equipment Building houses all circuit breakers, fuses, and overcurrent and undervoltage relays. Single phase 120 volt power is also transformed and distributed from the switchgear room.

(2) Water. The water system at the TSF area consists of two wells, a storage tank, and booster pumping station. Each well pump, rated at 1100 gpm, operates from level controls installed on the 500,000 gallon capacity storage tank. Pressure initiated fire water booster pumps start in the event of a large fire water requirement.

Raw water, chlorinated at the well, is supplied to the IET area through a 12 in. line from the wells and storage in the TSF area. Flow rates of 1600 gpm can be obtained for 4-hour periods when fire water or abnormal demands exist. Water pressure at the IET varies from 85 to 125 psi, depending on demand and location.

Two zero hardness 57 gpm zeolite water softeners supply soft water for the demineralizers, boilers, air washers, and the diesel generator cooling system at the IET. The system consists of two softening tanks and a brine saturation tank for reconditioning the softener. Two mixed-bed deionizers produce 18 gpm each of one megohm purity demineralized water. Normally, zeolite softened water is used to feed the demineralizers after filtering through a diatomaceous earth filter; however, raw water can be used directly with a resultant reduction in demineralizer capacity.

Potable water is obtained directly from the main raw water distribution system. A two inch header distributes the potable water to fountains, sinks, lavatories, etc.

(3) Fire Protection System. Fire water is supplied from the 12 in. raw water supply line and distributed by an 8 in. loop header to four standard fire hydrants located within the IET fenced area and to fire water systems within the IET buildings. The system is designed to supply 1000 gpm of water for a 4-hour period.

The IET Control and Equipment Building is provided with hose stations, deluge spray systems, portable CO_2 fire extinguishers, and dry chemical fire extinguishers. Two hose stations are located in the hallway in the utility room, one in the parking area, and one in the decontamination area. Seven 15 lb CO_2, three 10 lb CO_2, and one dry chemical portable fire extinguishers are situated in convenient locations throughout the building.

Deluge spray systems are located in the diesel room, boiler room, coupling station service room, and the coupling station access room. The diesel room and coupling station service rooms contain rate-of-rise temperature detecting alarm systems with manual activation required on the spray systems. In addition, the coupling station service room also has a fixed temperature detector alarm. The boiler room wet-pipe deluge system actuates automatically at 180°F when the fused heads melt.

The Test Cell contains one fire water hose reel located near the coupling station for purposes of decontamination following the destructive test. Normally this hose station will be isolated from the fire water system during test operations to eliminate potential water near the test package. A liquid-metal fire fighting unit with 300 lbs of Met-L-X will be located on the test railroad dolly along with portable Met-L-X bottles and CO_2 fire extinguishers located in the stairway to the test cell.

The Tank Building, serving as the nitrogen unloading and vaporizing station, has two CO_2 extinghishers. The Fuel Transfer Building has automatic alarms and a deluge system which starts automatically or manually.

(4) Heating and Ventilating System. The IET Control and Equipment Building contains an extensive heating and ventilating system with zone control. Zones 1 and 2 supply the control and data processing rooms and the utility area from a supply duct on the south side of the building; zone 3 supplies the coupling station, service room and decontamination rooms from a supply duct on the north side of the building. Steam requirements to the heating coils are furnished by one of the two 125 hp fire-tube boilers. All three zones have 100% makeup air which is filtered through a 2-inch coarse glass filter and two 1/2-inch layers of fine spun glass which provide a filtering efficiency of 99.4% based on particle count of atmospheric dust. Actual tests on particles of 0.9 microns DOP are to be conducted which will indicate the effectiveness of the filters for air contamination control.

Although the destructive test will be done under strict meteorological conditions, zones 1 and 2 will be shut off and the supply and exhaust dampers closed to prevent the possibility of contaminated air entering the control and utility area of IET. The zone 3 supply fan will remain on to maintain a positive pressure in the coupling station area and the air washer will be in operation to remove part of the iodine that could possibly be present. The zone 3 exhaust fan will be off to maintain the highest possible positive pressure; this will force the exhaust air to escape to the Test Cell from around the shielding door in the coupling station. Constant air monitors will be used to sample the supply air of all three zones and give warning if any air contamination occurs.

Temperature, pressure, and relative humidity are controlled in the control room, offices, and the large room housing the instrument repair shop and the data analysis offices. Individual exhaust dampers automatically control the air flow to each room.

Heated air in the winter and evaporatively cooled air in the summer are supplied to the diesel room, switchgear room, boiler room, equipment room, rest rooms, and personnel tunnel. Supply air dampers to the boiler room automatically close during a fire by melting of fusible links. The fire dampers to the diesel and electrical rooms must be closed manually in order to maintain adequate diesel combustion air. The personnel tunnel has an independent exhaust fan and pressure controlled damper.

Heated air in the winter and evaporatively cooled air in the summer are supplied to the coupling station, service room, hallway, and decontamination rooms. An exhaust blower pulls air from the service room through the hallway and into the decontamination area. Fire dampers in the air supply to the decontamination area and coupling station are interlocked with the fire protection system.

The Test Cell is heated to a minimum of 60°F in winter with a 1,000,000 Btu space heater circulating inside air. A 12,000 cfm blower exhausting to the stack provides 12 air changes per hour when ventilation of the Test Cell is required. Outside air is supplied through building infiltration.

The Fuel Transfer Building is heated with a thermostatically controlled steam unit heater. The guardhouse has an electric unit space heater with manual or automatic control.

(5) Liquid Waste. The Test Cell floor is sloped to a drain trench traversing the center of the cell floor. Liquid waste flows by gravity from the trench into a 15,000 gallon hot waste storage tank via a 10 in. line. Maximum flow through this system has been calculated at 630 gpm. A hot waste coarse-filter sump is installed in the 10 in. line between the building trench and the hot waste storage tank with a removable screen basket inserted in the filter sump to

collect radioactive debris 3/16 in. diameter and larger. The screen, pipe, and trench are cadmium coated to prevent criticality in the event damaged fuel is washed into the drain system.

An annunciator system alarms when the waste tank level reaches two-thirds full and again when it is full. The alarms are monitored in the weather and radiation instrument room.

One 2 in. line from the decontamination area, and one 2 in. line from the trench in front of the coupling station drain into the 10 in. line terminating in the hot waste tank. The portion of the 10 in. line under the Test Cell is constructed of stainless steel while that part outside the building is carbon steel. The storage tank is carbon steel coated on the inside with one coat of Phenoline primer No. 300 and two coats of Phenoline No. 302.

The method of liquid disposal from the hot waste storage tank is determined by sample analysis of the tank contents. The quantity and activity levels of the wastes will dictate disposal by one of two available methods:

(1) transport by tank trucks to the Chemical Process Plant (CPP) or to the evaporator in the TSF area, or

(2) pump to grade. (The 2 hp hot waste pump delivers 50 gpm to grade or to the truck station outlet.)

(6) Communications. Telephone, intercom, and radio make up the communications network at the IET. The telephones are provided and serviced by Mountain States Telephone and Telegraph Company. A radio transmitter receiver unit is located in the control room for communication with the shielded locomotive, the hot shop, and roving mobile units outside the IET. Emergency battery power is available to all communication systems in event of a commercial power failure.

2. Examination Area

The Examination Area, designated as Building 607 is essential to the conduct of the SNAPTRAN test program and the accomplishment of program objectives. The technical support facilities housed in the building, namely, the hot shop, hot cells, metallurgical and chemical laboratories, etc., are of primary importance and will be used as follows: perform remote maintenance on the reactor package, perform chemical, physical, and metallurgical examination of the reactor fuel during the kinetic tests and following the destructive test, and charge NaK to the reactor system for the destructive test. Other facilities and services in the complex such as the cold assembly area, decontamination facility, etc., will also be utilized. The facilities and equipment in Building 607, including the rolling stock for transporting the reactor between IET and Building 607, are described in the following sections. The floor plans for the building are shown in Figs. 28, 29, and 30.

a. Hot Shop

(1) Construction. The hot shop is an area 160 ft long x 51 ft wide x 67 ft 6 in. high. The east end of the hot shop is partitioned to form the special equipment services cubicle. This cubicle is used for repairing the overhead manipulator and overhead crane when the hot shop contains radioactive material.

The west wall contains the locomotive and rolling stock entrance door. It is a sliding, bi-parting concrete door with a staggered joint. Movement of this door is accomplished by electric motors located outside the hot shop and controlled from the control console.

A personnel labyrinth is provided in the southwest corner for access to the hot shop. A monitoring and change room is located just outside this labyrinth.

The hot shop floor is designed to sustain a uniformly distributed load of 250 lbs/sq ft over the entire floor area. It is permissible to position a load equally distributed on four legs (space 10 ft x 12-1/2 ft), not to exceed 25,000 lbs per leg.

(2) Turntables. Two turntables which are essentially flush with the floor have been provided to rotate radioactive devices so that work may be viewed from the hot shop windows. Remote control allows rotation and automatic selective indexing of the turntables.

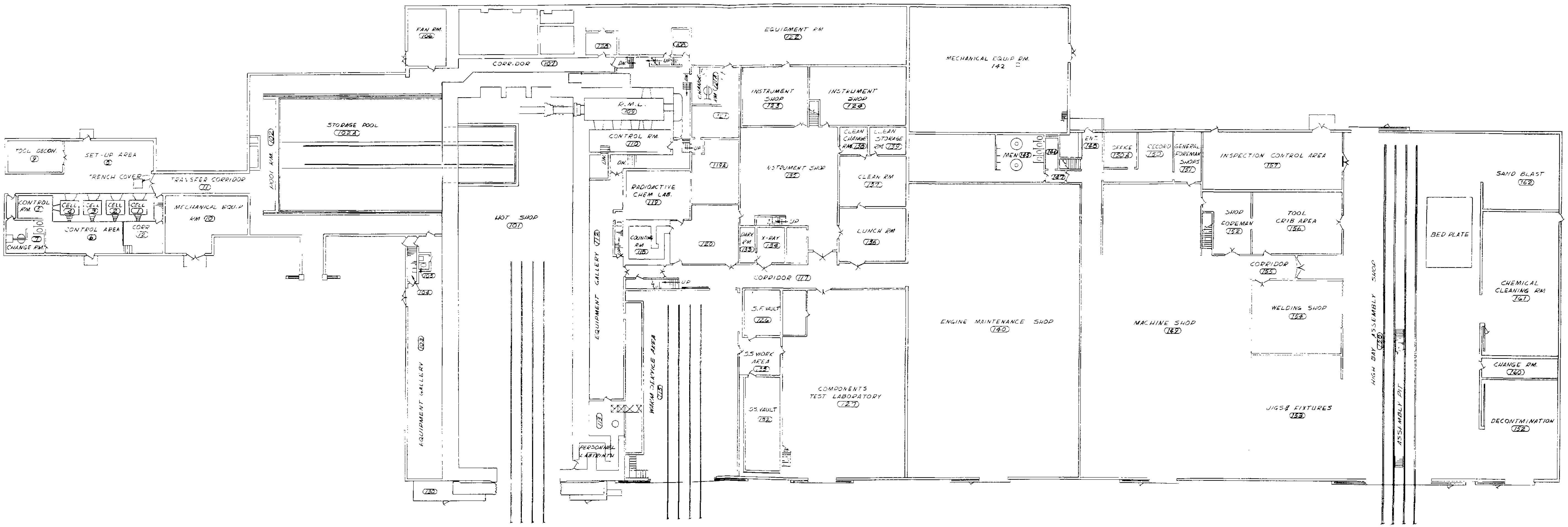


Fig. 28 - TAN Building 607 - First Floor Plan

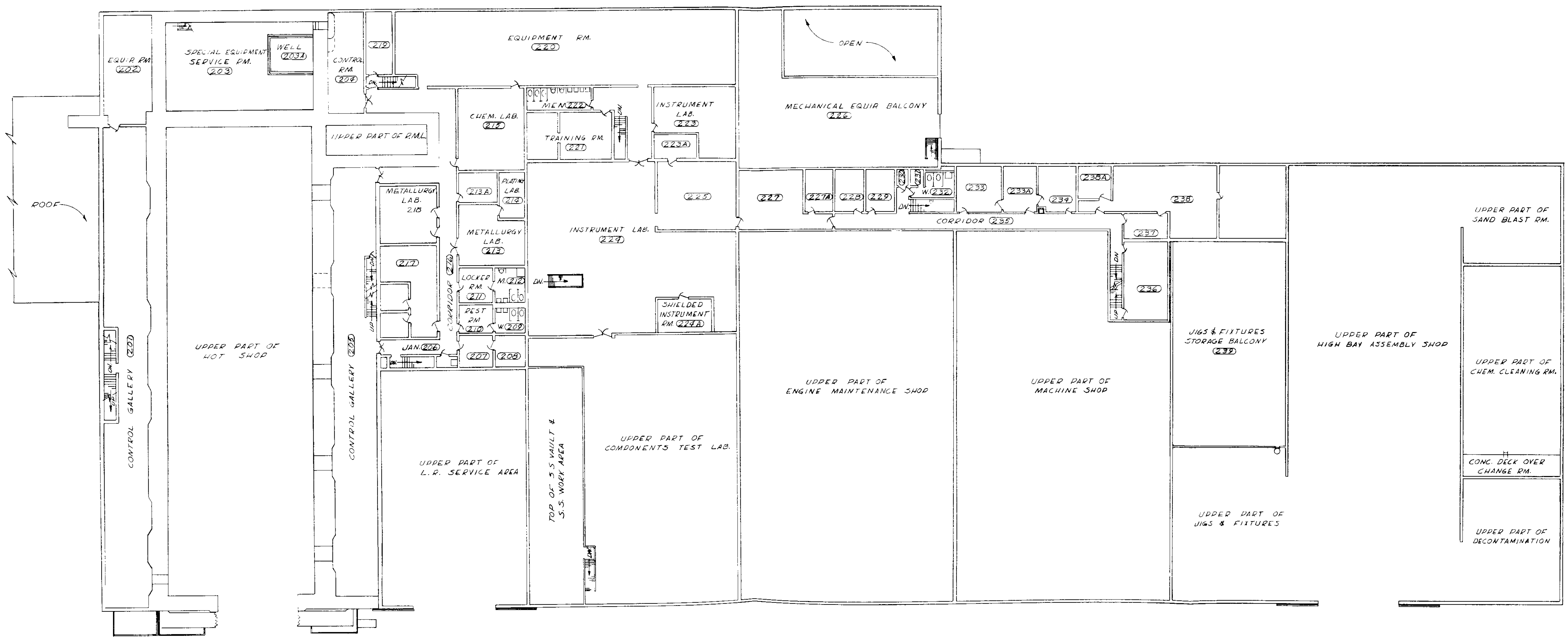


Fig. 29 - TAN Building 607 - Second Floor Plan

Each turntable has an overall diameter of 17 ft 6 in. and is capable of supporting a 60 ton load. The outer edge of the turntable is provided with a stainless steel skirt which revolves in a water-filled trench, providing an airtight seal between the turntable pit and the hot shop.

Two control stations are provided for each turntable and are located at windows adjacent to the turntable in the upper and lower operating galleries. Controls for directional rotation, speed and indexing are located at each control station.

(3) 100 Ton Crane. The hot shop heavy crane is a 100 ton overhead, single dolly, double-hoist crane especially equipped and adapted for operation by remote control. It is used to lift, hold, and transport heavy assemblies, tools, and fixtures, and to retrieve other types of handling equipment should they become inoperative.

Should electrical failure occur to the hoist mechanism, the hazardous load may be lowered by an auxiliary electrically operated hydraulic system. The hydraulic pressure can be directed to release both the mechanical load brakes and the solenoid holding brakes on both the main and auxiliary hoists.

The control stations for the heavy crane are located at the hot shop windows and are interlocked so that only one station may have control of the crane at a given time and control cannot be "stolen" from the operating station by a standby station. Each control station contains all controls necessary for the remote operation of the crane.

(4) Manipulators. The remote handling equipment in the hot cell is of the general purpose type. It was engineered and supplied by the General Electric Laboratory for the General Electric Company, Idaho Test Station. This equipment has been engineered to service a variety of "hot" mechanisms with the greatest possible versatility, and to handle future designs with a minimum of modification.

The hot shop contains two wall-type manipulators on each long wall and a heavy-duty overhead manipulator, all of which can be coordinated to work together. The heavy-duty overhead manipulator also serves as a crane follower.

(a) Heavy-Duty Overhead. The remote controlled overhead manipulator is a heavy-duty, rectilinear type machine designed to use various types of tools.

Mounted on the manipulator trolley are telescoping tubes which may be rotated through a distance of 350° in either direction from their neutral position. The telescoping tubes may be extended or retracted a distance of 25 ft at a maximum speed of 25 fpm. A drum-and-cable system provides the power for extension and retraction. If a power failure occurs in one of the drum drive motors, the other is powerful enough to lift the tube to enter the service area. If both motors fail under load, spring loaded brakes prevent the load from falling.

Attached to the telescoping tube is a clevis, arm, wrist and general purpose hand. A load of 3000 lbs can be lifted with the arm and hand in a straight-down position. A detachable hook on the clevis can lift loads up to 5000 lbs.

The overhead manipulator is controlled from one of the four control consoles located in the control galleries. The consoles are interlocked so that only one has control of the manipulator at a time, and control cannot be "stolen" from one console by another. The control console contains all controls necessary for operation of the manipulator and also contains indicators which show gripping force, rotation, and wrist position.

(b) General Mills Wall-Type. The wall manipulators consist essentially of General Mills, Model C rectilinear manipulators mounted on a track and boom system. These standard manipulators have been modified to include wrists and wrist torque indicators. Their control consoles have been modified to include the controls for the track and boom system.

The track and boom system is designed to operate on the two long walls of the hot shop, (two manipulators for each wall) at a maximum distance of 24 ft from the wall. Each manipulator operates on its own boom system, but each pair operating on one wall have in common the horizontal track system for that wall. The bottom of the horizontal boom on which the manipulator operates will clear an elevation of 30 ft from the floor in an "up" position, and in the "down" position will permit the operation of the manipulator hand at the floor. The boom will swing in a horizontal plane about a pivot to the wall line.

Each wall manipulator and its supporting boom can be controlled from a control console located in any viewing window on the same side of the hot shop. The control system is designed to enable transfer of control from one console to the next, but at the same time is interlocked to prevent accidental transfer of control.

Each manipulator is provided with a hook hand and a parallel jaw hand. Tools designed and supplied to be used with these manipulators are as follows:

(1) impact tools in three sizes,

(2) socket wrenches and extension bars,

(3) special extension bars,

(4) welding torch, and

(5) portable tool racks.

One of the wall manipulators has been modified by replacing the boom with a Grade-All hydraulic telescoping boom to provide more versatility and an extended reach.

(c) Argonne Master-Slave. A pair of Argonne master-slave type manipulators are mounted in 10 in. tubes passing through the hot shop wall above one of the hot shop windows. These manipulators are designed for a 6 ft 10 in. wall thickness, for work loads in the order of 4 lbs per hand, and for the radioactivity levels encountered in the hot shop. They are provided with positive locks so they may be locked in any position desired by the operator. These manipulators are adaptable for small,light work where some sense of feel is necessary. Either the wall manipulators or the heavy overhead manipulator may be used to deliver work to the master-slave manipulators.

(5) Control Console. The dispatcher's control console provides a central unit for remote operation of the outdoor turntable, railroad signal, hot shop door, outdoor viewing, communications control for the locomotive, and control of emergency power. This unit is divided into three basic sections:

(a) Selective Power Controls. Provision is made for controlling the power supply to the various electrical equipment in the hot shop building during emergency power operation. Selected hot shop equipment may be operated to the capacity of the emergency power supply. An ammeter is provided on the selective power panel for the dispatcher's guidance as to the load on the system.

(b) Communications System. A communications center for the remote facilities operation is provided at the dispatcher's control console. Through this radio system, the operator is able to communicate with the IET control room and with the shielded locomotive operator.

(c) Railroad System Control. Since the dispatcher will have control of the shielded locomotive operations, the control console contains many controls for the railroad system. Miscellaneous controls necessary for the locomotive system are found on the control console as follows:

(1) hot shop door controls,

(2) alarm system controls,

(3) turntable rail sanding controls, and

(4) turntable floor light controls.

Television monitors and controls on the control console allow the dispatcher to observe the movements of the locomotive as it approaches the turntable and the hot shop. One camera is mounted near the locomotive turntable and the other in front of the hot shop.

The indicators and controls necessary for the operation of the railway turntables are displayed in a pictorial layout on the control console. A selsyn indicating system is used to determine when the turntable and a particular track are in alignment. Block signal lights indicate the approach of the locomotive within a 100 ft distance of the turntable periphery.

(6) Drain System. There are six drains in the hot shop and two in the special equipment services cubicle for contaminated liquid waste. The northwest corner drain and the one near the personnel labyrinth connect to a 6 in. stainless steel pipe which goes to the liquid waste disposal building 616. The drains in the northeast corner, the end of the drain trench, and the two turntables, which consist of a pit drain and a gutter drain, connect outside the hot shop to a 6 in. stainless steel pipe which also goes to disposal building 616. The two 6 in. drain lines leading from the hot shop connect to a common sump. The waste is then pumped to three 10,000 gallon storage tanks. From these tanks, the waste is sent through an evaporator for concentration, and then disposed of by storage or discharge to a disposal well.

(7) Heating and Ventilating. The heating and ventilating system for the hot shop is integral with that for the warm services area and RML. Under ordinary circumstances fresh outside air is routed through a blower into the warm services area. The warm services area air is exhausted through three exhaust openings into a duct system by a 32,700 cfm exhaust blower to the intake of the hot shop supply system. The duct downstream of the 32,700 cfm blower has dampers and a gooseneck opening to the outside. By closing the duct damper and opening the gooseneck damper, an emergency negative pressure of 1/8 in. of water can be provided in the warm services area.

The hot shop supply system also has an outside opening which pulls 15,420 to 41,420 cfm of fresh air into the system. The air from the outside and the air from the warm services area then go through a filter, steam coil, damper, and air washer to a 41,420 cfm supply blower. The air supply then passes through a steam reheat coil and branches to the hot shop, the special equipment services room, and the RML. Exhaust air is discharged to the stack.

Air pressures are maintained at a negative 1/8 in. of water pressure in the warm services area and a negative 1/4 in. of water pressure in the hot shop, RML, and the special equipment services room.

(8) Fire Protection. Fire protection is provided in the hot shop by a fog system in the ceiling. This is a 4-zone system, with three zones in the hot shop proper and one zone in the special equipment services cubicle. Each zone operates independently by either automatic or manual action. All hot shop fire-water systems will be blocked off any time the reactor is in the shop.

(9) Services. Electrical and fluid services are brought into the hot shop on pedestals spaced along the walls of the hot shop. These services are remotely controlled at the hot shop windows where provisions are made for "on-off" switching, pressure regulation, and in some cases, flow regulation.

The service pedestals provide the following services: air, oxygen, acetylene, telephone, power, demineralized water, raw water, shield water, crane release, torch ignition, intercom system, television outlet, and camera shelf.

b. Special Services Equipment Area

The remote handling equipment in the hot shop has been so designed that maintenance of this equipment can be performed with a minimum of personnel contact. A special equipment services cubicle has been provided at the rear of the hot shop for maintenance of the overhead manipulator, overhead crane, and the General Mills manipulators. This service cubicle is provided with sliding shielding doors so that it may be shielded from the hot shop. The overhead manipulator and overhead crane may be brought to the services cubicle, decontaminated, and maintained by contact method. A special fixture has been provided so that the General Mills wall manipulators may be remotely removed from the track and boom system and transported by either the overhead manipulator or the overhead crane to the services cubicle.

c. Radioactive Materials Laboratory

The radioactive materials laboratory (RML) is located adjacent to the southeast corner of the hot shop. It is equipped and used for remote inspection, cutting, and other operations of a more delicate nature.

The RML periscope is provided for the close inspection of objects in the inspection cubicle. Objects under this periscope may be viewed with magnifications of 1X, 3X, and 9.6X powers and may be photographed by use of a camera attached to the periscope.

This area is serviced by four Model 8 master-slave manipulators, two Argonne #6 manipulators, and two General Mills type manipulators. Each master-slave manipulator services only a small area directly in front of its mounting window location, but the bridge mounted General Mills manipulators service the entire cubicle. A 3 ft extension hand has been provided to extend the work volume and usefulness of the General Mills manipulator.

The inspection cubicle cutting equipment is remotely operated Elox arc cutting equipment which has been provided for the remote cutting of small sections of radioactive material under controlled conditions to minimize the spread of contamination in the inspection cubicle.

Contact maintenance has been assumed possible in the case of the RML equipment. The design of the inspection cubicle and its equipment is based on the assumption that radioactive materials may be removed from the room, the room decontaminated, and contact maintenance performed. Acid resistant materials have been used in the construction of this equipment to facilitiate decontamination.

The RML has two floor drains with removable stainless steel screens and sediment baskets. These drains are used as contaminated acid drains and are 3 in. stainless steel pipe. Contaminated acid can be drained to an acid neutralizing sump where it can be neutralized before disposal to the three 10,000 gallon tanks located at Building 616.

The following services are furnished to the RML: acid, high pressure air, vacuum sampling line, raw water, mask air, demineralized water, and instrument air.

The fire protection system for the RML consists of six 100 lb CO_2 cylinders manifolded to four discharge nozzles in the RML ceiling. This system is automatic or manual with flow limited to 1950 cfm.

d. Storage Pool Area

A storage pool area located adjacent to the hot shop is used for transfer of material to and from the hot shop.

The storage pool area has a standard 15 ton, single trolley, single hoist, bridge crane for transfer of material within the storage pool area. The power to the crane is on the emergency power circuit and can therefore be controlled from the dispatcher's console. Normally it is controlled from a push-button pendant station on the trolley.

e. Hot Cells

The hot cell facility, consisting of four hot cells and miscellaneous work areas, was designed and built to perform post-irradiation examination of reactor fuel and mechanical components. The cell working space is divided into four equal areas, the type and degree of contamination that can be handled in each being slightly different.

High density concrete was used in the walls with normal density concrete used for the ceilings. The high density concrete walls between the cells reduces the maximum radiation contribution from one cell to another.

The change room is located along the only passageway from the contaminated area to the clean area. Hand-wash fountains and pass-through showers are available for personnel decontamination.

An intercom system is used both as a means of communication during installation and checkout of equipment and as a sound monitor during operations. A second intercom connects the control and setup areas with the remainder of the building.

f. Auxiliary Facilities

Also housed in the examination area are the following facilities: decontamination room, sand blast room, chemical cleaning room, storage area, components test laboratory, instrument and control laboratory, clean area, inspection and control area, high-bay assembly area, maintenance shop, machine shop, source and special materials vault, welding shop, warm-services area, and auxiliary facilities (locker room, shower area, etc.).

g. Rolling Stock

A shielded locomotive, which is a personnel carrier and prime mover to be used in the hot radiation field of the test area, is housed and maintained in the TSF area.

Water and lead shielding is provided for the cab to permit transporting of personnel. Entrance to the shielded cab can be made through a retractable hatch in the bottom of the cab. Operator stations are provided in front of glass-liquid windows at each end of the cab to allow the operator to view the track. Because of the somewhat limited vision from within, and because of the necessity of foolproof operations of the locomotive, the operator is provided two-way radio contact with the dispatcher in the hot shop. A signal light system at the railway turntable and a guidepost system along the tracks is also provided.

Many safety features have been built into the locomotive. In case of failure of the air system, the brakes of the locomotive are disengaged, thereby allowing some auxiliary source to pull the engine and its load. In case of failure of the power system on the locomotive, the remotely operated coupling system of the locomotive will fail safe so that the locomotive remains attached to its load. In case of emergency exit, one of the glass zinc bromide shielding windows is designed as an escape hatch. This window is latched into the shielded cab and may be unlatched and pushed along a monorail mounting to allow exit from the locomotive.

One other piece of equipment associated with the shielded locomotive is the 4-track test dolly. The reactor package will be mounted on the dolly for all test operations, and for transportation to and from the Test Cell (Fig. 2).

V. SAFETY ANALYSIS

A. Receiving, Handling, and Storage

The potential hazard of a criticality accident must be considered in the handling and storage of fissionable materials. The distinct activities involving the handling and storage of new fuel, irradiated fuel, and reflecting materials are considered in this section.

1. New Fuel

New Fuel will be transported under armed surveillance by Railway Express from Canoga Park, California, to Pocatello, Idaho. Phillips Petroleum Company will assume responsibility for transportation of the fuel by truck from Pocatello to the NRTS. Transportation of the fuel will be in accordance with procedures outlined in the Nuclear Safety Guide, TID-7016[13]. Five fuel rods will be shipped per shipping container. The container design is shown in Fig. 31.

The fuel will be received by Phillips Petroleum Company at an authorized accountability station to be established at the Test Area North (TAN) within the NRTS. The fuel will be placed in the storage vault at the Examination Area in the shipping containers. A fuel log showing the fuel assembly number, fissionable material loading, and date of storage in the vault will be maintained. Health Physics surveillance will be maintained at all times during the transfer to the vault.

The arrangement of fuel in the storage vault will be safe under flooded conditions, which is expected to constitute the worst credible accident in the vault. Neutron sources, contaminated fuel or "hot" fuel will not be stored in the "cold" fuel vault. Other fissionable materials not in the category of fuel assemblies will be packaged in "always safe" containers for storage in the vault. The vault loading and storage will be reviewed and approved by the Phillips Petroleum Company AED Nuclear Safety Committee (Section VI-A) and will not be exceeded or changed without prior written approval.

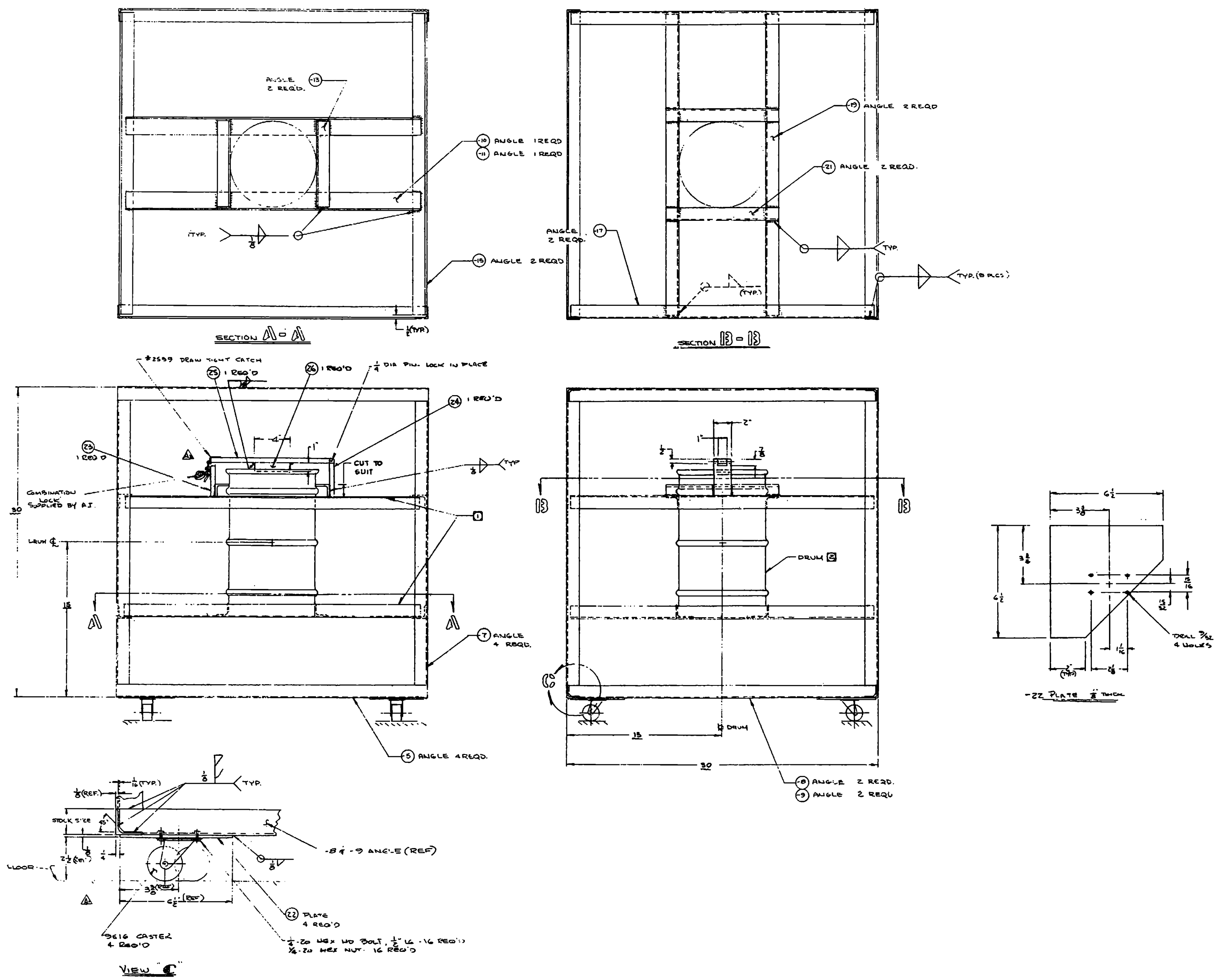


Fig. 31 - Birdcage Assembly for Fuel Shipping and Storage

Approval for all transfer of fuel to or from the reactor vessel or movement of materials in proximity of the reactor must be obtained from the Nuclear Test Group Leader. During core loading operations the fuel will be kept in the "always safe" shipping containers until the actual loading takes place. Only one fuel rod will be transferred to or from the vessel or container at a time. The Nuclear Test Group Leader will be responsible for all fuel handling in the Test Cell. His function will be to observe, in detail, all movements or manipulations. Monitoring instrumentation will provide audible indication of the neutron level.

It will, at times, be necessary to transport the SNAPTRAN 2/10A-1 package by rail between the IET and the TSF area with fuel rods loaded in the reactor vessel. To assure that the planned arrangement of the package is maintained, the drum drive clutches will be disengaged, all scram springs will be in an operative condition, and the drums will be "locked out" in their least reactive position. The shielded locomotive will have neutron level detectors mounted on the side closest to the test dolly and the level indication will be displayed in the control cab of the locomotive. As the package will be covered with a weather shield, it is not conceivable that water could come in contact with the beryllium reflector. Transfer of the package when containing fuel will not be made during extreme weather conditions, such as snow, hail, or rainstorms.

2. Irradiated Fuel

The procedures for handling irradiated fuel are identical with those for handling new fuel, except that remote handling techniques may be necessary. These techniques are not expected to create a criticality hazard.

Storage of irradiated fuel will be in a different vault from that used for new fuel storage. The same procedures for storage and transfer to and from the storage vault are to be used for irradiated and new fuel.

Damaged fuel will be loaded into shipping containers identical to that used for new fuel. The maximum allowable amount of fissionable material per container will be reduced to an amount equivalent to about three fuel rods. Transfer of the test package, which may contain damaged fuel, from IET to TSF following the Destructive Test is discussed in Section V-E.

Shipment of fuel from the TAN site will be in accordance with established procedures currently in use by Phillips Petroleum Company.

3. Neutron Reflecting Materials

The SNAP 2/10A reactor is a high leakage core ($\sim 40\%$ of all neutrons born in fission leak from the system) and is also highly undermoderated (H/U atomic ratio ~ 60). It is extremely sensitive to any reflector or moderator material placed near the core. Virtually any material placed close to the reactor will increase the reactivity of the system. For this reason, special precautions must be taken with respect to all operations that might result in changes in reflection capability, especially with respect to the most commonly available good neutron reflector - water.

All movement of materials in the vicinity of the loaded core will be under the supervision of the Nuclear Test Group Leader as specified in Section VI-A, especially reflector materials because of the high sensitivity of the core to these materials.

As discussed in the above named section, only one person will be allowed within a barrier placed at a radius of six feet from the reactor when the core contains fuel sufficient to constitute a critical mass. It is estimated from a "fat man" accident analysis that one person could not contribute more reactivity than $1.50 to the system.

To assure that no water will come in contact with the loaded core, all water supply systems in the Test Cell and hot shop will be disabled. In Section IV-B, the IET water supply in closest proximity to the reactor is described. This water supply, a fire hose reel located in the coupling station, will be isolated from the fire water system by a locked gate valve located below the coupling station. As this hose reel will be used only for decontamination, it will not be activated until after the destructive test.

All areas in the hot shop in which work is performed on the test package or reactor vessel while fuel is in the core will have their water systems deactivated and drained. No other extraneous reflector or moderator materials will be allowed in these areas. In addition, the reactor will at all times be protected from the weather, and the canal in the hot shop will be covered with a shield designed to prevent entry of the core into water.

Because of this sensitivity and the varying effect of different reflecting materials, measurements of the reflector worth will be made early in the experimental program. An estimate of the magnitude of the effects of an infinite reflector in close proximity to the core can be inferred from the following empirical equation based on data obtained from SNAP critical experiments:

$$\rho/\beta\ (\$) = 6 \left[\frac{1 - \sqrt{1 - \frac{r^2}{R^2}}}{1 - \beta\sqrt{1 - \frac{r^2}{R^2}}} \right]$$

where ρ/β (\$) is the maximum reactivity effect in dollars, r is the outside radius of the fixed beryllium reflector in inches (7.0 in.), R is the inside radius of the added infinite reflector in inches ($R - r$ = gap between fixed beryllium and added reflector), and β is the reflector albedo (~ 0.50).

B. Operator Error and System Failure

In analyzing the safety of the reactor system, the possibility of operator error and/or a component or system failure must be considered and the consequences evaluated. The considerations which enter into the safety analysis are discussed below.

1. Operator Error

a. Startup Accident

In order to prevent a startup accident during conduct of the SNAPTRAN tests, automatic protective instrumentation, redundancy in nuclear detectors and recorders, and special design restrictions to assist in administrative control have been supplied. The automatic protection instrumentation consists of a fast electronic scram system actuated by either a short reactor period (10 sec) or a high power level (1000 watts). Two channels of instrumentation capable of providing period or level scram have been supplied. In addition, a third neutron level channel and four channels of startup instrumentation (B-10 pulse counters) are provided. Key actuated interlocks have been included on the control console to prevent an unauthorized reactor startup and assist in the enforcement of administrative controls by insuring that supervision is present during startup. The possibility of operator error is reduced by standard procedures which require that the Assistant Operator and the responsible Supervisor be in attendance and at all times be cognizant of any operations affecting the reactivity of the system.

However, should a startup accident occur, the hazards to operating personnel are negligible, since during all nuclear operations operating personnel are located inside the shielded control room. The radiation dose rates to other on-site and off-site personnel will also be within acceptable levels. This conclusion is based on the results of a startup accident analysis in which it was assumed that reactivity was added at the maximum ramp addition rate ($0.16/sec), and that the period level trip failed to operate. The resultant energy release was calculated to be less than 1.0 Mw-sec for any scram actuation time less than 150 msec. The scram actuation time is estimated to be less than 25 msec, and for this reason, the energy release can be said to be negligible.

b. Incorrect Placement of Drums for Intended Transient

In order to initiate a transient test, one, two, or three of the control drums must be placed in a position such that insertion of the remaining drum(s) results in the desired reactivity addition. The magnitude of the nuclear excursion, therefore, depends upon the placement of the control drum(s). Since placement of the control drums depends upon human judgment, the possibility of an error must be considered. In order to minimize the possibility of an error, three independent calculations of the desired reactivity addition and the required drum positions will be made. In addition, before the transient test is initiated, the drum position settings will be verified by the Reactor Operator, the Nuclear Test Group Leader, and the Assistant Operator. By use of this procedure, the possibility of inserting an incorrect amount of reactivity is considered negligible.

However, even if the amount of reactivity added exceeded the desired amount, no hazard to operating or other personnel would exist, since operating personnel are restricted to the shielded control room and all other NRTS personnel are excluded from the test area during operation. The radiation dose to personnel outside the test area would be within acceptable levels.

c. Premature Transient Test Initiation

The SNAPTRAN 2/10A-1 reactor incorporates three modes of reactivity control: shim control, transient initiation, and scram. Because these three modes of control are combined in the same control drums, the possibility and consequences of prematurely or inadvertently initiating a transient test have been carefully considered. The control system design, as described in Section III-D, incorporates many operational interlocks which force the operator to adhere to a definite arming and fire sequence in order to initiate a transient test. An operator error could, therefore, result only in a premature, and not inadvertent, transient test initiation. To minimize the possibility of a premature initiation, administratively controlled key-switch interlocks have been incorporated at strategic locations in the logic sequence. These interlocks control the step and impulse drive cylinder gas supply, the arming circuits, and the sequence timer which initiates the firing sequence.

Although the premature initiation of a transient test is considered extremely unlikely, such an initiation does not constitute a hazard since evacuation of the test area would have already been effected.

d. Control Drum Movement

Personnel in the Test Cell will not be permitted to make checks or adjustments to control system components which involve drum movement when fuel sufficient to constitute a critical mass is in the reactor. Insertion of a control drum by the Reactor Operator when the Test Cell has not been cleared of personnel is prevented by administrative control of two console power key-switches under the control of the Nuclear Test Section and the Health Physicist. Nuclear test procedures require that the Health Physicist be in the Test Cell when any personnel are in that area and that the Health Physics console power key-switch be in his possession. In addition to key-switch control, scram actuators will be located in the Test Cell.

2. Control System Component Failure

One type of failure that must be considered in the safety evaluation of a reactor system is failure of the mechanical components of the control system. Several types of failure have been considered during periods of shutdown and operation, and those aspects of the control system and operating procedures which reduce the probability of such failures and prevent a failure from becoming a hazard are discussed below.

a. Failures When Reactor Is Shutdown

(1) Loss of Scram Condition. When the reactor is in the scrammed condition, failure of the scram mechanism does not constitute a hazard since the drums are already in the least reactive position. However, if the scram mechanism were disabled through failure of the scram spring, for example, it is possible that a drum could rotate into a more reactive position. However, a single failure in a scram spring will not disable scram as the spring is held in compression. It is not considered credible that a scram spring could

fragmentize in a single failure to the point where the compressive force would be lost.

In order to prevent such an occurrence, proper performance of the scram system will be maintained by continued checks of operational behavior and by technical surveillance from a member of the Experiments Section during any operations when the core contains fuel.

(2) Drum Drive Malfunction. Unintended insertion of drums by the motor drives requires simultaneous malfunctions in: (1) the two control power key-switches, (2) the spring-return-to-neutral insert-withdraw switch, (3) the motor selector switches, (4) the scram reset switch, and (5) the tooth clutch selector switches, all of which are functionally independent. The probability of this series of failures creating a hazard by allowing the insertion of two drums is not considered credible.

Unintended insertion of drums by the pneumatic drives would require simultaneous failures of: (1) two series-connected solenoid-operated normally closed inlet valves on each drive, (2) two parallel-connected solenoid-operated normally open vent valves on the step drives or two series-connected solenoid-operated normally closed vent valves on the impulse drives, and (3) administrative procedures requiring manual lockout of the bottled nitrogen supply during shutdown. Again, the probability of this series of failures creating a hazard by allowing the insertion of two drums is not considered credible.

b. Failures During Operation

(1) Scram System Failure. Operating procedures will include scram checkout on each drum prior to every startup. Each drum will be driven to the position of maximum reactivity from which it will be scrammed. The drive motor current will be monitored during the insertion to verify that the drive motor is working against the proper spring force. The time-history of the drum motion during scram as shown by the drum position recorder will be compared to previous tests to verify that scram action is normal.

Finally, the control system is so interlocked that it is not possible to have more than two tooth clutch clamps in use at any one time and that at least two drums are capable of being scrammed at all times.

(2) Mechanical Obstruction of Drum Motion. Obstruction of the drum motion by thermal expansion of the reactor during high temperature operation is prevented by idler wheels attached to the vessel structure which will bear against the drum shafts should a clearance problem develop.

Obstruction of drum motion from the lodging of foreign material between a control drum and the fixed beryllium will be prevented by careful mechanical inspection, cable dressing, cleanup, etc.

(3) Pneumatic Drive Failure. During the arming and firing sequence for an impulse test, there is a period of time, not greater than one second, in which a failure in the pneumatic system of the impulse drive could lead to a greater power excursion than expected. Such an excursion would require a decrease in pressure on the pneumatic drive piston thus allowing the drum to rotate more slowly, or a loss of pneumatic pressure in the scram lockout cylinders causing the scram lockout mechanism to relax and obstruct the rapid motion of the impulse drums.

Pneumatic failures associated with the initiation of a step test could result in smaller, but not greater, power excursions than expected.

3. Instrument Failure

Protection against instrument failure will be effected by frequent performance checks on neutron startup channels, power level channels and safety circuits, drum position indicators, and core temperature channels. In addition, redundancy is provided for all these measurements, i.e., there will be four completely independent startup channels, at least one linear power channel with indicator and recorder, one log power channel with indicator and recorder, three separate indications of drum position, and several process recorders on core temperature.

4. Electrical Failures

The reactor control system is designed to operate from a 24-28 volt dc power source. The control system, the operational instrumentation, and the drum drive motors are powered by a "failure-free" battery source. Charge is maintained by floating the battery on a constant voltage charger using commercial power. Loss of commercial power interrupts control room lighting and the operation of the battery charger. Emergency control lighting is provided automatically by battery-operated flood lights mounted in strategic locations.

C. Test Program Hazards

1. Introduction

The SNAPTRAN 2/10A-1 safety test program includes reactor kinetic tests. The factors of importance to evaluation of the reactor dynamic behavior and the hazards attendant to conducting the kinetic tests have been considered. The possible physical processes, calculational model, results of calculations and radiological hazards are discussed in this section.

One of the objectives of the SNAPTRAN excursion program is to experimentally determine the relationship between given reactivity insertions and such things as energy release. These relationships, which are generally referred to as self-limiting behavior, are the resultant effect produced by the combinations of various physical effects which act to accelerate or quench the chain reaction. These effects are usually called "shutdown mechanisms" and in the SNAPTRAN reactor include a prompt temperature coefficient, hydrogen moderator loss, and core disassembly. Despite the complexity of the interaction of these factors, the lack of detailed knowledge regarding some of the effects, and the simplifying assumptions necessary for calculational purposes, estimates of the reactor behavior can be made which permit an evaluation of the hazards associated with the proposed experiments. It should be emphasized that the experiments are intended to measure the quantities which, for the present, must be calculated to estimate the hazards of the experiments.

2. Shutdown Mechanisms

The first major reactivity effect mentioned above is the prompt negative temperature coefficient which is influenced by two principal factors: core expansion and a near-thermal spectral change. The first factor, that is core expansion, reduces the fuel and moderator atom densities within the core and also changes the neutron leakage from the core. In the case of the SNAPTRAN 2/10A-1 reactor, radial expansion of the fuel-moderator due to a temperature increase is limited during a rapid power rise by the inherent heat transport delay between the fuel and the fuel support members (grid plates and vessel) and, therefore, is not expected to contribute significantly to the self-shutdown process. However, axial fuel-moderator expansion is not limited and, therefore, may be expected to contribute to self-shutdown. The second factor, that is, a near-thermal spectral change, is determined primarily by the effect of temperature on the scattering cross section of the ZrH_x moderator and can also be expected to contribute to self-shutdown.

The second major reactivity effect is due to the loss of hydrogen moderator which is a result of the elevated temperature dissociation of hydrogen from the ZrH_x lattice, diffusion of gaseous hydrogen through the fuel-moderator material, and subsequent escape from the core. The escape of hydrogen requires that the internal pressure within the fuel rod be sufficient to rupture the cladding; however, rupture of the cladding does not necessarily lead to core disassembly. The movement of the hydrogen to regions of lower moderator worth or escape from the core volume will ultimately produce a large negative reactivity effect. On the other hand, a positive reactivity effect may be produced by the first stage of the above processes due to the change in scattering cross section in going from the bound state to the free gas state.

The third major reactivity effect is core disassembly, which is dependent on the assumptions made regarding the laws governing the hydrogen dissociation and diffusion rates. The various possible assumptions lead to three modes of core disassembly. In the first mode, the hydrogen diffusion through the fuel-moderator is assumed to be negligible during a short period transient, resulting in a hydrogen pressure buildup at dislocations (microscale imperfections) in the fuel-moderator. The pressure within the fuel-moderator lattice could become sufficiently great to cause a sudden disintegration of the fuel-moderator accompanied by a vessel rupture and complete core disassembly. In the second mode, the hydrogen diffusion is assumed to be instantaneous during a short period transient. Under these conditions, extreme pressures are rapidly generated within the fuel can (clad) resulting in destructive disassembly of the reactor fuel rods and vessel in addition to the escape of hydrogen moderator. A third mode of core disassembly is due to core meltdown. This could occur assuming sufficient heat is transferred from the fuel-moderator to the clad. Under these conditions, the cladding will melt before the internal pressure in the fuel can is large enough to bring about violent disassembly of the core.

It is apparent that these postulated shutdown mechanisms are not only dependent on the energy produced in the core but also on the time rate of increase in the energy. Further, the reactivity effects are not discrete mechanisms, but rather a combination of complex interrelated processes. The negative reactivity contribution from one or more of these interrelated mechanisms may, therefore, appear as the reactor undergoes progressively more severe excursions.

3. Kinetic Calculations

The evaluation of the hazards attendant to conducting the kinetic tests has been based on the calculations discussed in this section. The calculational model and the results of calculations with the SNAPKIN digital computer code are based on information obtained from Atomics International[14]. The additional considerations[15] to be examined may affect the details of the predicted system performance but will not affect the evaluation of the safety of the test program because an order of magnitude increase in the predicted energy release can be safely accommodated.

The axial expansion contribution to the prompt temperature coefficient has been calculated by Atomics International to be approximately -0.05¢/°F, independent of temperature. The magnitude of the spectral contribution depends on the nature of the hydrogen binding in the zirconium-hydride lattice. The nature of the hydrogen binding is not completely understood at present, but in the models chosen by Atomics International to describe this binding, the spectral contribution is certainly negative. Measurements of the total prompt temperature coefficient of the S2DR, which has the same core composition as SNAP-TRAN 2/10A-1, indicate that the magnitude of the net prompt temperature coefficient is not less than -0.05¢/°F.

An analysis has been made for a range of ramp and step reactivity inputs by the use of the SNAPKIN code developed at Atomics International. SNAPKIN solves the space independent reactor kinetics equations for power, fuel temperature, inverse period, energy release, and excess reactivity as a function of time for an arbitrary reactivity driving function. This analysis included two shutdown mechanisms, one of which was the axial fuel expansion coefficient, and the other a hydrogen loss mechanism. In the case of the latter mechanism, the model assumed that no hydrogen loss occurs until the fuel temperature has attained that temperature at which the static dissociation pressure is sufficient to exceed the ultimate strength of the cladding. This temperature (~ 1500°F) was obtained from work reported in NAA-SR-5890[16]. At that temperature the cladding is assumed to fail and further hydrogen loss from the fuel-moderator is then calculated by extrapolation of the diffusion rate data reported in NAA-SR-3700[17]. When the hydrogen escapes the fuel-moderator it is also assumed lost from the core. In this model, the rate of hydrogen loss (prior to actual melting of the fuel) is too slow to affect peak power or the energy release for the larger reactivity inputs considered by Atomics International in the transient analysis. For these inputs, then, an assumption of <u>no</u> hydrogen loss prior to fuel melting would not measurably change the results of the calculations.

Figs. 32 through 40 present the results of the SNAPKIN calculations in the form of plots of power, average fuel temperature, and energy release as a function of time for step reactivity insertions of $0.20 to $5.00 in which the starting power was 100 watts; and for ramp insertion rates of $0.005 to $0.16 per second, in which the starting power was one milliwatt.

For the larger reactivity inputs, it will be observed that the melting point (3200°F) of the fuel is attained. Since SNAPKIN at present does not incorporate a melting model, the calculation is terminated at this point. However, in order to estimate an upper limit for the magnitude of the energy release, it was assumed that the core temperature remains at the melting point as further fissions supply energy for the melting of fuel and the dissociation of hydrogen. The latent heat of fusion (8.5 Mw-sec) of all of the fuel and the dissociation energy of all of the hydrogen (83 Mw-sec) was added to the value of the energy release calculated up to the beginning of melting (78 Mw-sec). Due to the assumptions used, the results of these calculations, which are summarized in Table V-1, show that the maximum nuclear energy release, irrespective of reactivity input, is 170 Mw-sec.

TABLE V-1

RESULTS OF REACTOR TRANSIENT CALCULATIONS

Reactivity Input	Maximum Temperature °F	Energy Release Mw-sec
$0.002/sec ramp	1700	35
$0.005/sec ramp	1800	40
$0.01 /sec ramp	1900	43
$0.02 /sec ramp	1950	45
$0.03 /sec ramp	2000	47
$0.08 /sec ramp	2050	51
$0.10 /sec ramp	2100	51
$0.12 /sec ramp	2100	52
$0.14 /sec ramp	2100	52
$0.16 /sec ramp	2100	53
$0.20 step	800	11.5
$0.40 step	1500	27.5
$0.60 step	1670	34.6
$0.80 step	1800	40
$1.00 step	1950	45
$1.20 step	2050	49
$2.00 step	3200	170
$2.50 step	3200	170
$3.00 step	3200	170
$3.50 step	3200	170
$4.00 step	3200	170
$4.50 step	3200	170
$5.00 step	3200	170

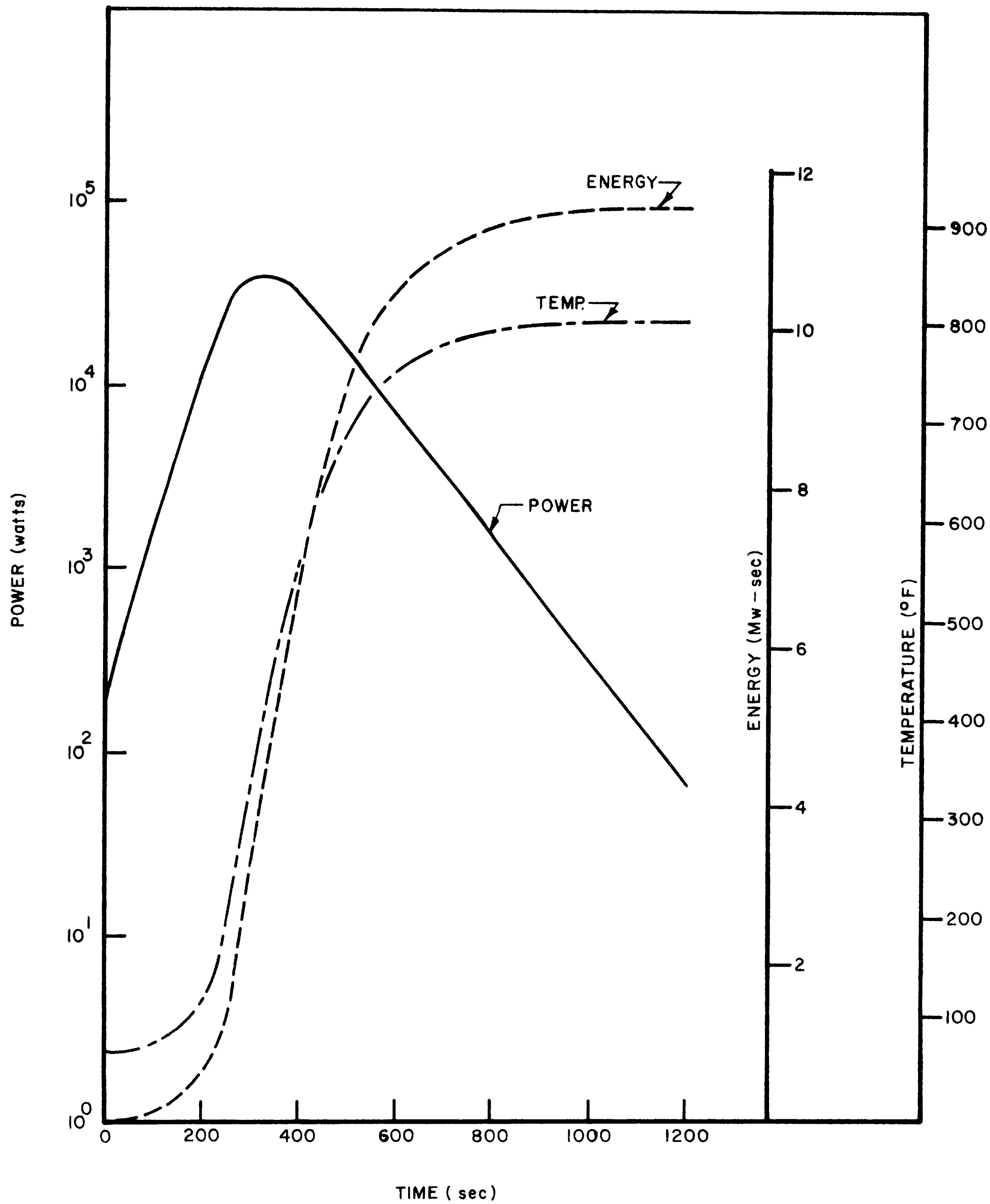


Fig. 32 - Calculated Power, Temperature, and Energy Curves For a $0.20 Step Reactivity Insertion

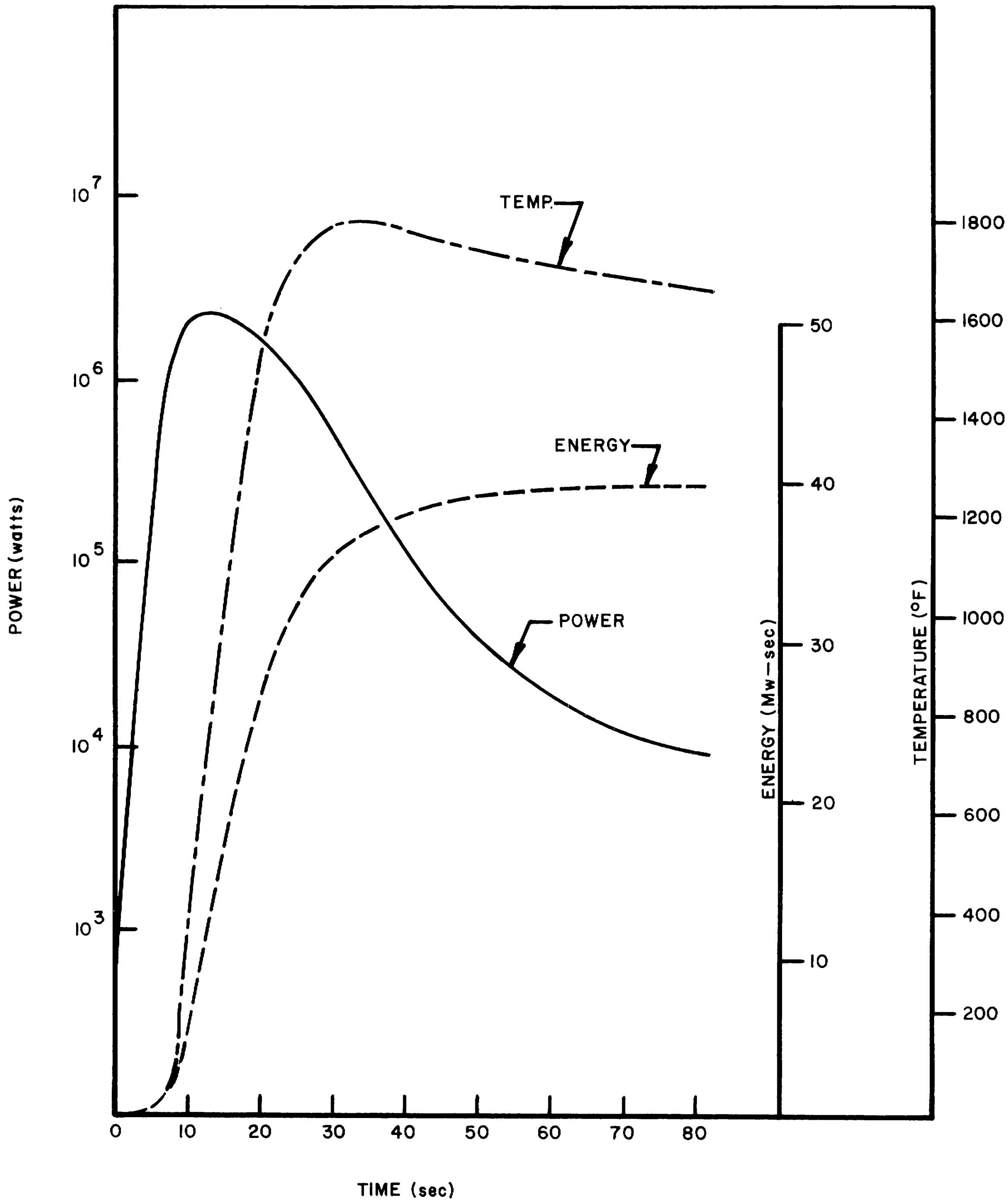


Fig. 33 - Calculated Power, Temperature, and Energy Curves For a $0.80 Step Reactivity Insertion

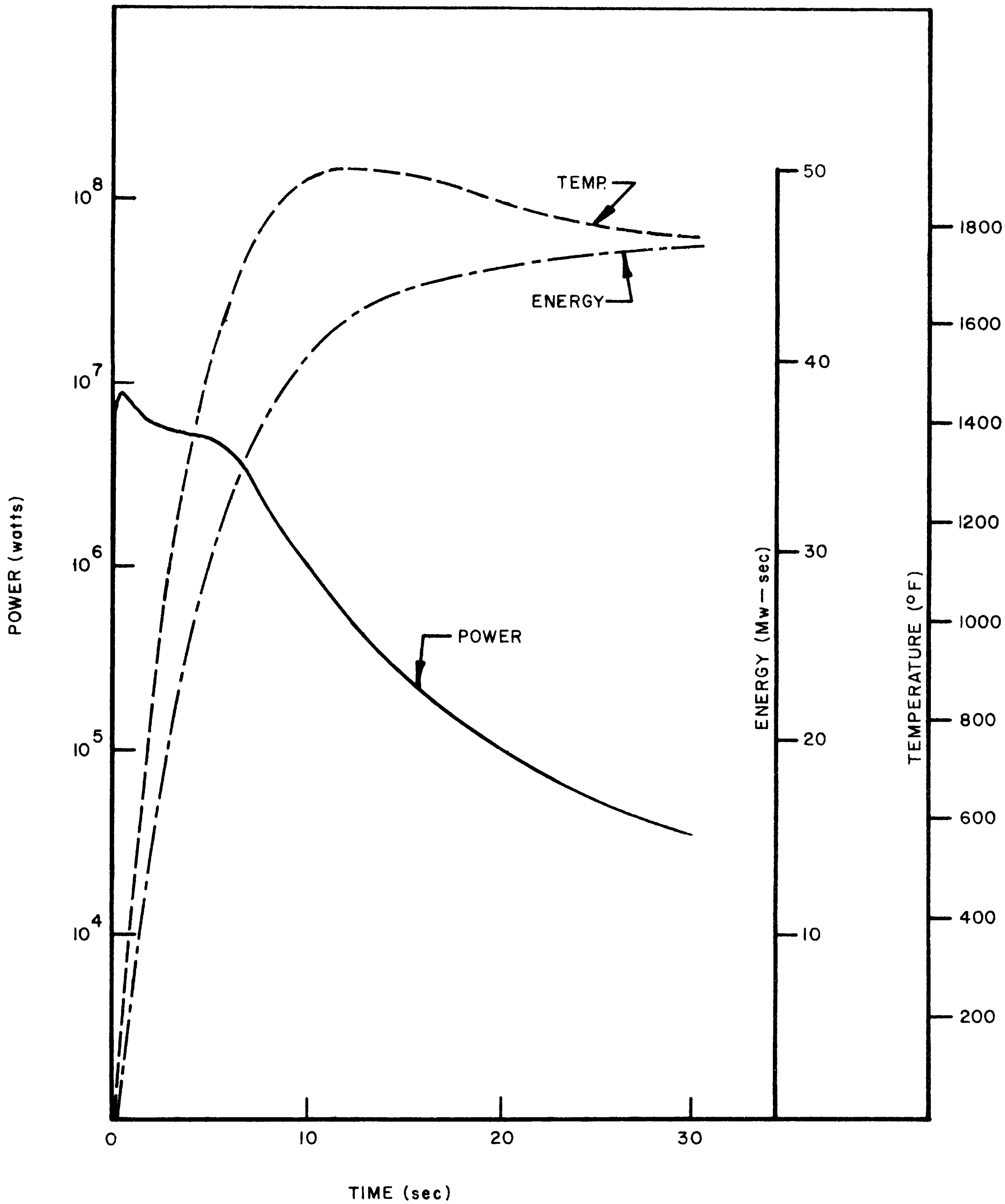


Fig. 34 - Calculated Power, Temperature, and Energy Curves For a $1.00 Step Reactivity Insertion

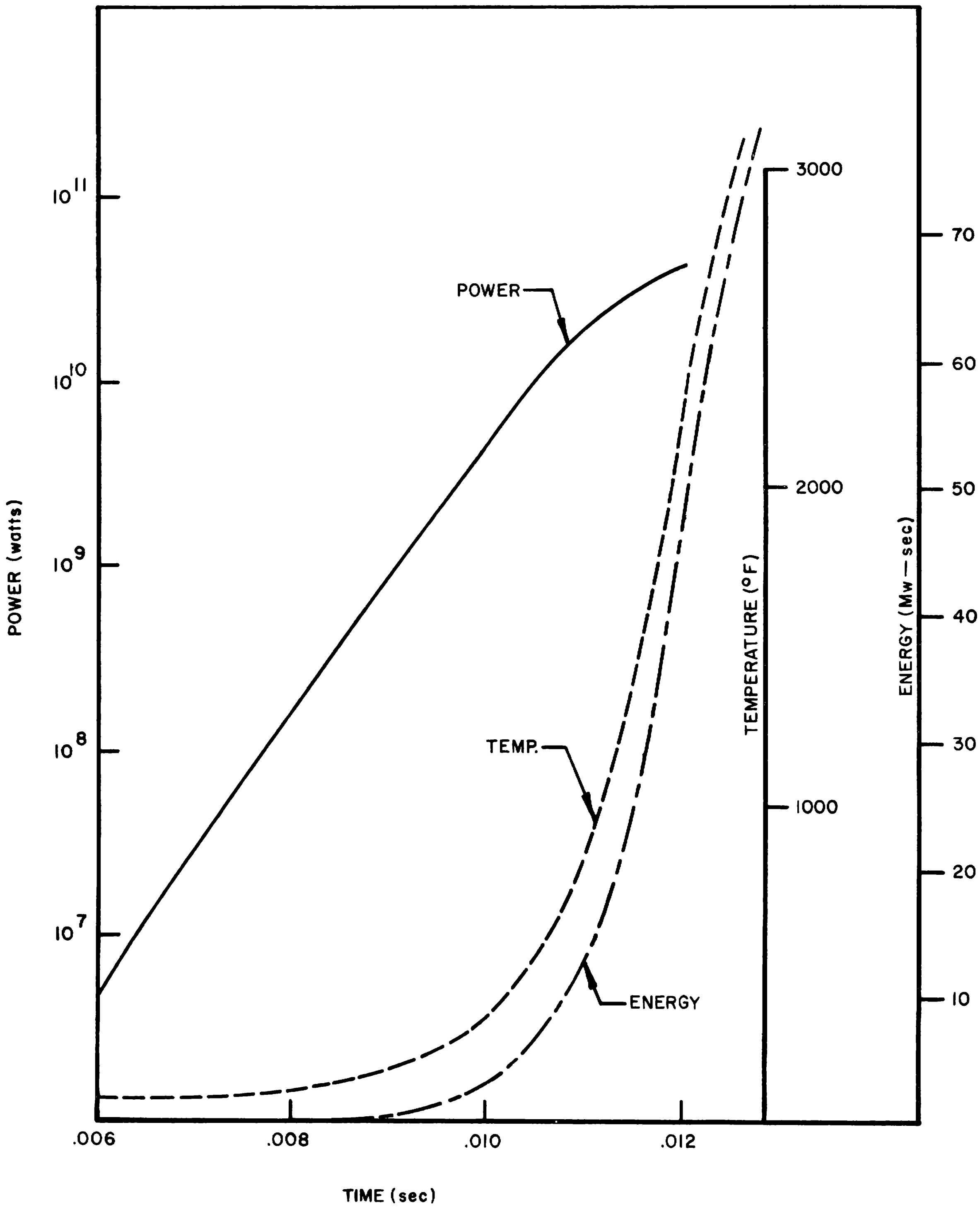


Fig. 35 - Calculated Power, Temperature, and Energy Curves
For a \$2.50 Step Reactivity Insertion

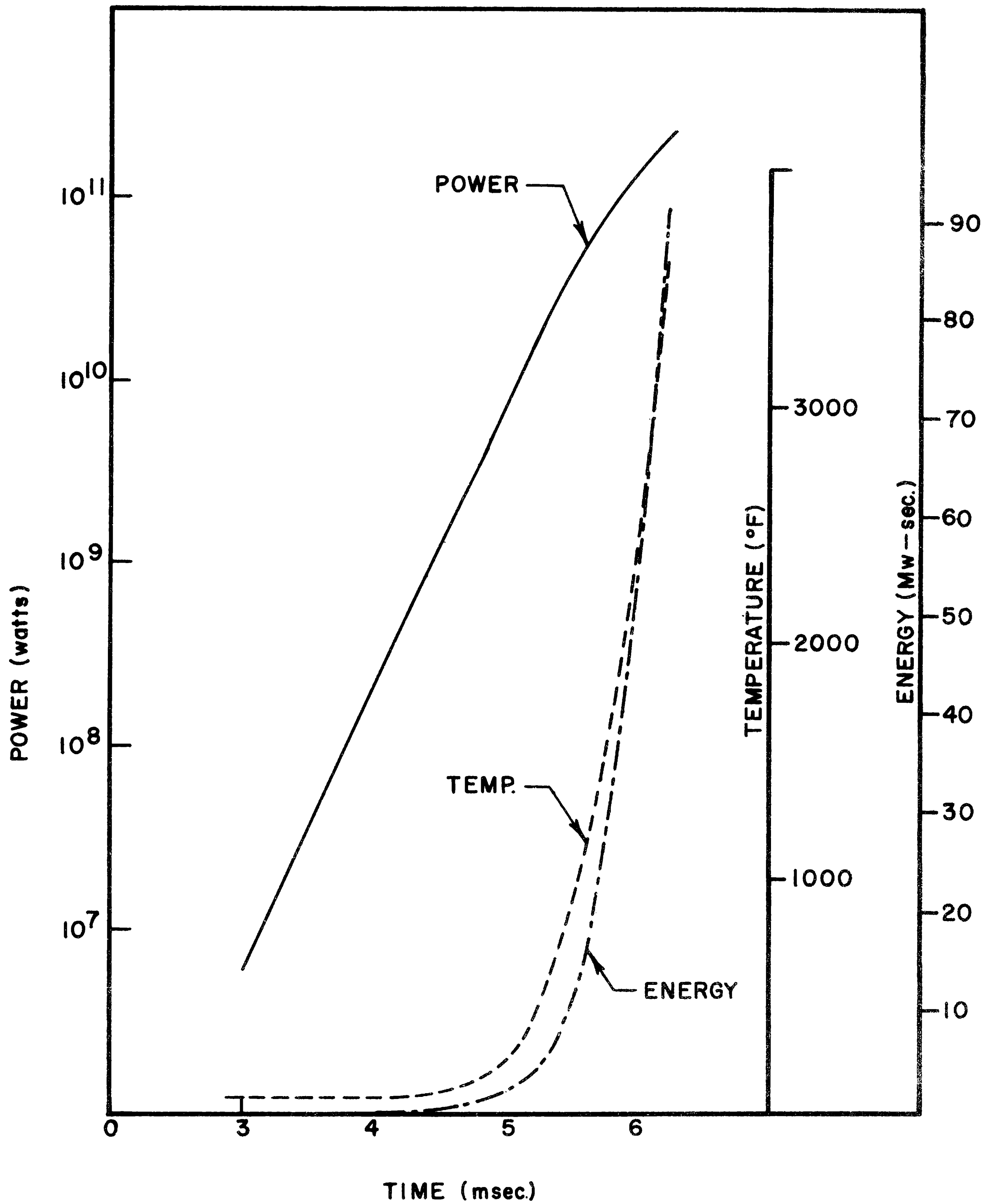


Fig. 36 - Calculated Power, Temperature, and Energy Curves For a \$4.00 Step Reactivity Insertion

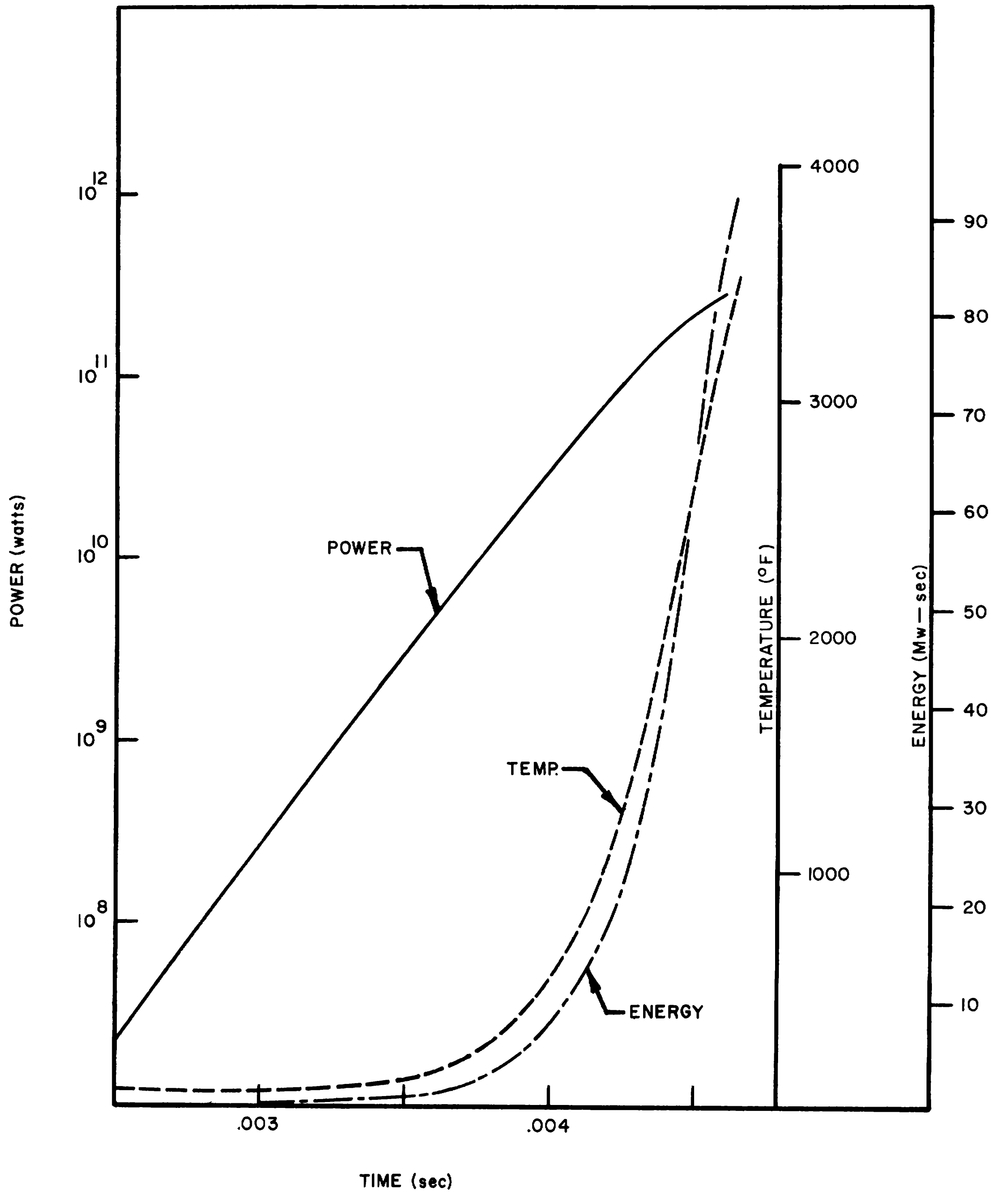


Fig. 37 - Calculated Power, Temperature, and Energy Curves For a $5.00 Step Reactivity Insertion

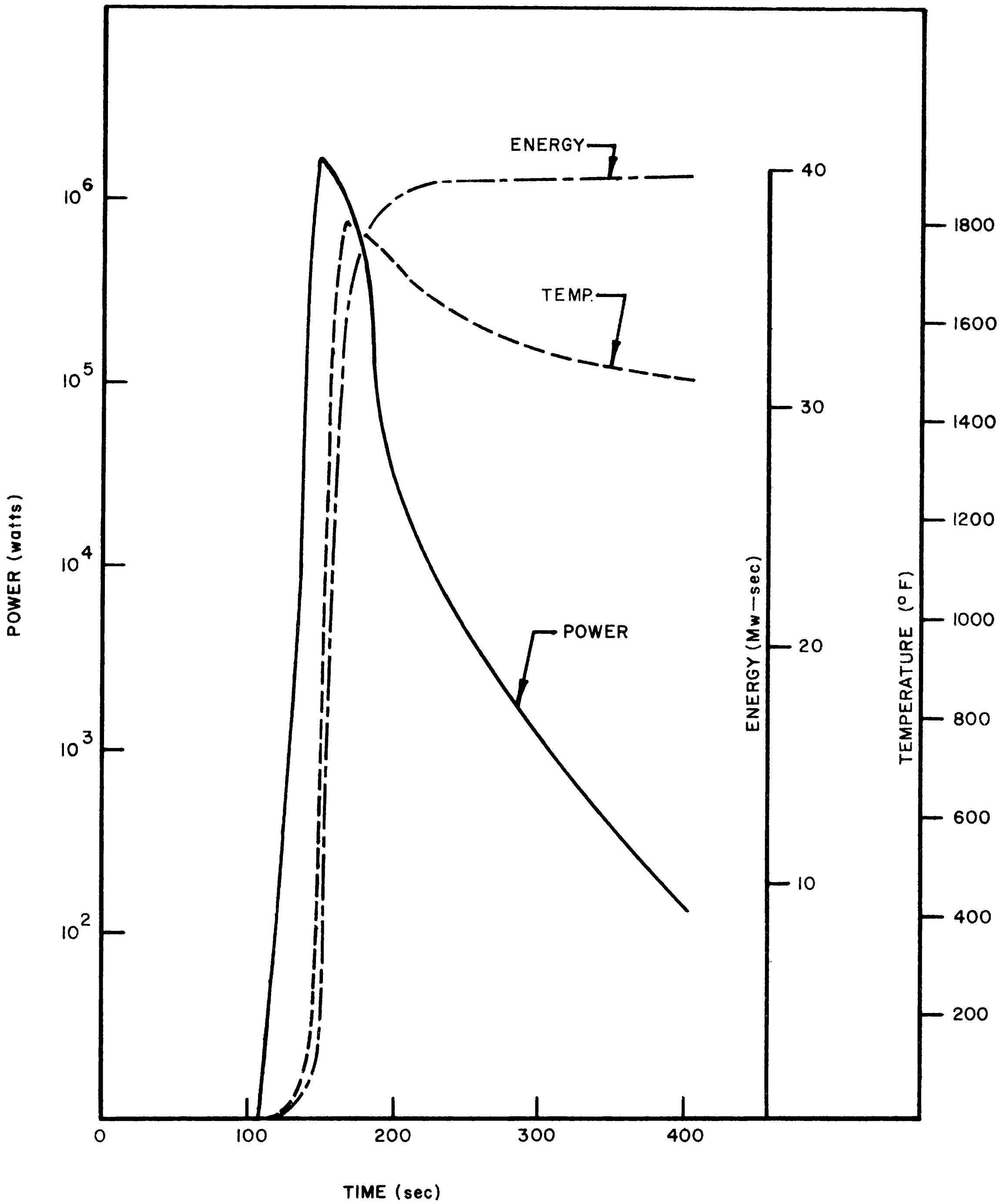


Fig. 38 - Calculated Power, Temperature, and Energy Curves For a $0.005/sec Ramp Reactivity Insertion

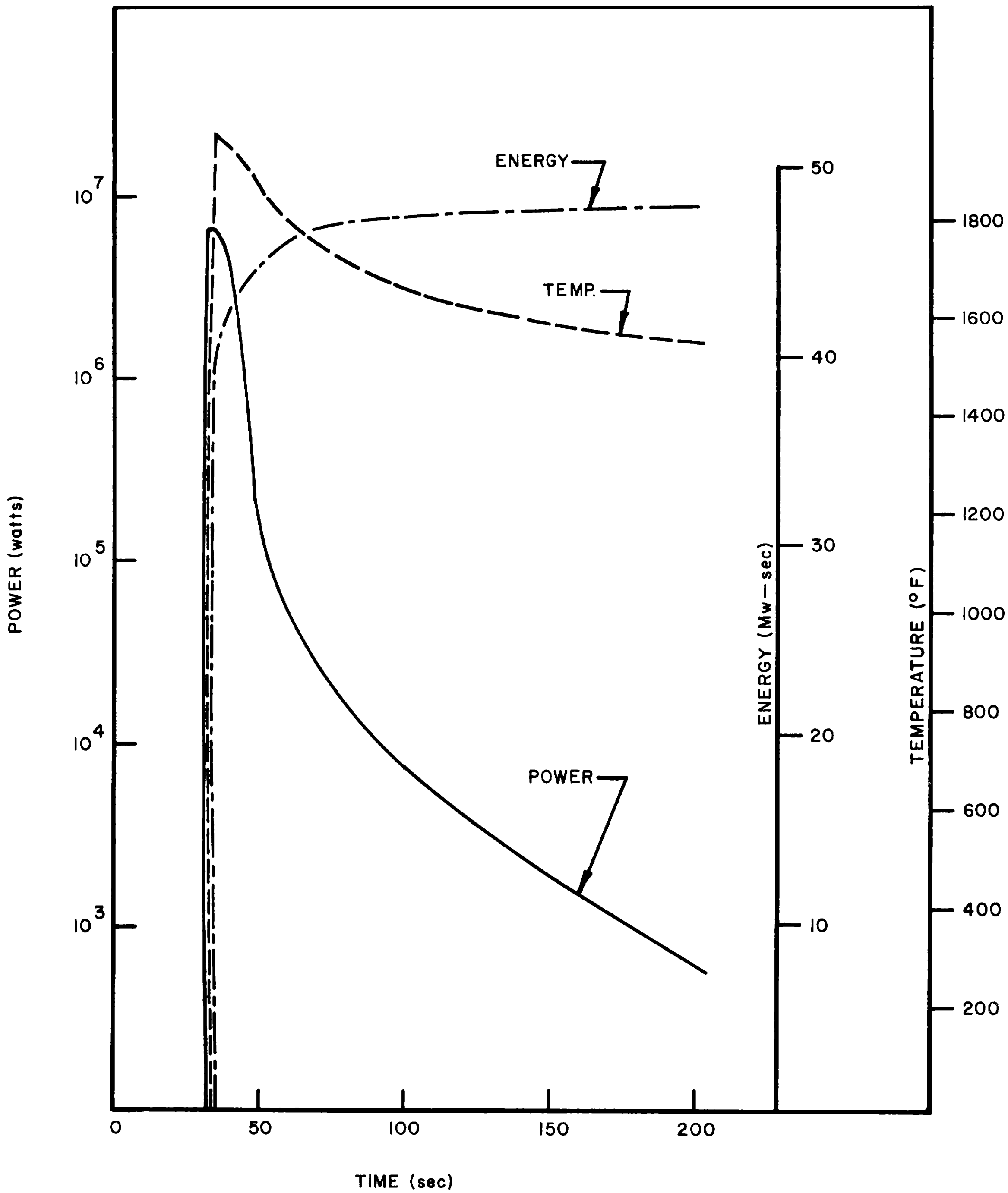


Fig. 39 - Calculated Power, Temperature, and Energy Curves For a $0.03/sec Ramp Reactivity Insertion

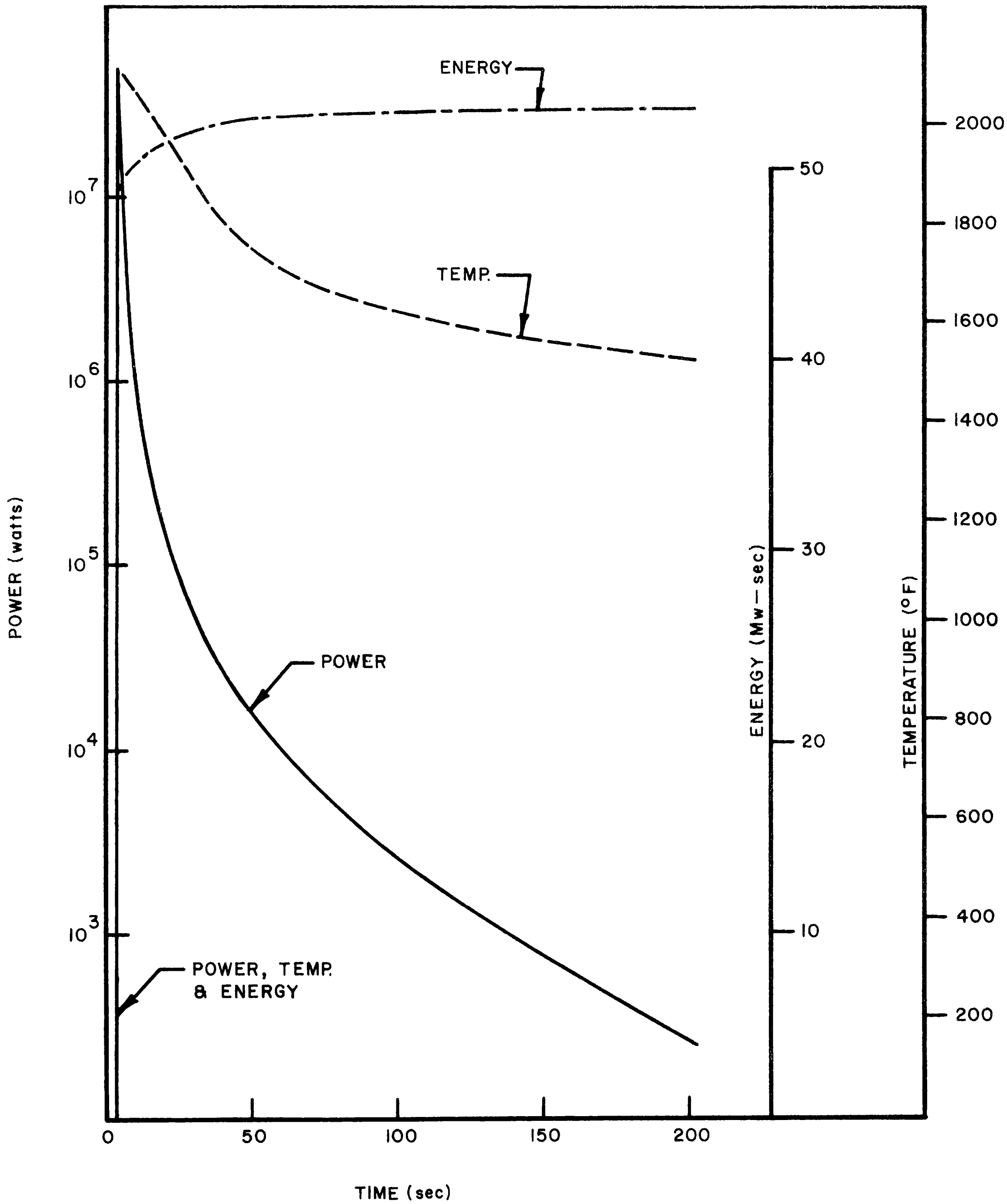


Fig. 40 - Calculated Power, Temperature, and Energy Curves For a $0.16/sec Ramp Reactivity Insertion

A digital computer code, "BOOMER", was also used by Atomics International to calculate the transient behavior (consisting of power, energy, hydrogen gap pressure, fuel rod temperature, and the hydrogen loss) of the SNAPTRAN core commensurate with a $5.00 reactivity step input. This code is based on a mathematical model of the SNAP 2/10A core which describes the heat transfer, reactor kinetics, and hydrogen release mechanism. In essence, this model consists of the six delayed group monoenergetic neutron kinetics equations, the spatially discretized partial differential equations and algebraic equations describing the heat transfer and hydrogen diffusion within an average fuel rod, chopped cosine flux distribution, distributed prompt temperature reactivity feedback, grid plate temperature reactivity feedback and the coupling equations.

One of the provisions in the model allows for the hydrogen loss from the fuel to become "power limited" when the fuel temperature exceeds some arbitrarily specified value. This provision was made in consideration of results of experimental work done for the purpose of obtaining hydrogen loss rates from uranium-zirconium hydride fuel at high temperatures. The results of electrical pulse heating experiments indicate, for those transients studied, that the hydrogen loss from the fuel specimen is proportional to the power input when the sample temperature reaches 2200°F. It is possible that, had larger power inputs to the pulse heated sample been available, the sample temperature could have increased above 2200°F and exhibited larger hydrogen loss rates. On the other hand, it seems improbable that the fuel temperature could increase beyond 3000°F before releasing the hydrogen at a "power limited" rate. In any event, the hydrogen could certainly not remain bound to the lattice throughout the melting process, as the melting itself will destroy the lattice.

Figs. 41, 42, 43, and 44 give the power, energy, peak temperature, and amount of hydrogen retained as a function of time in a $5.00 destructive transient. Fig. 45 gives the energy release as a function of step reactivity input for both the SNAPKIN and BOOMER calculations. The "power limiting" provision for the BOOMER calculations was activated at a temperature of 3000°F. The BOOMER calculational results were

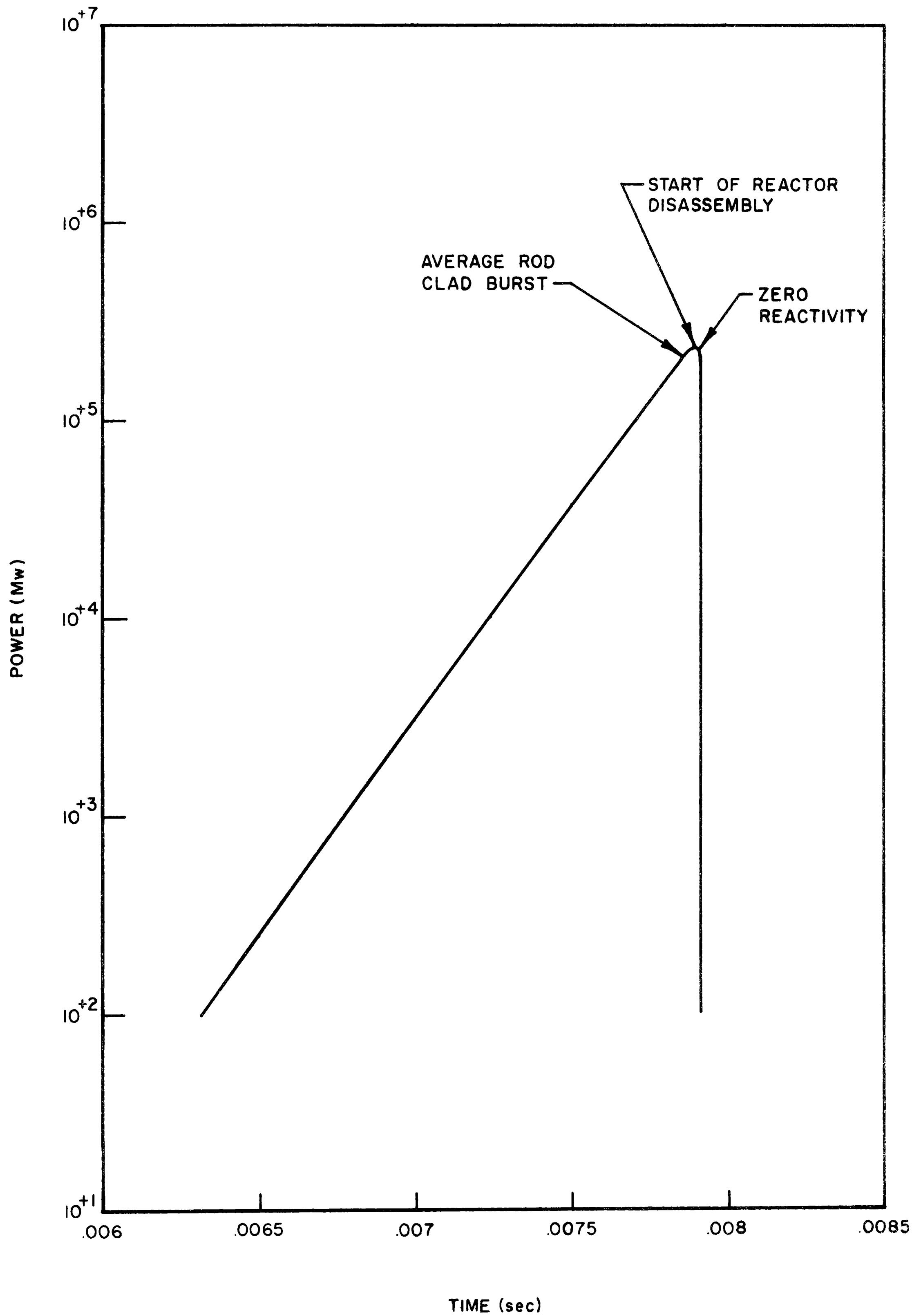


Fig. 41 - Calculated Power Curve For $5.00 Step Reactivity Insertion - "BOOMER" Code

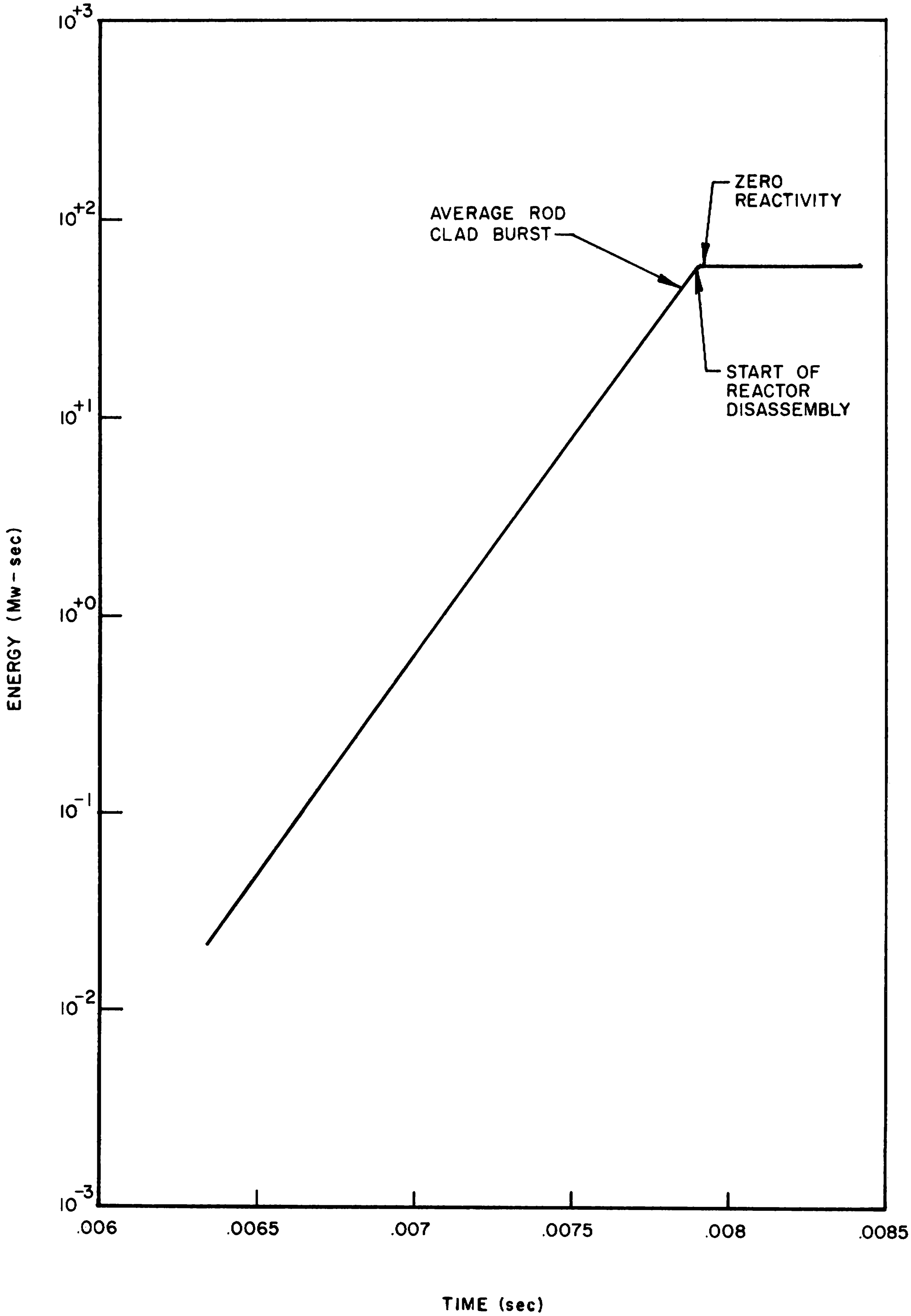


Fig. 42 - Calculated Energy Curve For $5.00 Step Reactivity Insertion - "BOOMER" Code

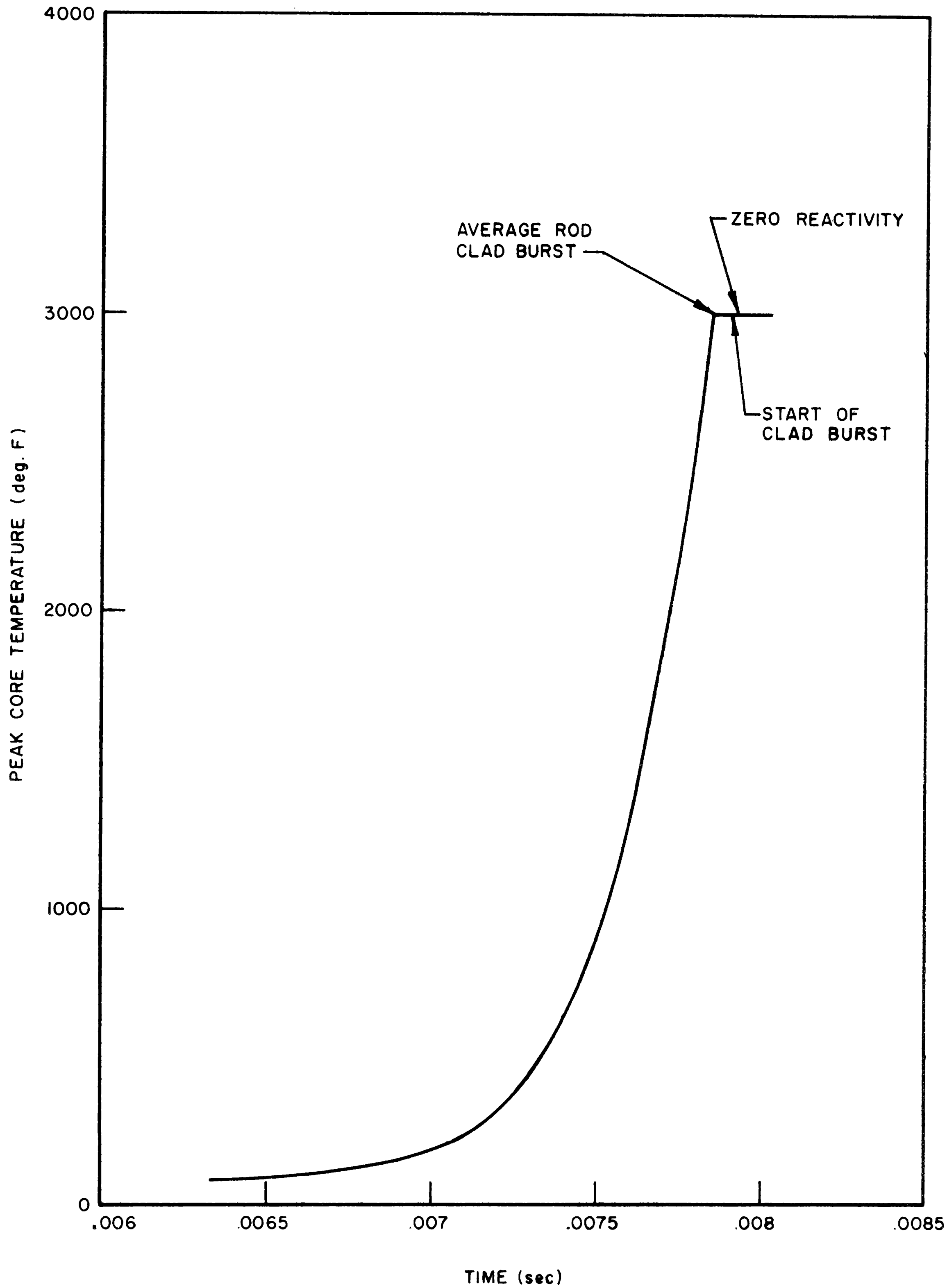


Fig. 43 - Calculated Peak Temperature Curve For $5.00 Step Reactivity Insertion - "BOOMER" Code

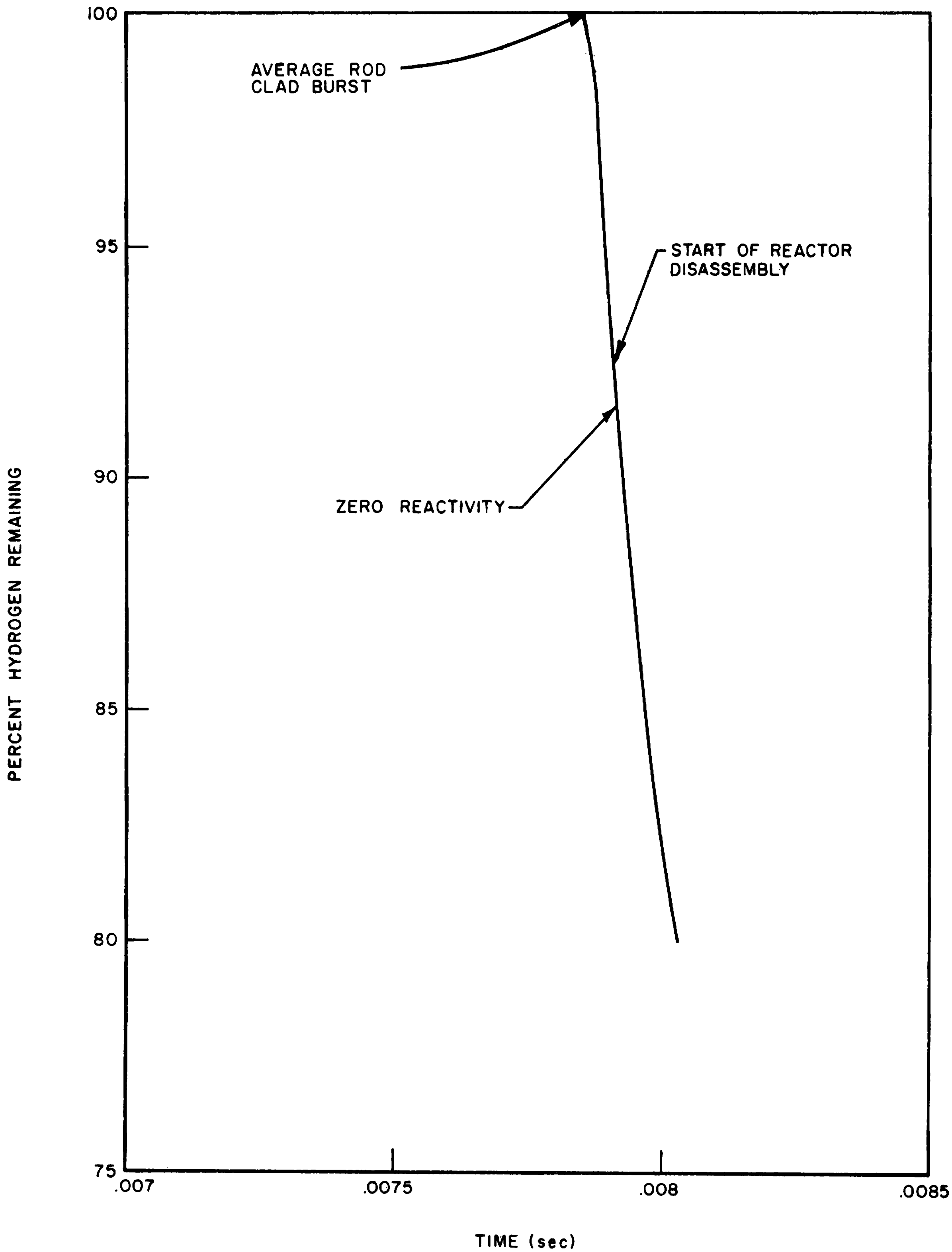


Fig. 44 - Amount of Hydrogen Retained in Core Following a $5.00 Step Reactivity Insertion

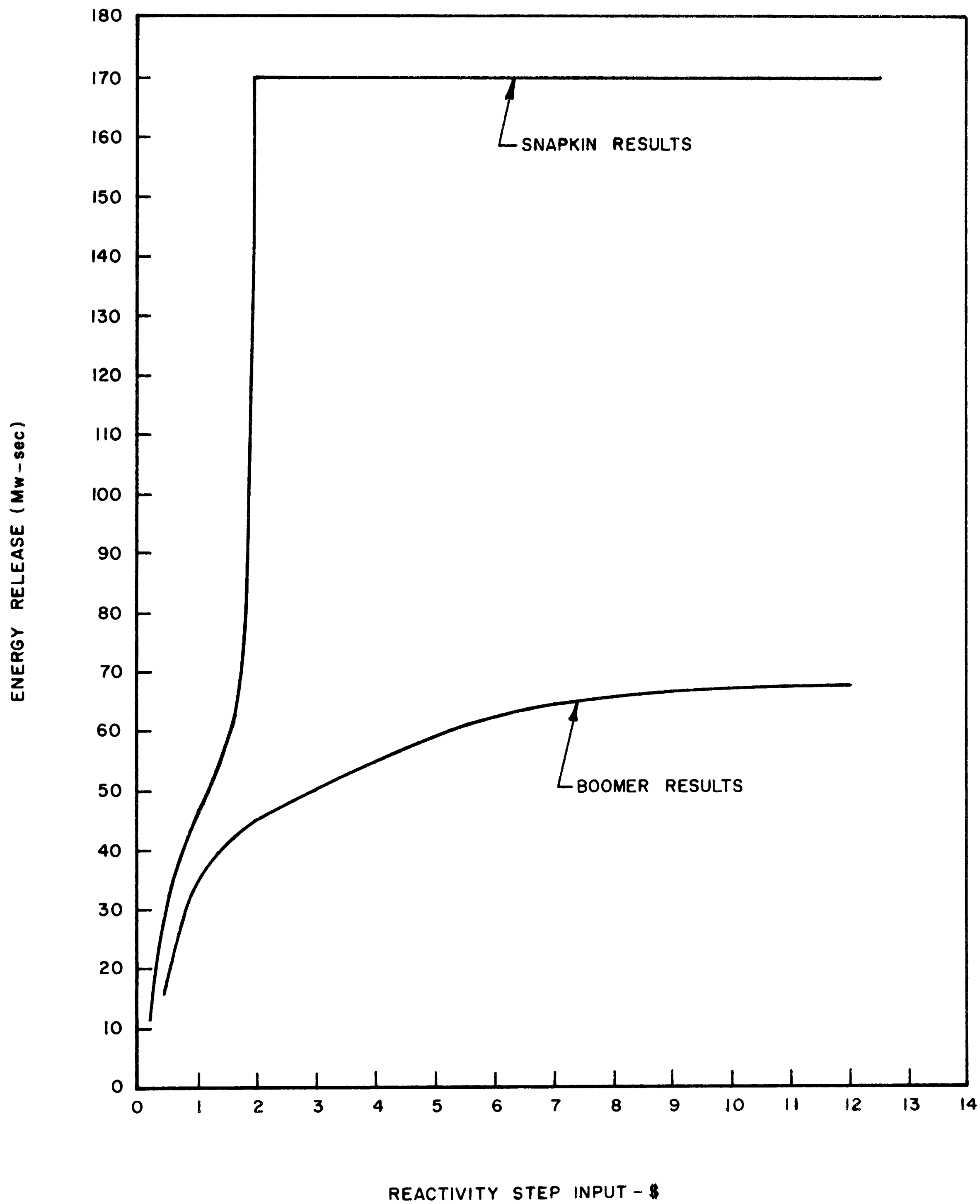


Fig. 45 - Maximum Energy Release as a Function of Step Reactivity Input For Two Calculational Models

modified to include the effect of reactor disassembly by considering the propagation of a sonic pressure wave resulting from hydrogen release. The results are for a core filled with NaK. Each figure includes the time of average rod clad burst, the time core disassembly is effected, and the time of zero reactivity. The calculational results of the BOOMER code show that 55 Mw-sec of energy is generated in the transient resulting from the $5.00 step reactivity input, in contrast with the results of the SNAPKIN calculations, which give the energy release as 170 Mw-sec. Of particular interest is the relative flatness of the curve for inputs of greater than $7.00. For these reactivities, core disassembly is a result of hydrogen loss from the outer ring of fuel rods rather than from the central rods. This makes the energy, at the time of start of core vessel disassembly, independent of the reactor period. From this time to the time the assembly becomes far subcritical is a short interval relative to the reactor period; therefore, little additional energy is released.

The significance of the BOOMER calculations lies in the fact that the addition of a core disassembly shutdown mechanism shows the substantial conservatism of the SNAPKIN calculational model. The use of a disassembly shutdown mechanism results in a rather small fraction of the hydrogen being dissociated prior to core disassembly. As a result, the energy release calculated using this model is therefore considerably less than that calculated using the SNAPKIN code, which assumes complete hydrogen dissociation prior to disassembly.

In addition to the Atomics International analysis using the SNAPKIN and BOOMER codes for axial fuel expansion, hydrogen loss and disassembly shutdown mechanism, the disassembly mechanism mentioned previously was also examined by Dr. M. A. Cook, Director of the Institute of Metals and Explosives Research, University of Utah. This mechanism concerns the explosive disassembly of the reactor core. Dr. Cook's analysis was based primarily on information relating hydrogen release from the fuel-moderator to the energy of the excursion[1], zirconium hydride dissociation pressure to temperature[16], and bursting time for the clad to the fuel (or clad) temperature[17]. The latter data depend on extrapolated hydrogen diffusion rates only, and do not reflect rates of hydrogen dissociation. Dr. Cook used the calculated results of a \$4.00 step reactivity insertion as an example of a rapid excursion.

From these data, it was concluded that the excursion energy must exceed 25 Mw-sec before the hydrogen generation from hydride dissociation becomes appreciable. The dissociation pressure of the fuel-moderator is high enough to burst the cladding, assuming a 1200 psi rupture pressure at 1500°F or above. As will be seen below, however, the energy required to release an amount of hydrogen sufficient to burst the cladding at 1500°F must be greater than 25 Mw-sec, and at higher energies an explosive disassembly could occur.

If a fast excursion of just sufficient energy to generate the burst pressure on the cladding were conducted, the hydrogen diffusion through the fuel-moderator would be too slow to constitute an explosion hazard. In fact, considering only diffusion rates, passage of hydrogen through the fuel-moderator should not occur rapidly enough to burst the cladding in less than one second at any temperature below about 1950°F, even if hydride dissociation were instantaneous. It is not likely that cladding rupture will constitute a limiting explosion condition because, at 1950°F, the internal pressure on the fuel-moderator (dissociation pressure of the hydrogen) could be greater than the strength of this material. On the other hand, if an excursion were run at just the right energy to raise the final temperature to 1500 to 1700°F, the amount of hydrogen generated would be very small, and the cladding would be ruptured in a completely non-explosive process. Such an excursion would be relatively slow (see Figs. 33 and 34), and heat losses would require the energy input to be greater than 25 Mw-sec.

For a 9,000 to 10,000 psi tensile strength of the fuel-moderator, it is probable that an explosion along dislocations (micron scale imperfections) will occur at a pressure of only 1.5×10^4 psi, which is the dissociation pressure at about 1900°F. However, a true (lattice disintegrating) explosion would probably require an effective pressure of 6.5×10^5 psi, which is the dissociation pressure at about 2600°F.

To estimate the energy required to raise the pressure, or energy density, to p_c, the critical explosion pressure, an equation of state is required. The general equation,

$$pV = nRT + \alpha(v,T)p$$

where α is a function of temperature and volume, reduces to the ideal gas law

$$pV = nRT$$

where p is small. Cook[18] suggests using an $\alpha(v)$ equation as a good approximation to the $\alpha(v,T)$ equation, where α is a function of v, the specific volume, only. This is the form used in the analysis. The equation of state may then be written for the critical pressure,

$$p_c(V - \alpha) = nRT, \text{ or}$$

$$p_c V\left(\frac{V - \alpha}{V}\right) = nRT \quad .$$

If the initial pressure is negligible (near atmospheric), the energy required to raise the pressure to p_c is

$$p_c V\left(\frac{V - \alpha}{V}\right) \quad .$$

Using the density of the fuel elements, the term $\left(\frac{V - \alpha}{V}\right)$ can be estimated from Cook's empirical $\alpha(v)$ relationship[18] to be about 1/5. Thus, the energy required to reach p_c is about $p_c V/5$. Taking $p_c = 6 \times 10^5$ psi and V = 8.5 liters (the volume of the fuel elements in the core, or the volume of the system just prior to gas expansion), and adding the energy to raise the temperature of the core to 2600°F, the total energy input required for the lattice disintegrating explosion is about 40 Mw-sec.

Results of SNAPKIN code calculations for temperature, energy, and power as functions of time following a $4.00 step reactivity insertion are indicated in Fig. 36. From the energy curve and the above energy requirement, it is estimated that a lattice disintegrating explosion can occur after about 5.9 msec. The data indicate that, if the power rise is sufficiently fast, there will be neither appreciable hydrogen diffusion nor heat loss before reaching 2600°F. Since the explosion can occur after an energy input of about 40 Mw-sec in a rapid transient, the time before such an explosion occurs can be estimated for other large reactivity insertions from energy-time curves by the method of Cook. In a similar manner, the energy required for an explosion along a dislocation can be estimated to be about 25 Mw-sec, and such an explosion can occur after about 5.7 msec in a $4.00 step insertion.

The energy required for an explosion can also be estimated directly from the temperature and energy curves, for comparison with the $p_c V/5$ estimate of Cook, assuming 1.5×10^4 psi is required for an explosion along a dislocation, and 6×10^5 psi for a true explosion. For example, in a $4.00 insertion (Fig. 36) 1.5×10^4 psi (a dissociation pressure equivalent to 1900°F) requires about 45 Mw-sec energy, and 6×10^5 psi (2600°F) requires about 70 Mw-sec. In the same way, a $5.00 step insertion (Fig. 37) gives about 38 Mw-sec for 1.5×10^4 psi, and about 62 Mw-sec for 6×10^5 psi. A $2.50 insertion (Fig. 35) gives 38 and 60 Mw-sec, while a $1.00 step does not reach required temperatures for explosions. However, the latter reactivity insertion would provide conditions similar to those required to burst the cladding.

Since the shutdown mechanisms discussed above are not completely understood, the safety analysis of the proposed tests has employed the most conservative aspects of each of these mechanisms.

It is estimated that the shortest self-limiting reactor period which the SNAPTRAN 2/10A-1 core can undergo without damaging the fuel rods is about five seconds. The experimental program includes a reactor destructive test that will require a reactor period of about 0.25 msec. During this test, complete destruction of the fuel rods is expected. In the region between no damage and complete destruction, additional tests will be performed. These tests, which are expected to involve some damage to the core, have been termed "partial damage tests".

However, it is also possible that partial damage might occur during longer period tests. The hazards involved to operating personnel for those tests in which the power excursion is not violent enough to disperse fission products over a wide area are discussed below. The hazards involved for tests in which fission product release can be expected are discussed in Destructive Test Hazards, Section V-D.

4. Radiological Hazards

The radiological hazards associated with the normal operation of an unshielded reactor are from direct radiation and possible fission product release. The direct radiation during operation consists of gamma rays associated with the fission process (fission gammas) and very short lived fission product decay, neutron leakage from the core, and neutrons produced by the γ,n reaction in the beryllium reflector surrounding the reactor. After reactor shutdown, the neutron flux from the core will be negligible and the hazards will then consist of gamma rays from fission product decay, gamma rays from induced activity in the structural materials, and neutrons from the γ,n reaction in beryllium.

The following is the evaluation of each radiation hazard.

a. Direct Radiation

(1) During Reactor Operation. The radiological hazard associated with the normal operation of the unshielded SNAPTRAN 2/10A-1 reactor, assuming fission products are not being released to the atmosphere, is due to the direct radiation from the reactor.

The calculations of radiation levels were based on the following assumptions:

(1) reactor located in the IET Test Cell building,

(2) reactor operating at 100 watts (dose rates are proportional to power level),

(3) radiation shield materials between the reactor and the dose point are 3 in. of beryllium, 0.032 in. of stainless steel, and air, and

(4) average thermal neutron flux in the reactor of 6.23×10^8 n/cm^2-sec.

The biological dose rates from fast neutrons, prompt fission gammas, and short-lived fission product gammas as a function of distance from the reactor are presented in Figs. 46 and 47. The total dose rate, neutrons plus gammas, is presented in Fig. 48. The dose rates from capture gammas in the reactor and reflector materials, and neutrons from the Be (γ,n) reaction were determined to be less than 3% of the total dose rate. These sources are assumed to be negligible.

The predicted dose rates for various areas are shown in Table V-2.

TABLE V-2

DOSE RATES DURING CONSTANT POWER OPERATION

Location	Distance from Reactor (ft)	Dose Rate @ 100 w (mrem/hr)
Fenced area around TSF	> 4800	< 0.01
LCRE	> 7000	< 0.01
End of tunnel at IET	700	2.0
Security fence around IET facility	170	100
IET Control and Equipment Building (Control Room)*	40	0.01

* Shielded from reactor with earth and concrete

It is concluded from the calculations that personnel could have unlimited access to areas further than 800 ft from the reactor. However, the area enclosed by the obstruction fence will be an exclusion area and no admittance will be permitted during reactor operation. The obstruction fence, at its closest point, is approximately 5000 ft from the Test Cell.

To determine the gamma dose rate as a function of distance from the reactor, source geometry was assumed to be a cylinder for dose points

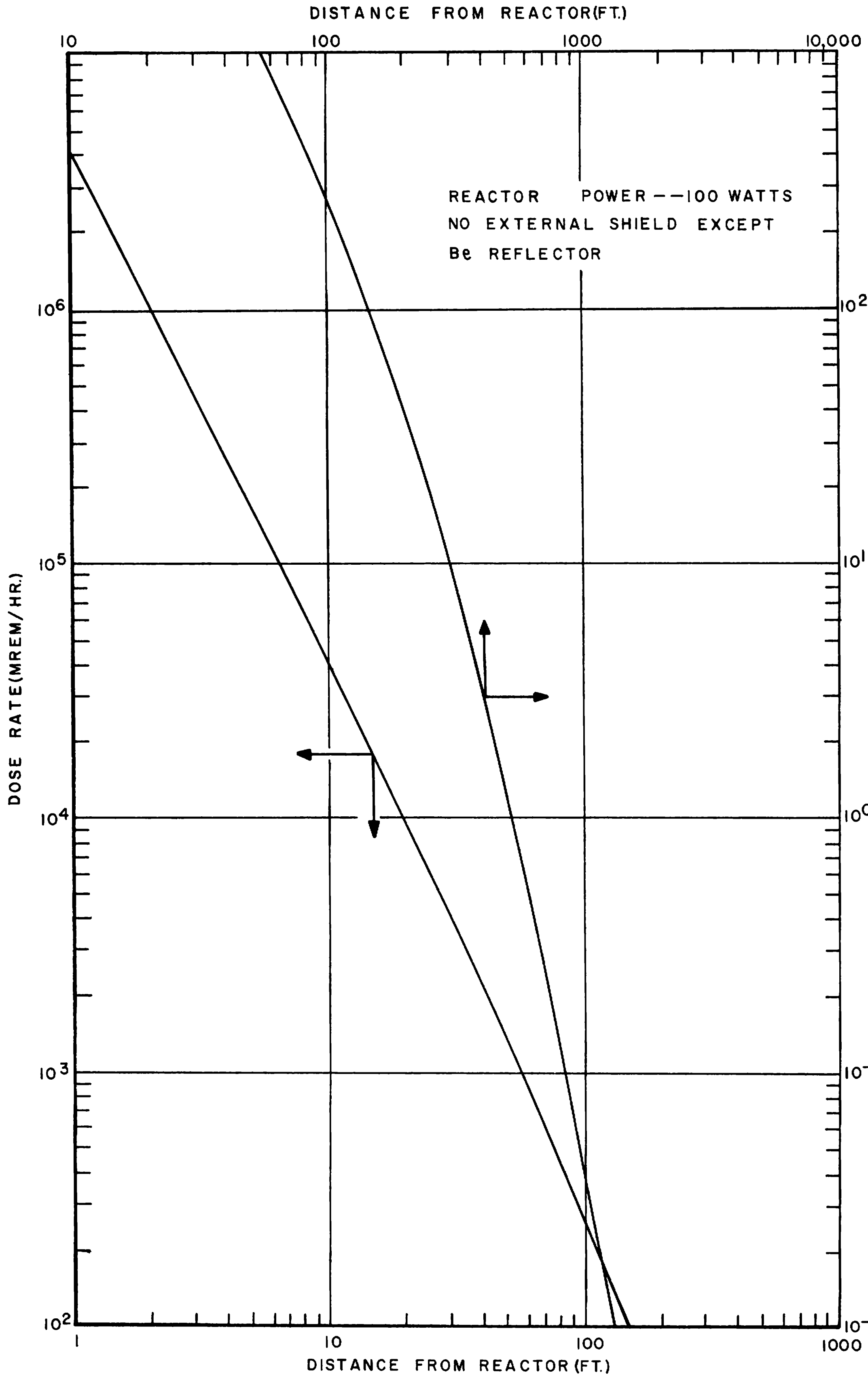


Fig. 46 - Fast Neutron Dose Rate From Operating Reactor

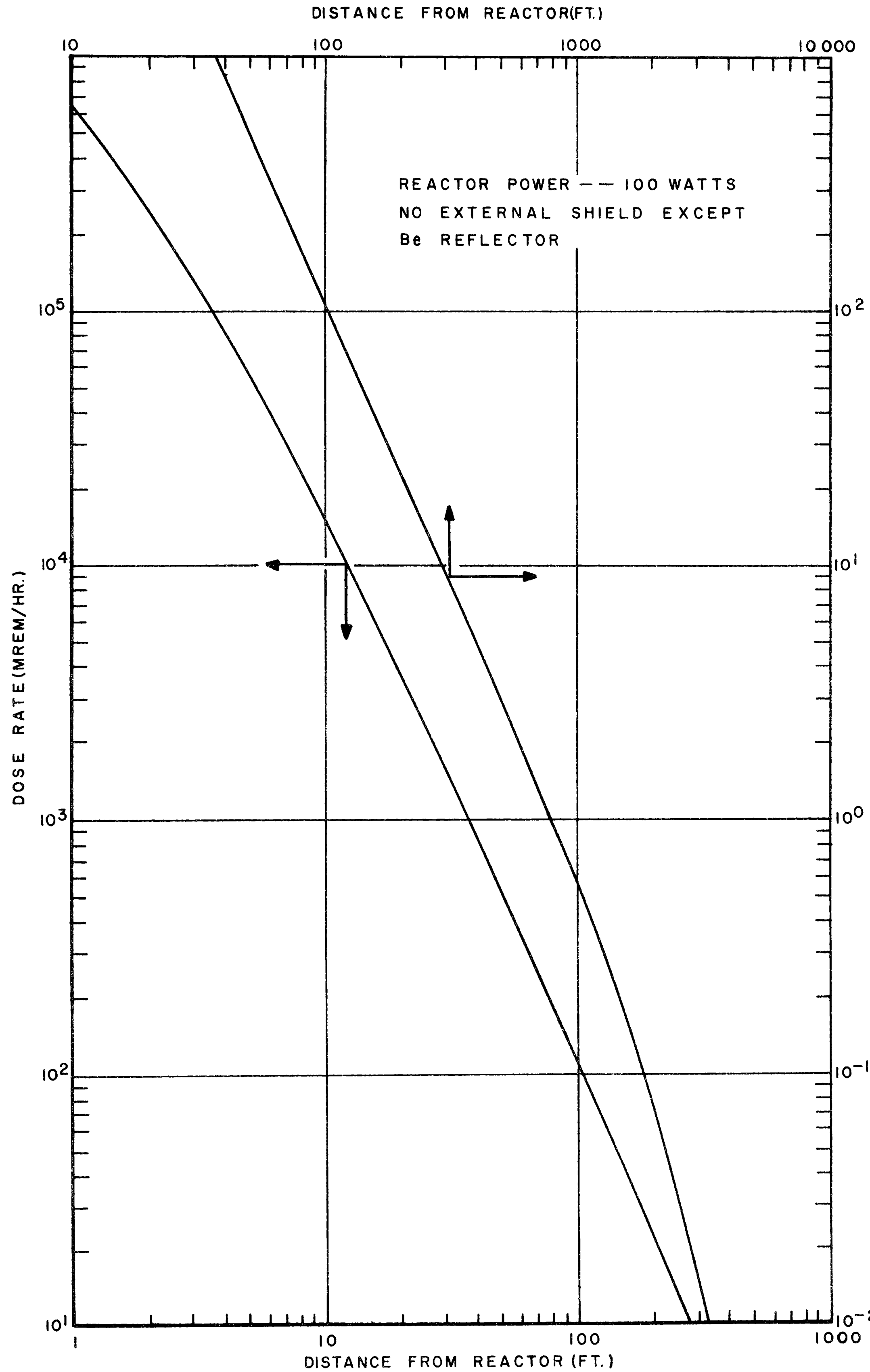


Fig. 47 - Gamma Dose Rate From Operating Reactor

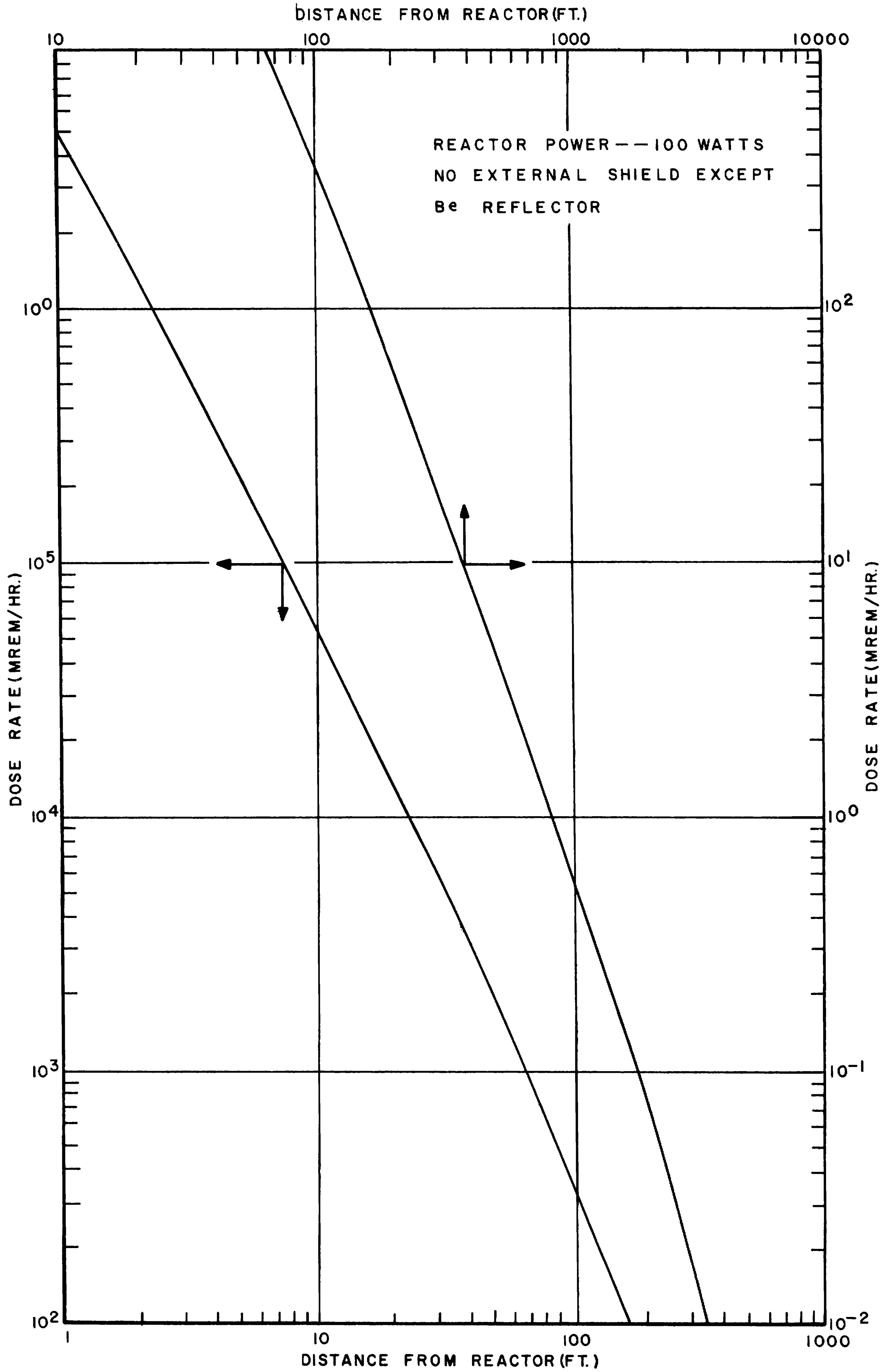


Fig. 48 - Total Dose Rate From Operating Reactor

up to a distance of 10 ft and a point source for distances greater than 10 ft. The techniques used are those presented in TID-7004[19].

Cylindrical source

$$\phi = \frac{B\, S_v R_o^2}{4(a + z)} F(\theta, b_2)$$

where:

ϕ = gamma flux (γ's/cm^2-sec)

B = buildup factor (dose buildup for water, point source)

S_v = gamma source density (γ's/cm^3-sec) of energy E

a = source to dose point distance (cm)

z = effective self-attenuation distance of the source (cm)

R_o = radius of cylinder (cm)

θ = $\tan^{-1}\left(\frac{h/2}{a+z}\right)$ where h is the length of the cylinder

b_2 = $\sum_i^n \mu_i\, t_i + \mu_s z$

μ_i = macro-cross section (linear absorption) of the ith shield

t_i = thickness of the ith shield (cm)

μ_s = macro-cross section (energy absorption) of the source material

$$F(\theta, b_2) = \int_o^{\theta} e^{-b_2 \sec\theta}\, d\,\theta$$

Point source

$$\phi = B \frac{S_o}{4\pi\, a^2} e^{-b_1}$$

where:

S_o = $(S_v)(V)$ γ's/sec

V = volume of cylinder (cm^3)

b_1 = $\sum_1^n \mu_i t_i$

Note: The point source was assumed to be inside the cylinder with an effective external shield equivalent to the self-shielding of the cylinder (μ_s,z) plus the shields external to the cylinder. An energy absorption cross section was used for the source material; thus, the buildup factor was determined only for the shields external to the cylinder. This is a conservative assumption.

The source geometry for the fast neutron dose points within 10 ft of the source was assumed to be a disc with a radius equal to half the length of the core. A point source was used for distances greater than 10 feet.

Disc source

$$\phi = \frac{B\, S_a}{2} \left[E_1(b_1) - E_1(b_1 \sec\theta) \right]$$

where:

$$S_a = \frac{S_o}{A_s} \ (\gamma\text{'s/cm}^2\text{-sec})$$

S_o = total neutron leakage from the core (neutrons/sec)

A_s = surface area of reactor (cm^2)

$$b_1 = \sum_i^n \mu_i t_i$$

$$\theta = \tan^{-1} \frac{R}{a}$$

R = radius of disc (cm)

$$E_1(b) = \int_b^\infty \frac{e^{-t}}{t}\, dt$$

Note: The conversion from neutron flux and gamma flux to dose was obtained from TID-7004 [19]. The neutron energy was assumed to be 1 mev.

(2) <u>After Shutdown from Normal Operation</u>. After the reactor has been shut down from an extended operating period or power impulse, assuming no fission product release to the atmosphere, the

major radiological hazard is from the fission product energy release in the form of gamma rays. In addition, gamma rays are also present from the decay of radioactive nuclei produced by neutron capture in the core materials and the material external to the reactor. In the case of the SNAPTRAN 2/10A-1 reactor the zirconium in the core and the stainless steel and beryllium external to the core are the primary contributors to the decay gamma source. It has been determined that the dose rate from capture gammas is very small; therefore, it is not included in the results.

The dose rate from fission product decay following a power impulse and an extended operating period is presented in Fig. 49. The predicted dose rate following the power impulse is based on a single 30 Mw-sec power excursion. The dose rate at times greater than 24 hours after the impulse will be approximately proportional to the number of excursions over a relatively short test period. The dose rate following an extended operating period is based on the assumption that the reactor had operated for 10 hours at 100 watts. Both power level assumptions are assumed to be conservative.

Curves of the fission product buildup and decay for the extended operating period are presented in Figs. 50 and 51. The decay rate following a power impulse is faster than for an extended operating period for an equivalent energy release.

Personnel access to the test cell and the immediate area surrounding the building will be prohibited until the area has been surveyed by the Health Physicist. Time limits for working in this area will be established after the radiation surveys have been evaluated. It is predicted that access to the test cell building after a 30 Mw-sec power impulse will be prohibited for at least 24 hours.

No time limits for access to the area outside the security fence are expected except when it is determined that fission products have been released to the atmosphere. In any case, the area will be surveyed and the necessary precautionary measures will be taken to insure the safety of personnel entering the area.

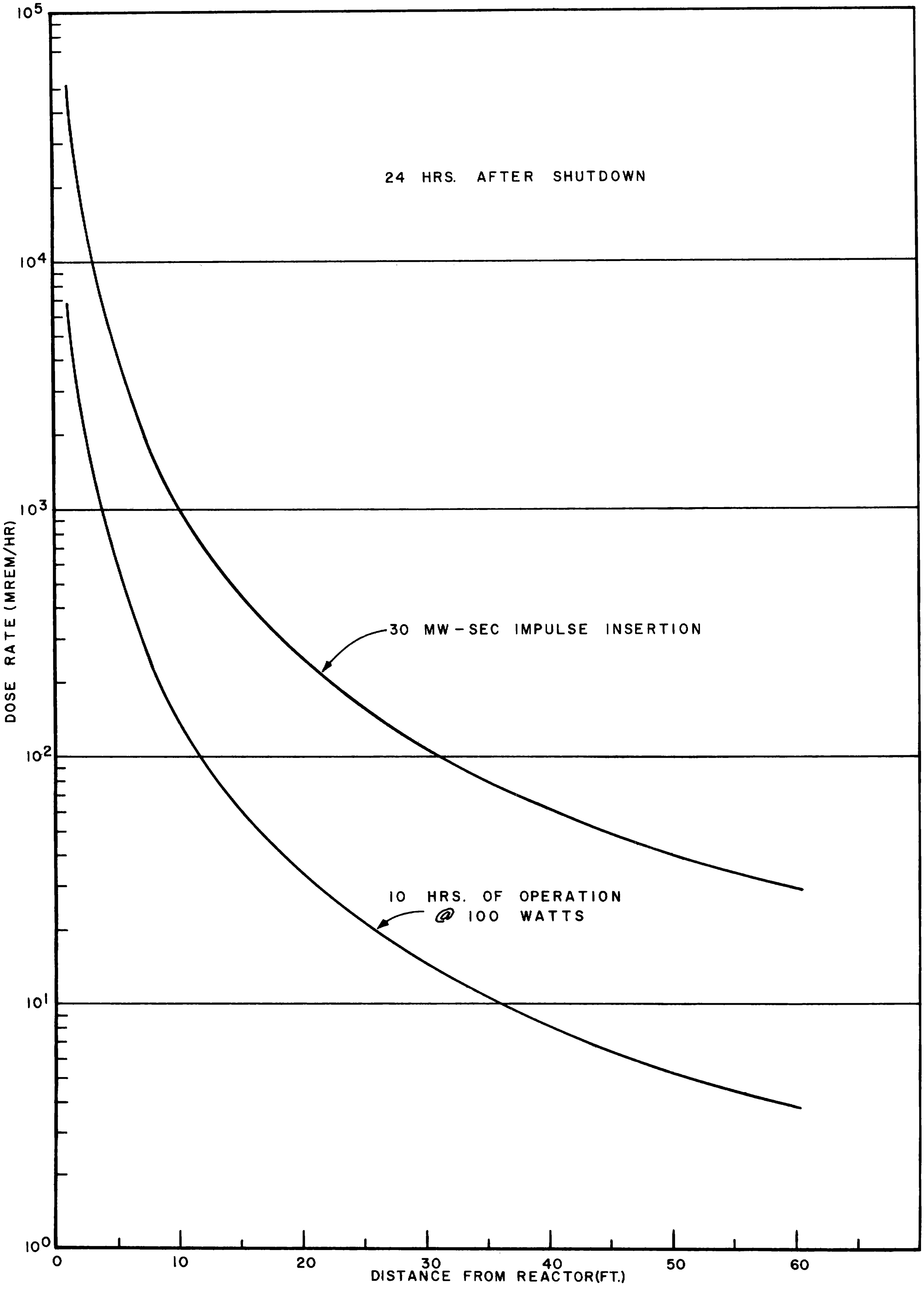


Fig. 49 - Gamma Dose Rate From Fission Product Decay

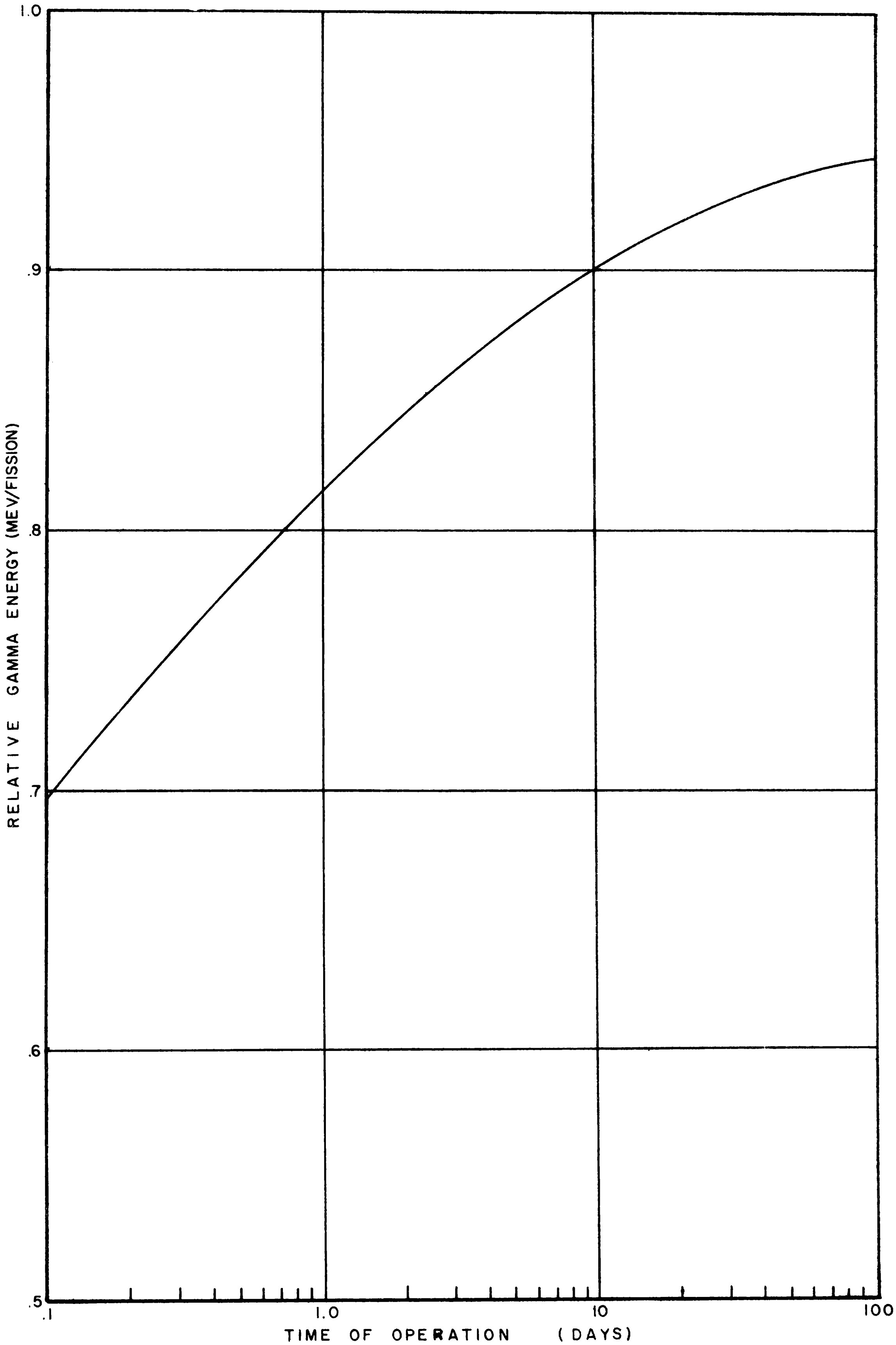


Fig. 50 - Gamma Energy From Fission Products for Various Operating Times

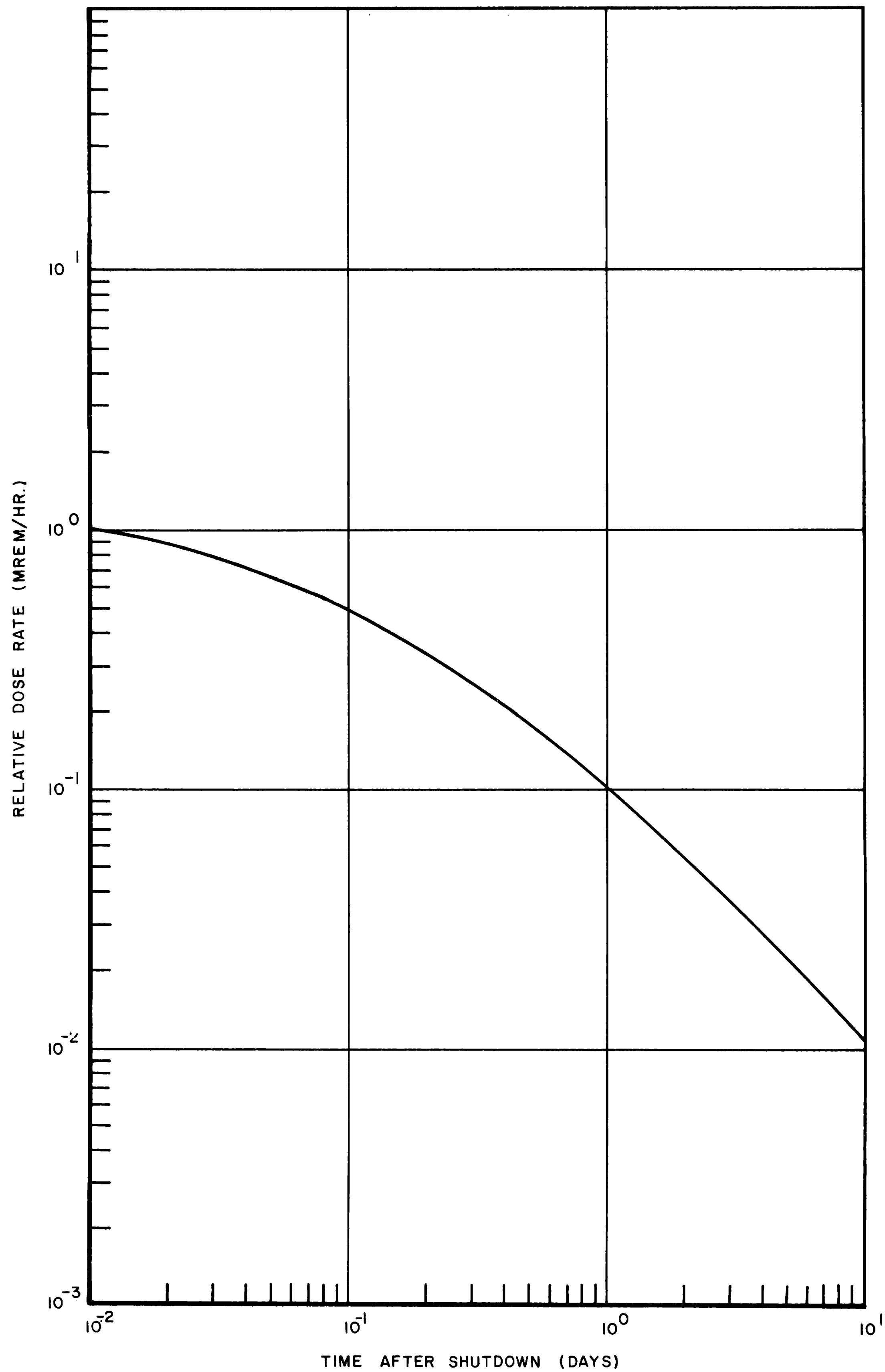


Fig. 51 - Fission Product Decay Following Ten Hours of Operation

In addition to the gamma radiation after shutdown, fast neutrons resulting from the Be (γ,n) reaction are also present. However, the dose rate from this source is very small in comparison with the fission product gammas and is not considered to increase the shutdown radiation hazard.

The fission product energy release as a function of operating and shutdown times for an extended operating period was determined by the techniques presented in WAPD-P-652[20].

$$E_\gamma(T_o,T_s) = K \int_{T_s}^{T_o+T_s} 1.6\ T^{-1.2} dT$$

where:

$E_\gamma(T_o,T_s)$ = gamma energy release as a function of operating time T_o, and shutdown time, T_s

$K = 3.1 \times 10^{10}\ \frac{P}{V}$ fissions/cm^3-sec during operation

P = reactor power in watts

V = volume of reactor (cm^3)

The fission product energy release from the impulse insertion was determined by the technique presented in WAPD-R(F)-38[21].

$$\text{Total power} = 3.1 \times 10^{10}\ P\ t_o\ G(\infty,T_s)/T_s \quad \frac{\text{Mev}}{\text{sec}}$$

where:

$$G(\infty,T_s) = A\ T_s^{-a}$$

and for

$$1.5 \times 10^2 \leq T \leq 10^6$$

$$A = 14.77$$

$$a = 0.2919$$

The fission product gamma energy spectrum as a function of time after shutdown was obtained from the curves in WAPD-R(F)-38[21].

The source geometry considerations were the same as those presented for the operation conditions.

b. Fission Product Release

The ability of the fuel element cladding to completely contain fission products under design conditions of temperature (NaK coolant at 1200°F) and power (50 kw) was demonstrated by the SER during a 1000 hour endurance run[22] and by the S2DR during over 500 hours of operation at these conditions. At no time was fission product activity detected in the NaK coolant of these reactors.

All tests in which fission product release can reasonably be expected will be conducted under meteorological control, as defined in Section V-D. However, an unplanned release could conceivably occur during long period transients ($< \$0.50$ step insertion) when fuel element damage is not expected. The consequences of an accidental release have been investigated using a nuclear energy release of 30 Mw-sec, which is an overestimate.

Since the maximum temperature ($\sim$ 1700°F) attained in the core would be considerably below the melting point of the fuel, the fission product release to the atmosphere should be quite small. To investigate the extreme consequences of the postulated accident, however, it was assumed that 100% of the noble gases and halogens escape to the atmosphere under severe inversion conditions ($n = 0.5$) with no effective stack height. Further, it was assumed that the released products are borne by a 2.5 mph wind directly to the area in question with no intermediate fallout.

Table V-3 presents the maximum exposure (TID) to personnel downwind of IET for all wind directions under such an accumulation of parameters. The calculations show that the radiological dosages received, regardless of wind direction, are below the permissible limits as recommended by the Federal Radiation Council.

Hazards resulting from fission product release accompanying the destructive test are discussed in Section V-D.

TABLE V-3

TOTAL INTEGRATED EXTERNAL AND THYROID DOSAGE

Town or Facility	Distance from IET (miles)	Cloud Dose (rem)	Thyroid Dose (rem)	TID (rem)
TSF Area	1.3	1.10	.960	2.06
Monteview	12	3.5×10^{-3}	3.7×10^{-2}	4.05×10^{-2}
NRF Area	22	5.8×10^{-4}	1.0×10^{-2}	$\sim 1.0 \times 10^{-2}$
Small	30	2.6×10^{-4}	5.4×10^{-3}	5.66×10^{-3}
Idaho Falls	39	1.2×10^{-4}	3.4×10^{-3}	3.52×10^{-3}

5. Component or Coolant Hazards

Some of the non-fissionable materials used in the SNAPTRAN 2/10A-1 core represent health hazards to operations personnel--either because they are toxic or because they can react violently with air or water. The hazards involved in reactor operations due to the unclad beryllium reflector, the NaK coolant, and the ZrH fuel alloy have been considered.

a. Beryllium Toxicity

The SNAPTRAN 2/10A-1 reactor will be surrounded by an unclad beryllium reflector. The presence of this beryllium presents several health hazards resulting from the toxicity of the metal, the oxide and vapors of beryllium[23].

The hazard associated with beryllium is beryllium poisoning, the most serious aspect of which is due to inhalation of beryllium or its compounds. Beryllium poisoning may manifest itself in one of three ways: (1) acute berylliosis, (2) chronic berylliosis, and (3) dermatitis.

During testing, the reactor and reflector will be enclosed in a thermal insulating box. A continuous nitrogen purge will pass through the box and to the atmosphere through the IET stack. Before being exhausted to the atmosphere, the nitrogen will be monitored to insure that the beryllium concentration in the stack effluent does not exceed tolerance limits of 0.01 $\mu g/m^3$ of air on the average monthly concentration.

The test cell atmosphere will also be monitored to detect any buildup of beryllium compounds in the building due to system leaks.

If at any time it becomes necessary to work with the beryllium, all work will be carried on by remote methods if possible. If this becomes impossible, and if direct handling of the beryllium cannot be avoided, the following procedure will be followed[24,25]:

(1) A "beryllium zone" will be established. All manual operations with the beryllium will take place within this area.

(2) Personnel working within the beryllium zone will be familiar with the hazards involved in working with beryllium and its compounds.

(3) No one will enter the beryllium zone without prior Health Physics clearance or work there for any extended time unless subjected to periodic medical examinations.

(4) No one with a history of pulmonary or skin disorders will be allowed to work in the beryllium zone.

(5) Personnel will be provided with a complete change of clothing upon entering the beryllium zone and will shower upon leaving the zone.

(6) The air in the beryllium zone will be adequately ventilated and will be continuously monitored to insure the following AEC conditions are met:

   (a) Stack effluent will not exceed 0.01 $\mu g/m^3$.

   (b) Concentrations of beryllium will not exceed 2 $\mu g/m^3$ averaged over an 8-hour day.

   (c) Beryllium concentration should not exceed 25 $\mu g/m^3$ for any period of time, however short.

(7) One self-contained breathing apparatus per man will be located in the immediate work area in case of emergency.

Although beryllium is a highly toxic material and is capable of producing a variety of diseases in humans, it can be handled safety if proper precautions are used. These precautions will be adhered to while working on or around the reactor and all personnel will be familiar with them.

b. Chemical Reactions

(1) Metal-Water Reactions. The only metal-water reaction that could possibly occur is a reaction between Nak and water during the time the core vessel is being charged with NaK in the examination area prior to the destructive test and during "cleanup" following the destructive test. During the time the NaK (approximately two pounds) is loaded and sealed in the core, no water will be allowed in the area. Also, the NaK storage prior to loading and any excess NaK following loading will be in an area having no water. As a result of these precautions, a hazard due to a NaK-water reaction is not considered credible.

During "cleanup" following the destructive test, a NaK-water reaction is a definite possibility. However, it is expected that the majority of the NaK will react with air during destructive disassembly of the core during the excursion. Any unreacted NaK remaining will present no explosive hazard during "cleanup" due to its fine dispersion and small amount.

(2) Metal-Air Reactions. NaK and zirconium present a possible source of metal-air reactions following the partial damage tests.

The greatest danger of a NaK-air reaction is during NaK charging for the destructive test and could result from an equipment failure causing a liquid-metal leak or from an accidental spill.

If either of these conditions occur, resulting in a NaK fire, Met-L-X extinguishers will be available. An S-350-A (300 lb capacity) Met-L-X extinguisher will be mounted on the test dolly and small portable units will be located in both the loading and storage areas.

The possibility of a zirconium-air reaction and the magnitude of such a reaction is discussed in Section V-D.

(3) Hydrogen Reactions. As the fuel material in the SNAP 10A reactor contains about 500 gram moles of hydrogen as the moderator, the possibility of hydrogen-air reactions must be considered. Two possibilities exist for a chemical reaction within an intimate mixture of hydrogen and air; the mixture may detonate or it may burn. Between the limits of 4.1 volume percent and 74 volume percent hydrogen, the mixture is flammable. If the mixture composition is between 19 and 57 volume percent hydrogen, detonation can occur.

Hydrogen escape from the fuel is effectively prevented by an inner ceramic coating which acts as a barrier to hydrogen diffusion through the cladding. Therefore, the contained hydrogen in the fuel rods is not expected to constitute a hazard during normal operations.

The case of the fuel rod failure is discussed under Destructive Test Hazards.

6. Hazards of Partial Damage Tests

As the STEP program progresses with the testing of the SNAPTRAN 2/10A-1 reactor, some limited damage tests will be performed. It is also quite possible that a planned nondestructive test will result in partial disassembly of the fuel rods or of both the core and the reactor vessel.

a. Fuel Rod Damage

The partial damage tests represent a potential hazard because fission products may be released to the atmosphere. Fission product monitors will be employed in the gas flow system which is used to control the temperature of the reactor. The Health Physicist will remotely monitor the air in the test cell to ascertain whether or not radioactive gases are present. This safety procedure will precede the admittance of personnel into the test cell after all tests.

b. Reactor Vessel Damage

Reactor vessel damage is considered unlikely during nondestructive nuclear testing. However, during partial damage tests it is feasible that failure of some reactor components could result in a power excursion of such magnitude that the reactor vessel would be damaged. If the critical temperature (1500°F) for the fuel rods is exceeded by a large margin it is expected that there will be disassembly of the reactor vessel as well as the fuel rods. An excursion of this magnitude is considered in the next section. If for some reason a planned nondestructive test results in partial or total disassembly of the reactor and its instrumentation, re-entry to the test cell will follow a standard re-entry procedure as described in Section VI-5. Precautions observed during re-entry will include safeguards against the initiation of a secondary criticality.

D. Destructive Test Hazards

A destructive test will be performed to model the maximum credible incident involving the SNAP 2/10A reactor. A step insertion of $4 into the system will cause the reactor power to rise on about a 0.25 msec period. Violent disassembly of the reactor could occur, expelling fuel debris and released hydrogen to the atmosphere. As the exact shutdown mechanism is unknown, several assumptions were made as to the consequences of this excursion:

(1) On the basis of the shutdown mechanism analysis previously discussed it is assumed the total nuclear energy release will not exceed 170 Mw-sec.

(2) Fission product buildup prior to the destructive test is negligible; therefore the total fission product inventory will be generated by the 170 Mw-sec excursion.

(3) Fracturing of the zirconium-hydride due to internal hydrogen pressure. This fracturing could conceivably reduce the zirconium-hydride to a grain size of 10 to 50 microns. This assumption leads to a higher radiological dose, due to the fallout velocity and hence higher deposition dose, than would be obtained from the assumption of fuel melting and subsequent release to the atmosphere as a vapor.

(4) Failure of all the fuel rods in the core.

(5) Radial propagation of the burst within the core.

(6) Expulsion of the fuel material from the core at a temperature of 3200°F (the fuel melting temperature).

(7) From the data of Lee and Brick[27], which give the compressive or rupture strength of beryllium crystals as $\sim$ 200,000 lb/in^2, it is estimated that relatively little shattering of the beryllium will occur since the pressures generated in the core will be considerably below the rupture strength. However, it was assumed that 1% (40.40 g) of the beryllium would fragment to grain size and, therefore, be available for release to the atmosphere.

The amount of O_2 required to burn all of the combustible material in the SNAPTRAN 2/10A-1 core is 764 mols O_2 or 3820 mols air. The air required would be contained in 9.4×10^4 liters at atmospheric temperature and pressure. If the explosion were to engulf all the air within about 10 ft in a small enough time to contribute all of the heat of reaction (1.2×10^5 kcal) for complete combustion of the fuel elements, the total explosion would involve about 1.3 to 1.4×10^5 kcal for a TNT equivalent of about 300 lbs.

Experimental evidence has shown zirconium-hydride in the small grain size assumed will react violently with air[28]; however, the extent of rate of the resultant zirconium-hydride-air and hydrogen-air reactions can not be predicted accurately. For the purpose of this analysis, the assumption was made that all the zirconium and hydrogen in the core will react with air and, therefore, are released to the atmosphere as fine particulate and vapor.

Since the test cell building will be removed prior to the destructive test, the possible generation of high pressures in the vicinity of the core is not expected to result in significant physical damage to the facility.

On the basis of the foregoing considerations, the hazards to personnel within the immediate area and in those areas lying downwind from the test site have been evaluated.

1. <u>On-Site Hazards</u>

a. <u>Radiological Hazards</u>

(1) <u>Direct Radiation</u>. The integrated dose as a function of distance from the reactor during the destructive test power excursion is presented in Fig. 52. This analysis is based on the assumption that the total nuclear energy release is 170 Mw-sec and that the prior fission product inventory is negligible. The dose presented is the total dose from fission neutrons, prompt fission gammas and short lived fission product gammas.

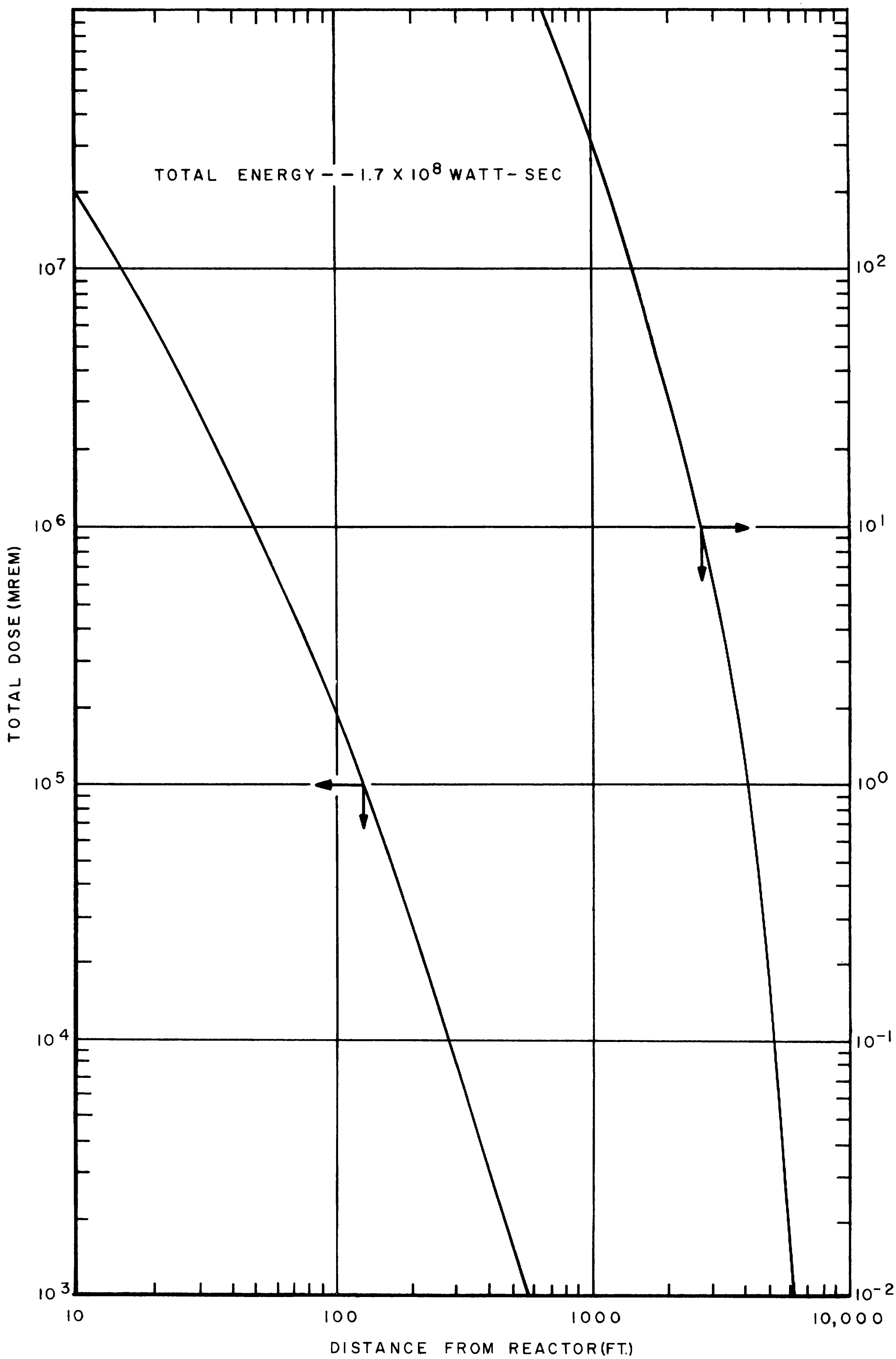


Fig. 52 - Total Dose From Direct Radiation For 170 Mw-sec Power Excursion

From the analysis, it is apparent that the integrated dose is negligible at distances greater than 4000 ft from the reactor. Since the closest unshielded area normally occupied by personnel (fenced area around the TSF area) is farther than 4000 ft, it is concluded that the total integrated dose to personnel in this area during the transient will be negligible (approximately 1 mrem). Access to the area within the obstruction fence, except in the shielded control and equipment building, will be prohibited. The maximum dose in the control room will be approximately 2.2 mrem at the point nearest the reactor. The analytical techniques used for this analysis are the same as those presented in Section V-C for determining the dose rate during normal operation.

(2) Radiological Hazards. In order to reduce the radiological hazards to personnel in the TAN area and to the off-site population, strict meteorological control will be exercised during any tests that may reasonably lead to loss of significant radioactive products to the atmosphere. The meteorological controls which have been agreed upon by Phillips Petroleum Company and AEC-ID are listed below:

(1) wind direction variance of 140° to 210°,

(2) minimum wind speed of 10 mph,

(3) lapse conditions, $n = 0.20$, and

(4) no precipitation.

These parameters shall be forecast to last for at least three hours following any test which may be destructive in nature. All forecasts will be provided by the U. S. Weather Bureau at the NRTS through ID Health and Safety Division. Because of the existing grid layout, which will facilitate area monitoring, it will be necessary that a mean wind direction of 210° ± 30° exists at the time of the destructive test.

Under these conditions, the passage of a cloud containing radioactive gases and debris will be over an uninhabited area for a distance of approximately 6.5 miles, as seen in Fig. 20. The gamma and beta dosage from the radioactive cloud formed after the destructive test will therefore constitute no great hazard to on-site personnel. However, the gamma dose rate from fuel material deposition will be significant for a period of time following the test. It will therefore be necessary to restrict access to the site area lying downwind of the test site. Figs. 53 and 54 give the deposition gamma dose rate for one hour, one day, and seven days following the excursion, assuming lapse conditions and a wind speed of 7.5 mph prevail at the time of the destructive test. The 7.5 mph wind speed was selected to obtain conservative dose rates. The gamma dose rate is computed assuming all the fission products are released to the atmosphere and that 86% of these (representing the non-gaseous products) are deposited. Within one day after the excursion, the deposition dose rate will have dropped to a maximum of ~ 2.85 mrem/hr, which is low enough to permit access to the downwind area for decontamination operations.

Even though the test will be conducted under strict meteorological control, the effect of a wind direction shift that would cause the cloud to pass over the TSF area has been investigated. This area is ~ 1.3 miles from the test site. The maximum dose received by any individual in this area would occur if he were to remain outside of any shelter during the passage of the cloud and to remain within the TSF area for a period of 30 minutes before being evacuated. This represents a conservative estimate, as previous evacuation times have been approximately 10 minutes. Under these conditions, the total integrated dose (TID) from the cloud would be 155 mrem, as shown in Figs. 55 and 56, including 35 mrem to the thyroid. The added exposure from the deposited fission products would amount to 46 mrem, as shown in Fig. 57, giving a total dosage of 236 mrem. This value is well below the 13-week exposure of 3 rem, the recommended maximum of the Federal Radiation Council[29].

Since the TSF area is the nearest facility to IET, the radiation exposure to personnel located at other facilities in the TAN area will be less than the above mentioned dosages.

In the event of some unforeseen circumstance requiring emergency procedures, such as a wind direction shift, the Phillips Petroleum Company Emergency Action Plan (Section VI-B) will be placed into effect.

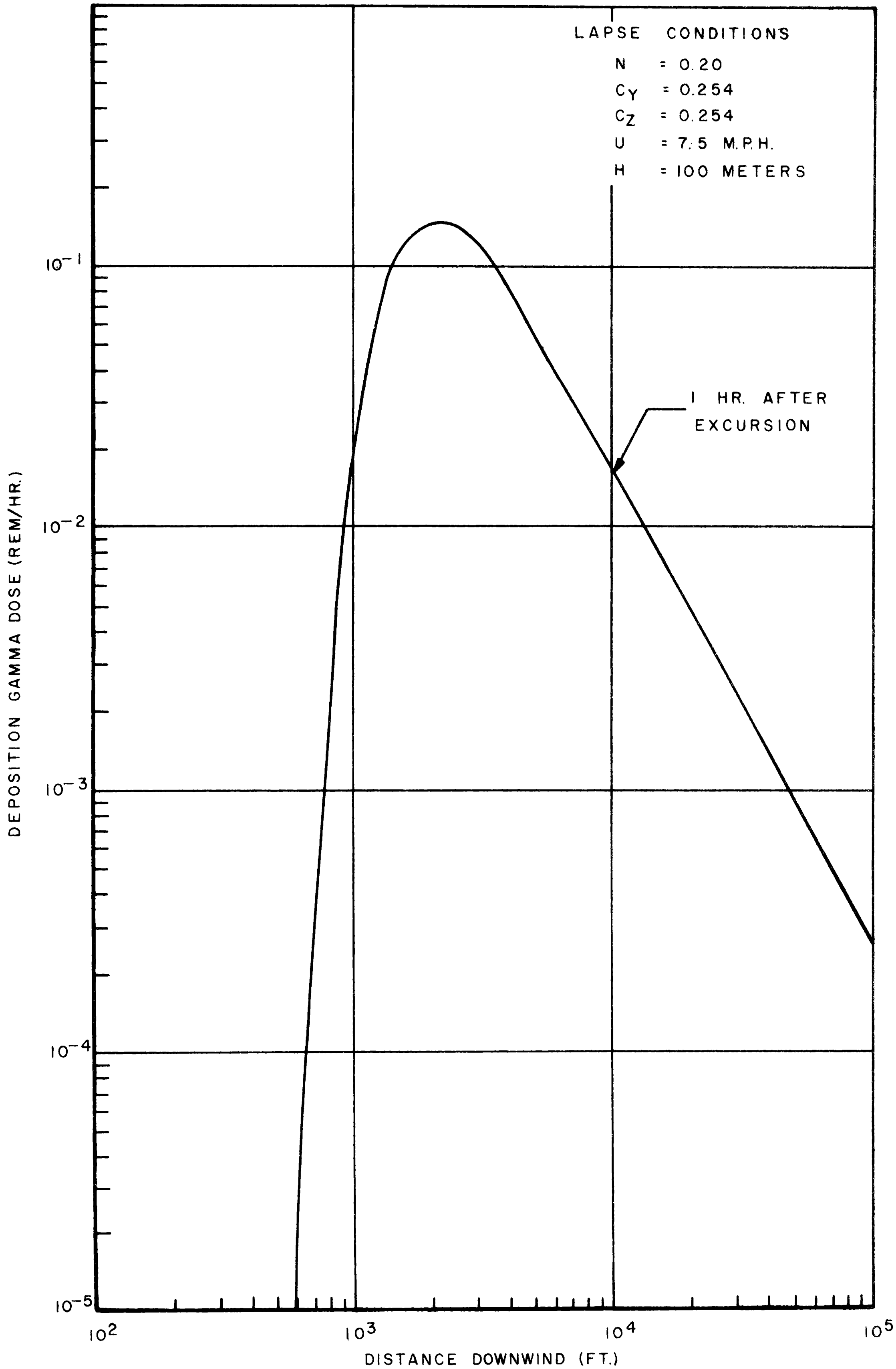


Fig. 53 - Deposition Gamma Dose One Hour After 170 Mw-sec Power Excursion

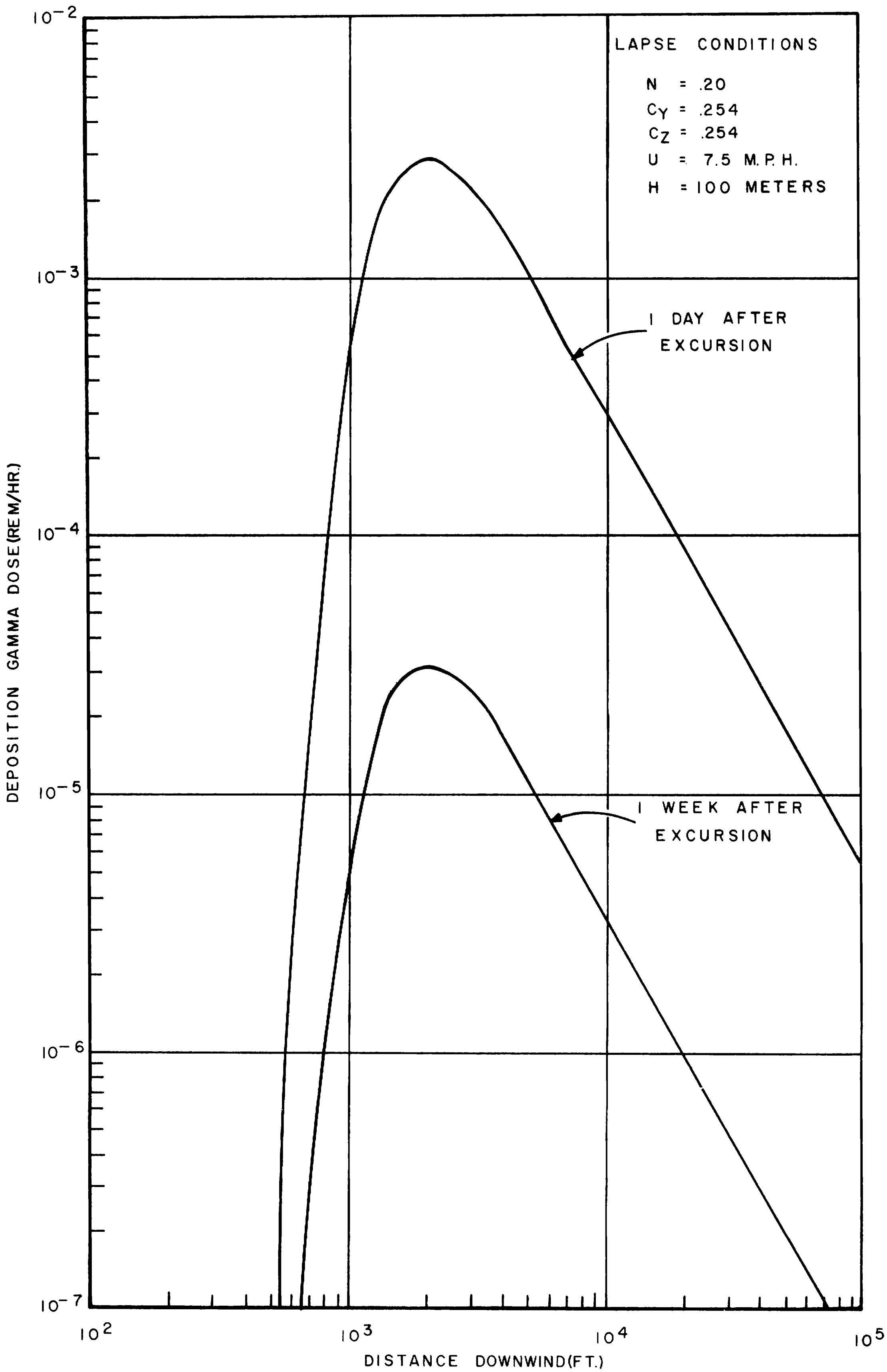


Fig. 54 - Deposition Gamma Dose One Day and One Week After 170 Mw-sec Power Excursion

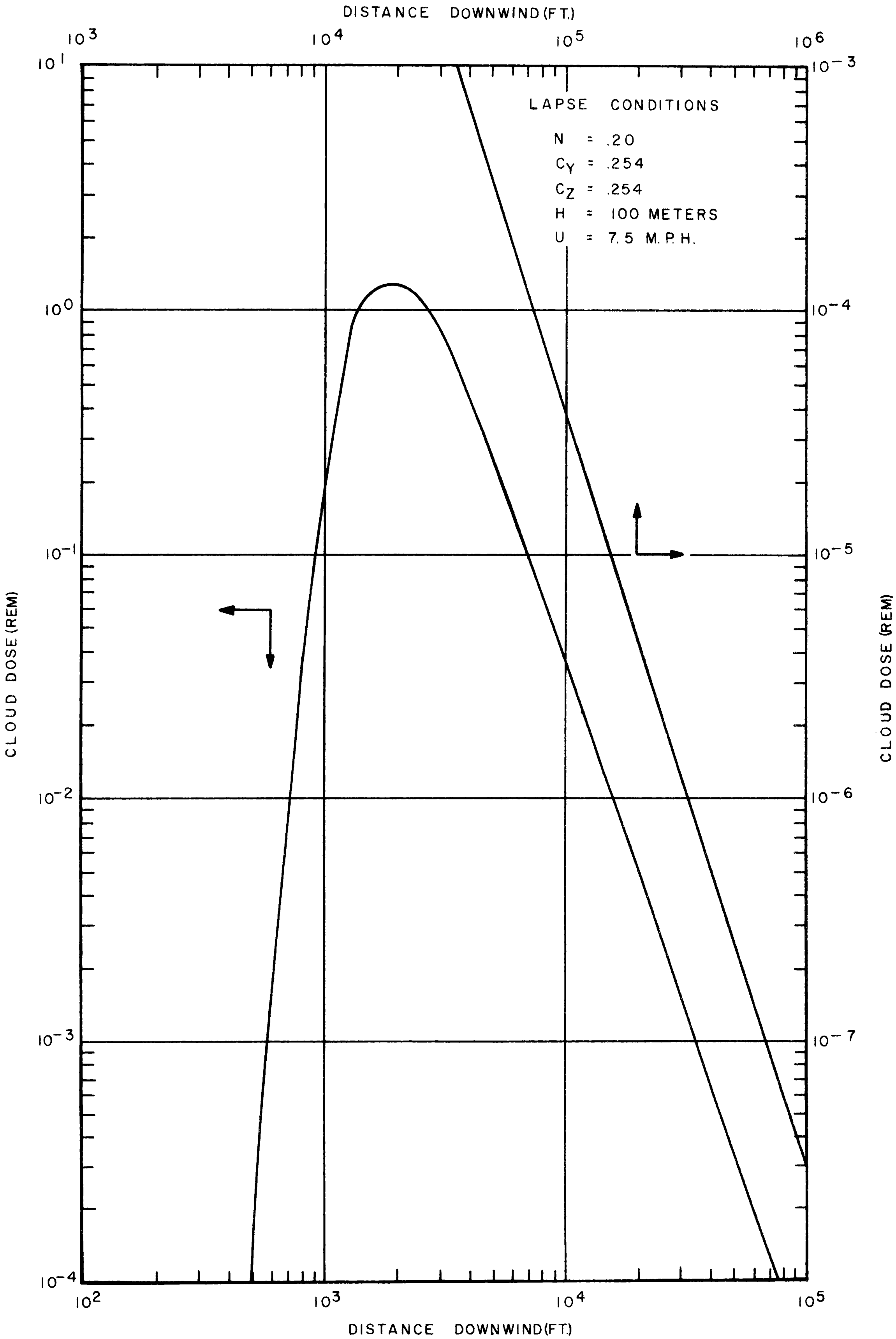


Fig. 55 - Cloud Dose For 170 Mw-sec Power Excursion
Lapse Condition, 7.5 mph Wind

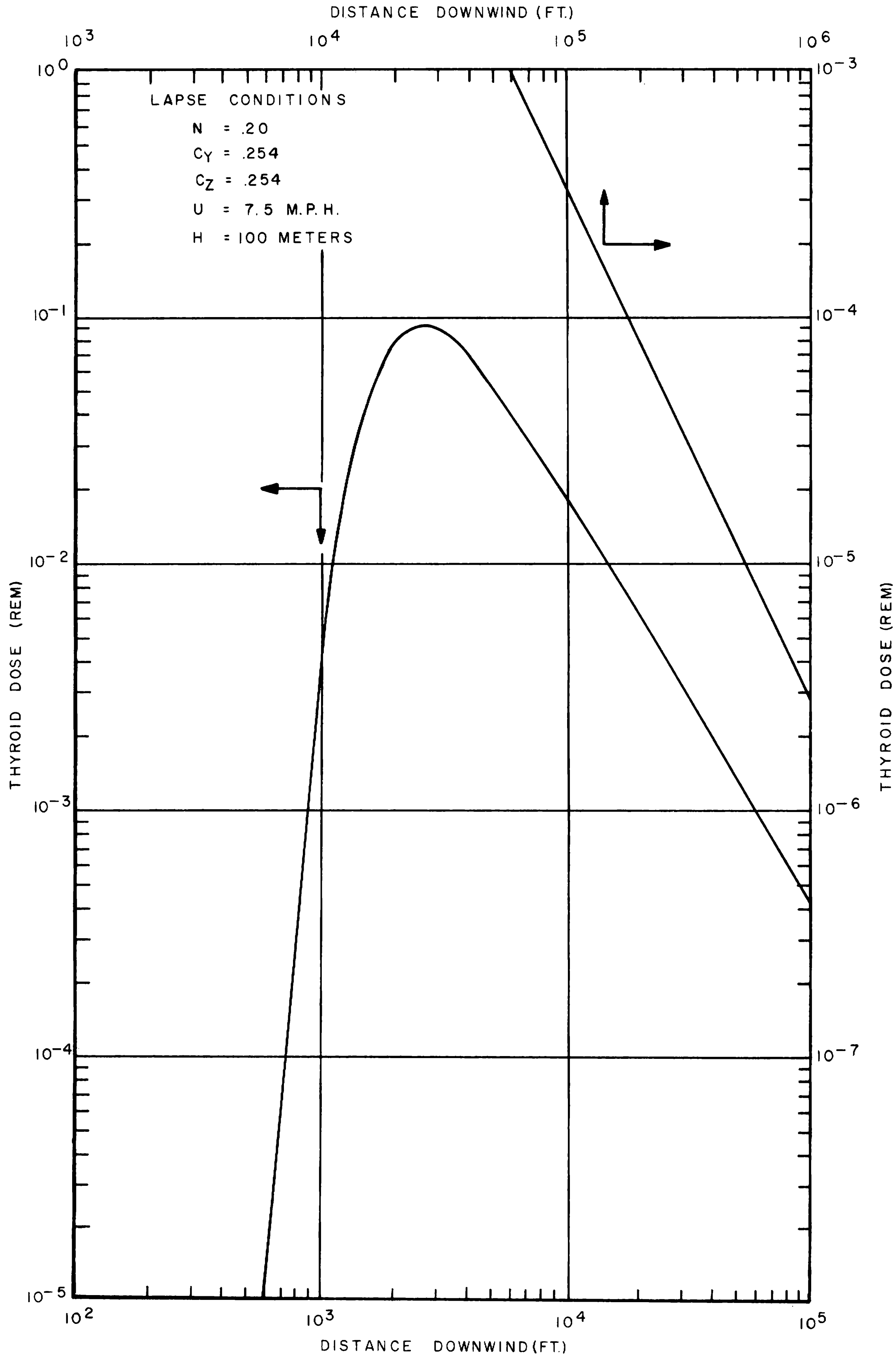


Fig. 56 - Thyroid Dose For 170 Mw-sec Power Excursion
Lapse Condition, 7.5 mph Wind

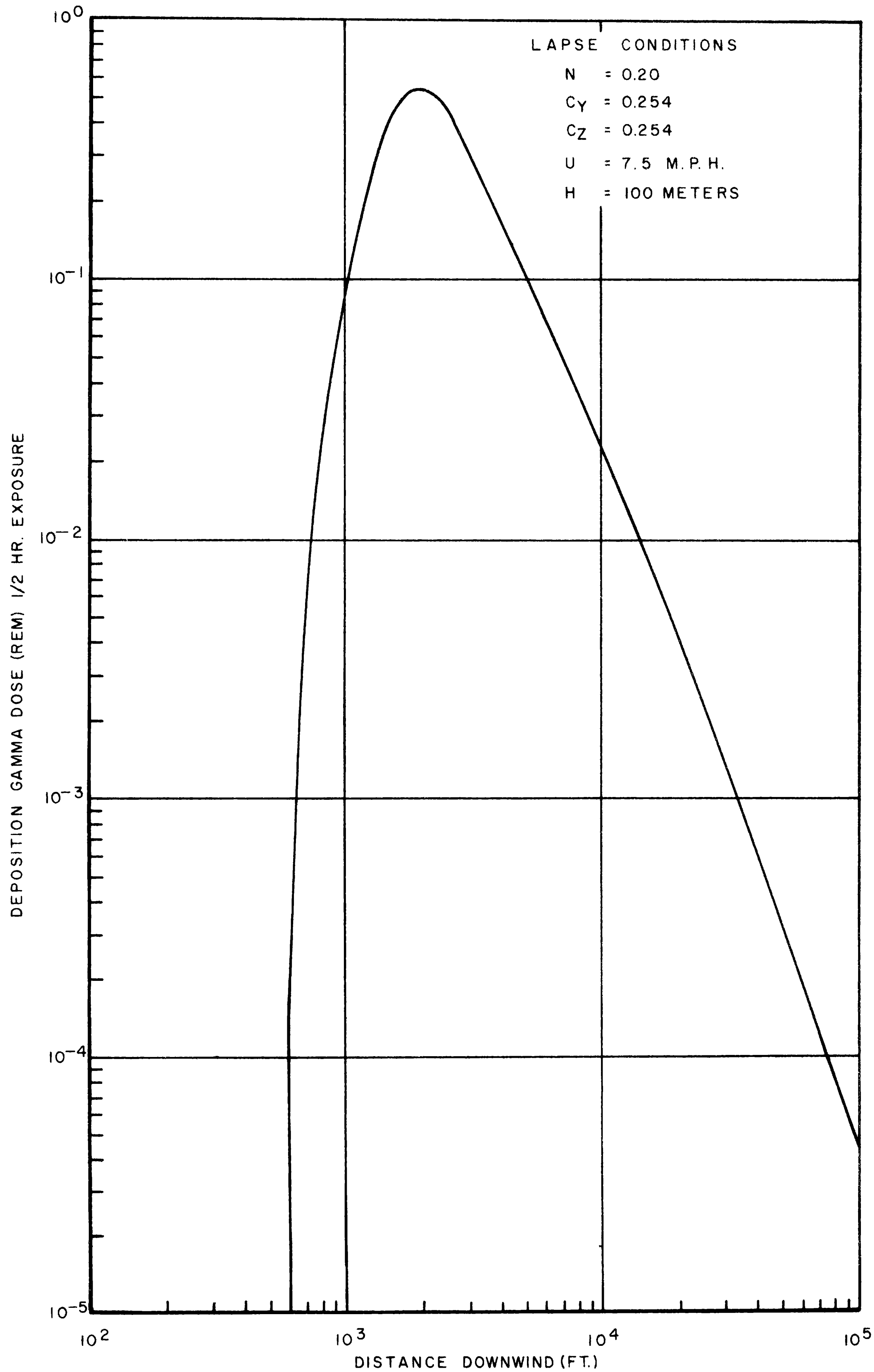


Fig. 57 - Deposition Gamma Dose For 170 Mw-sec Power Excursion (One-Half Hour Exposure)

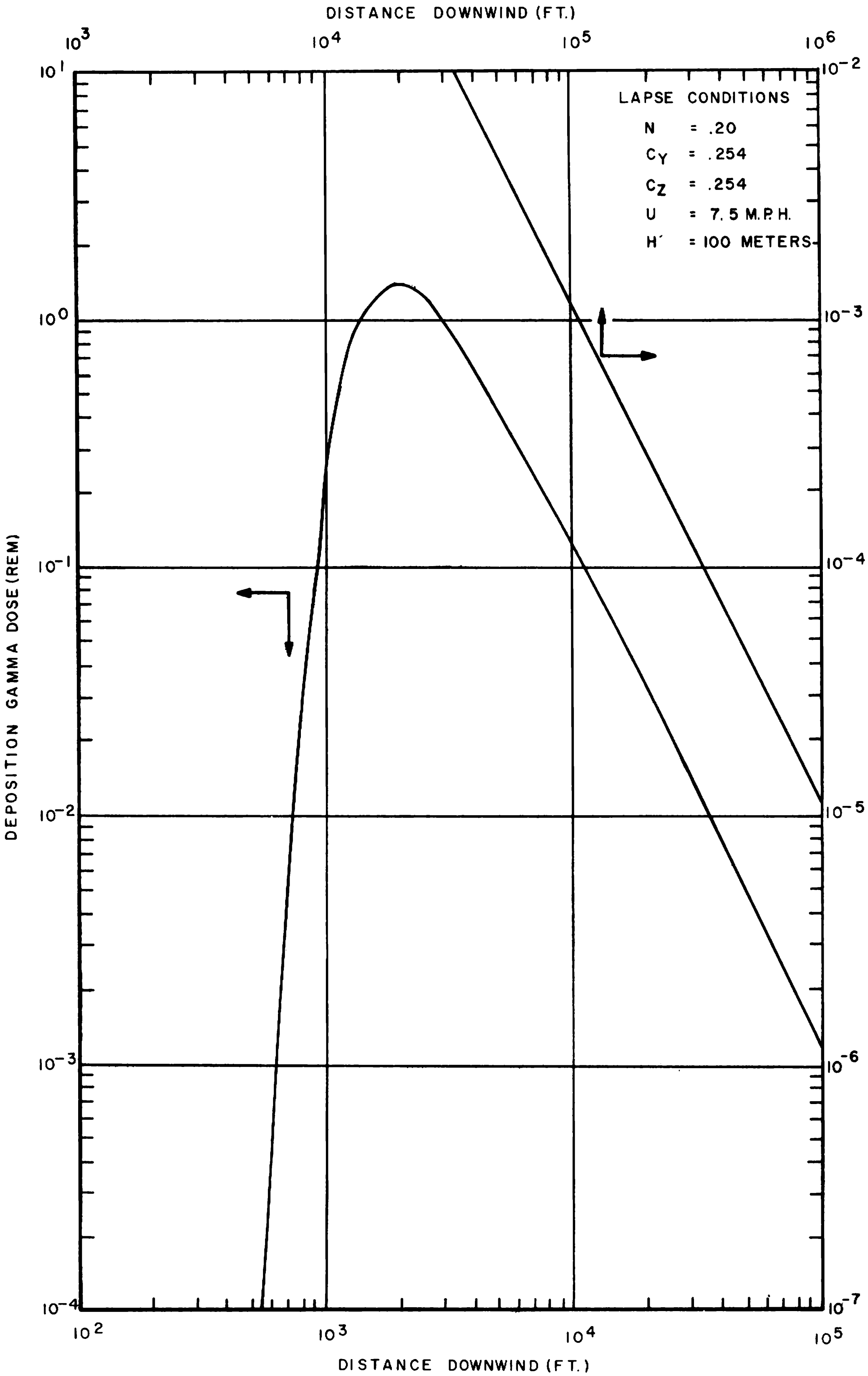


Fig. 58 - Deposition Gamma Dose For 170 Mw-sec Power Excursion Lapse Condition, 7.5 mph Wind

In the event of a wind shift, the next nearest on-site installation is the Naval Reactor Facility, which is located ~ 22 miles from the test site. The TID at this location would be 0.27 mrem from the cloud dose, including 0.25 mrem thyroid dose, and 0.85 mrem from the deposition gamma dose as shown in Figs. 55, 56, and 58. The total dose would then be 1.12 mrem which also is well below the maximum recommended 13-week dose of 3 rem[29].

It is therefore concluded that no serious hazard exists to on-site personnel from fission product activity.

The following equations were used in determining the radiological dosages to on-site and off-site personnel due to the passage of a radioactive cloud.

(3) <u>Total Integrated External Cloud Dosage (TID)</u>. The total integrated dosage from an instantaneous point source is computed by integrating the instantaneous point source equation with respect to time, ($t = x/\bar{u}$, with y, the cross wind distance = 0 and z = h. The total integrated dosage is the maximum amount of radiation to which a point at the ground may be exposed as the result of the passage of a cloud of diffusing substance[30].

$$TID = \frac{2Q\, f(t_1)\, A}{\pi c_y c_z \bar{u} (\bar{u}t)^{2-n}} \qquad \exp\left[\frac{-h^2}{c_y c_z (\bar{u}t)^{2-n}}\right]$$

c_y = Sutton's Diffusion Coefficient

c_z = Sutton's Diffusion Coefficient

} Values from Table 31 IDO-12015

(for inversion conditions, c_y is assumed to be 8 times the table values)

$f(t_1)$ = Correction for radioactive decay = $(t)^{-1.21}$

h = Cloud rise as determined by the use of Sutton's equation for the height of rise of a cloud of hot gases

n = Sutton's stability parameter

Q = Initial source strength expressed in curies and was calculated by the method presented in AECU-3066[30]

$\bar{u}$ = Mean wind speed

$\bar{u}t$ = Distance downwind from initial point of release

A = Conversion factor 2.5 rem - 1 curie sec/m^3 [31]

The cloud rise h was determined by the use of equation 6.9 as presented in AECU-3066[30].

$$h = \left[\frac{2(3m + 2p)H}{9c_p \rho \pi^{3/2} c^3 a}\right]^{\frac{1}{p + (3m/2)}}$$

m = $2 - n$

p = 1

a = 4.56×10^{-2} °C/meter

ρ = Air density = 4.36×10^{-4} g/cm^3

c_p = Specific heat at constant pressure for the gases of the cloud = 0.24 cal/gm - °C

c = Sutton's Diffusion Coefficient = 0.45 (meters)$^{n/2}$

H = Heat liberated (calories)

The total energy release, H, was calculated to be 2×10^8 calories (800 Mw-sec) which is the energy associated with the nuclear excursion, hydrogen, NaK, uranium, and zirconium release.

Assuming the excess heat of the cloud will be 2×10^8 calories, the height of rise of the cloud was calculated to be 309 meters or approximately 1000 feet. However, since the initial excess heat of the cloud can vary widely (depending on the chemical reactions occurring) the cloud may rise from a few meters to approximately 300 meters. A more reasonable estimate of the energy release predicts a cloud rise of 100 meters. However, if the most conservative case is assumed (zero stack height) the calculated dosages, beyond 2000 ft, would be no greater than those presented in Figs. 53 through 58.

(4) Total Integrated Deposition Gamma Dosage

$$D_g = \frac{BQf(t_2)v}{\pi c_y c_z \bar{u}(\bar{u}t)^{2-n}} \quad \exp\left[\frac{-Z^2}{(c_y c_z)(\bar{u}t)^{2-n}}\right]$$

$$Z = h - \frac{(\bar{u}t)(v)}{\bar{u}}$$

$$B = 3.60 \times 10^2 \text{ curies } \frac{\text{sec}}{\text{m}^2} = 1 \text{ rem} \qquad (30)$$

$$f(t_2) = \text{Correction for radioactive decay} = \frac{1}{0.21}(\bar{u}t/\bar{u})^{-.21} \qquad (30)$$

v = The terminal velocity (The assumption was made that all particles would be in the 10 to 50 micron range, approximate range of zirconium grain size, and thus the terminal velocity would be the same for all particles. The value used was for a particle size of 30 microns with a density of 6 gm/cm^3 and was determined to be 16.8 cm/sec.) (32)

(5) Gamma Dose Received from Fallout Due to One-Half Hour Exposure

$$D_g = \frac{Qvf(t_3)C}{\pi c_y c_z \bar{u}(\bar{u}t)^{2-n}} \quad \exp\left[\frac{-Z^2}{(c_y c_z)(\bar{u}t)^{2-n}}\right]$$

$f(t_3)$ = Correction factor for radioactive decay =

$$\int_{\bar{u}t}^{\bar{u}t+1800} t^{-1.21}dt \qquad (30)$$

$$C = \text{Conversion factor } 10\frac{\text{rem}}{\text{hr}} = 1\frac{\text{curie}}{\text{m}^2} \qquad (30)$$

(6) Integrated Thyroid Inhalation Dose

$$D_{TH} = \frac{(2)\sum Q_i K_i}{\pi c_y c_z \bar{u}(\bar{u}t)^{2-n}} \quad \exp\left[\frac{-h^2}{(c_y c_z)(\bar{u}t)^{2-n}}\right]$$

K_i = Conversion factor

I-131 - 329 rem = 1 curie sec/m^3

I-132 - 12.4 rem = 1 curie sec/m^3

I-133 - 92.3 rem = 1 curie sec/m^3

I-134 - 5.66 rem = 1 curie sec/m^3

I-135 - 25.3 rem = 1 curie sec/m^3

Q_i = Number of curies of the iodine isotope I-131 through I-135 (33)

b. Beryllium Contamination

The on-site beryllium concentration is not expected to present a serious hazard to personnel due to the close meteorological control to be exercised during the destructive test. Therefore, only lapse and inversion conditions were considered with a wind speed of 7.5 mph and a release height of 100 meters. For the lapse conditions, the beryllium concentration reaches a maximum value of 10 $\mu g/m^3$ at 1950 ft from the release point (Fig. 59). The air concentration at 18,300 ft, as shown in Fig. 60, under inversion conditions is below AEC recommended limits.

All beryllium concentration values were computed using the following equation[9]:

$$x = \frac{2Q}{\pi^{3/2} c^3 (\bar{u}t)^{3(2-n)/2}} \qquad \exp\left[\frac{-(Z)^2}{c^2(\bar{u}t)^{2-n}}\right]$$

where:

$$Z = h - \frac{V_d(\bar{u}t)}{\bar{u}}$$

V_d = Velocity of deposition, 10 cm/sec

Q = Beryllium initially released, 40 grams

$c^3 = c_x c_y c_z$

$c^2 = c_y c_z$

All other values are defined under Radiological Calculations.

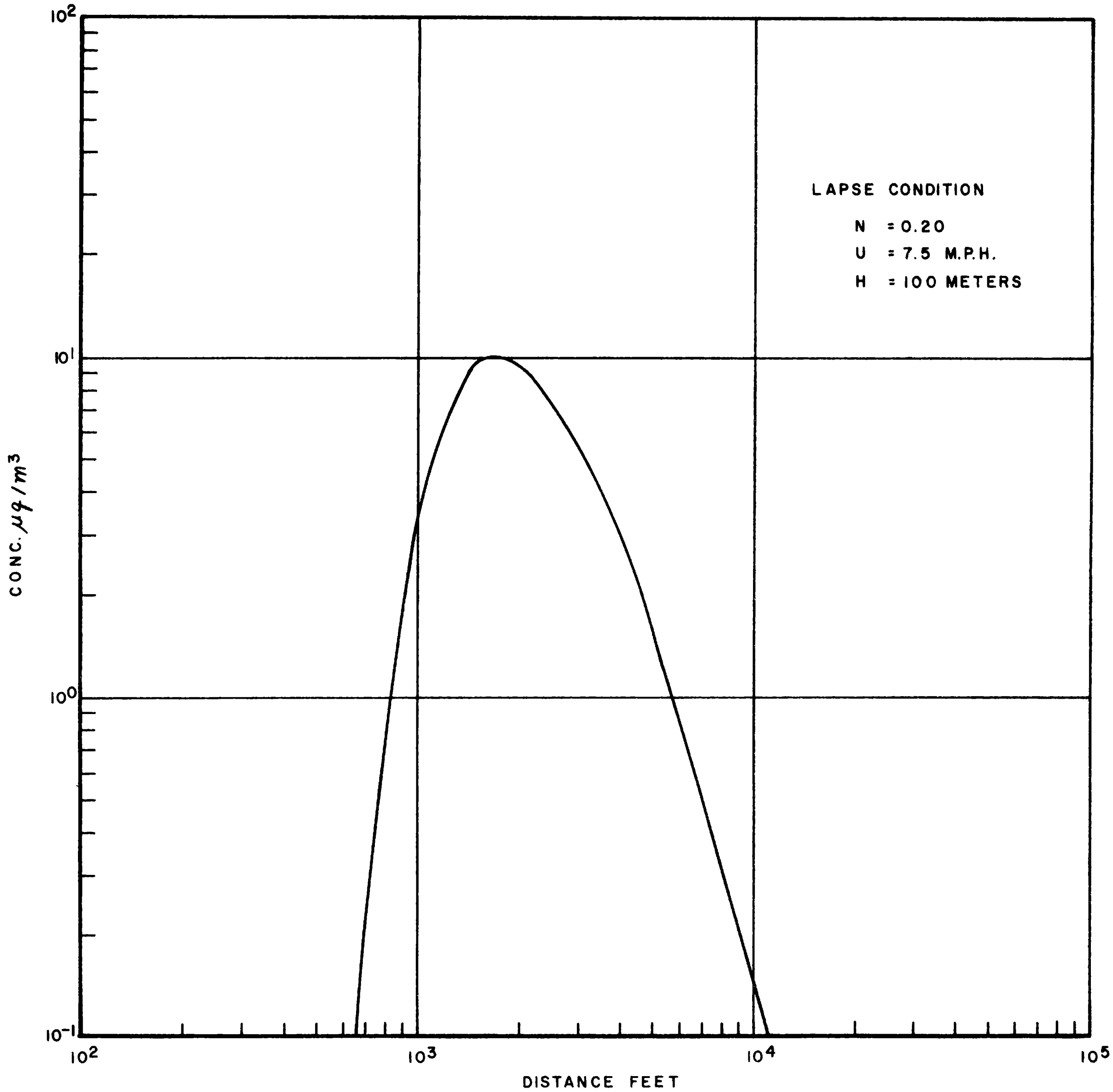


Fig. 59 - Beryllium Concentration, Lapse Condition

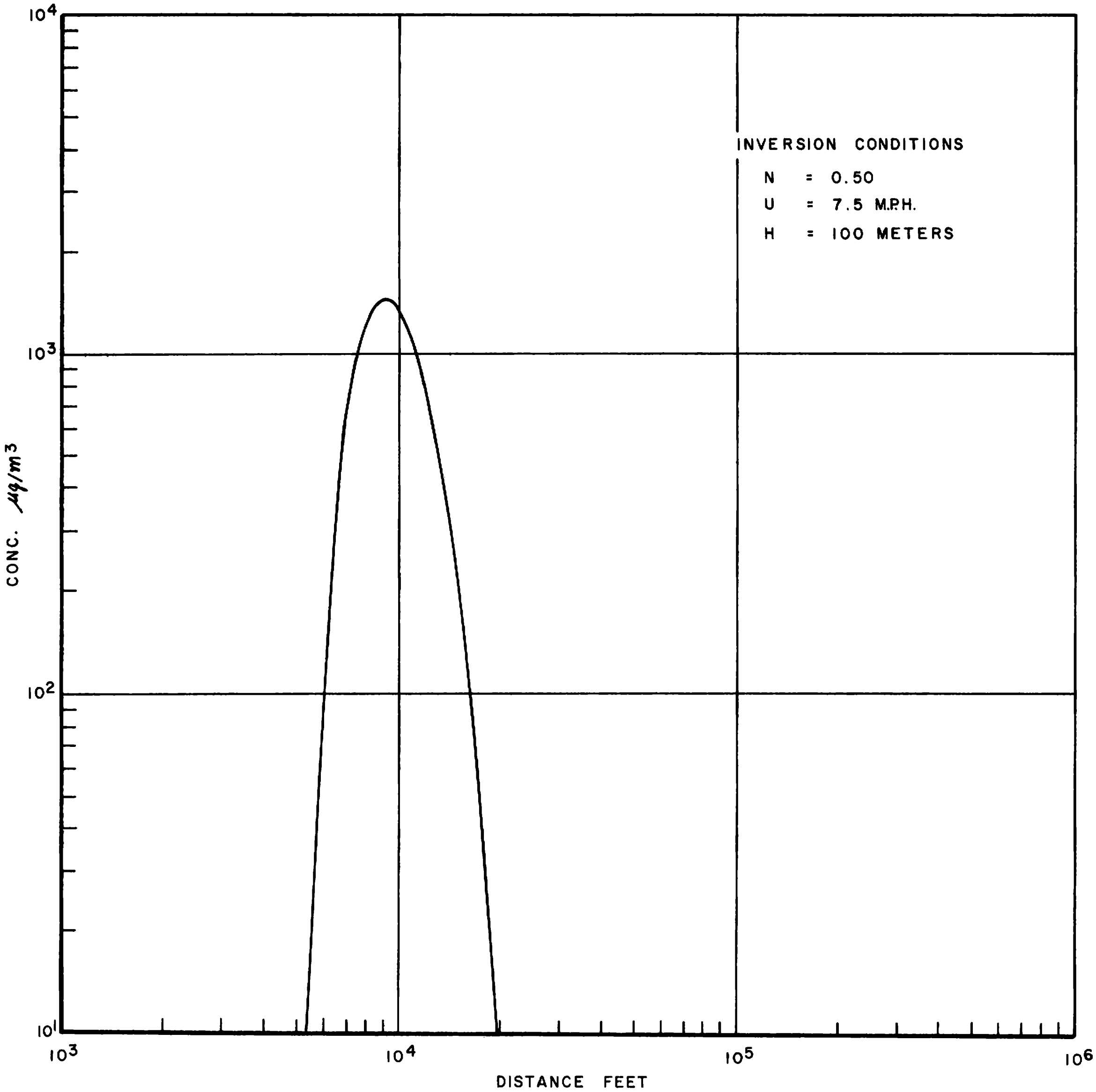


Fig. 60 - Beryllium Concentration, Inversion Condition

c. Secondary Criticality Assembly

Partial core meltdown or physical rearrangement of the core during the destructive test could conceivably result in a critical configuration with no control for shutdown available. However, the possibility of this hazard is slight as the energy release will be enough to cause disassembly of the reactor and to cause some loss of the hydrogen from the fuel.

Reactivity change due to hydrogen loss is negative and estimated to be \$0.64 per % H_2 lost. Thus, dissociation combined with physical disarrangement should result in a subcritical configuration. The requirement that there will be no large obstructions or large amounts of highly reflecting or moderating material in the immediate reactor area will preclude the possibility that materials not in the core could appreciably contribute to the criticality hazard.

2. Off-Site Hazards

a. Radiological Hazards

Under the conditions of meteorological control discussed in the previous section, the nearest inhabited area downwind from the test site is approximately 6.5 miles, and the nearest town, Monteview, is approximately 12 miles. The assumptions used in the computation of cloud and fallout exposures were the same as those used in the discussion of on-site hazards; namely, 100% release of the core to the atmosphere, with 86% of the fission products appearing as fallout and 14% appearing as gaseous products.

The TID from gaseous products and fallout are shown in Figs. 55 and 58 as a function of downwind distance from the test site. The thyroid dose from inhalation of the gaseous products is shown in Fig. 56. As seen in the preceding figures, the TID beyond 80,000 ft (approximately 15 miles) is less than 2 mrem. Table V-4 lists the exposure to the population of towns lying to the north and northeast of the test site as shown in Fig. 20. The exposures listed in Table V-4 were based on a 7.5 mph wind and lapse conditions; however, the destructive test will be run only when the wind speed is greater than 10 mph. Since the radiological dose is inversely proportional to wind speed, the greater the wind speed the smaller will be the tabulated values presented in Table V-4.

TABLE V-4

TOTAL INTEGRATED DOSAGE FROM CLOUD AND FALLOUT UNDER METEOROLOGICAL CONTROL

Town	Dist. from IET (miles)	Cloud (mrem)	Fallout (mrem)	Thyroid (mrem)	TID (mrem)
Monteview	12	0.15	3.0	0.90	4.05
Winsper	21	0.03	0.90	0.25	1.18
Camas	26	0.015	0.65	0.18	0.85
Small	30	0.009	0.42	0.13	0.56
Dubois	33	0.007	0.38	0.11	0.50

Since Monteview is the closest town to the test site, the effect of a total instantaneous washout at this point was considered. The resultant TID to the population in this area would be approximately 54 mrem assuming all fission products are in the gaseous state. This assumption results in an overestimate of the dose since the gaseous contribution to the cloud is expected to be approximately 14%.

Although the tests will be run under meteorological control, the results of conducting the test with no control were estimated. Figs. 61, 62, and 63 give the cloud, fallout, and thyroid exposure for inversion conditions at the same wind speed as used in the controlled case (7.5 mph). The effect of doubling the wind speed for lapse conditions is shown in Figs. 64 and 65. The effect of doubling the wind speed for inversion conditions is shown in Figs. 66 and 67. The exposure to the population of towns lying in all directions from the test site is given in Table V-5. These exposures were computed assuming the wind blows directly from the test site to the town in question.

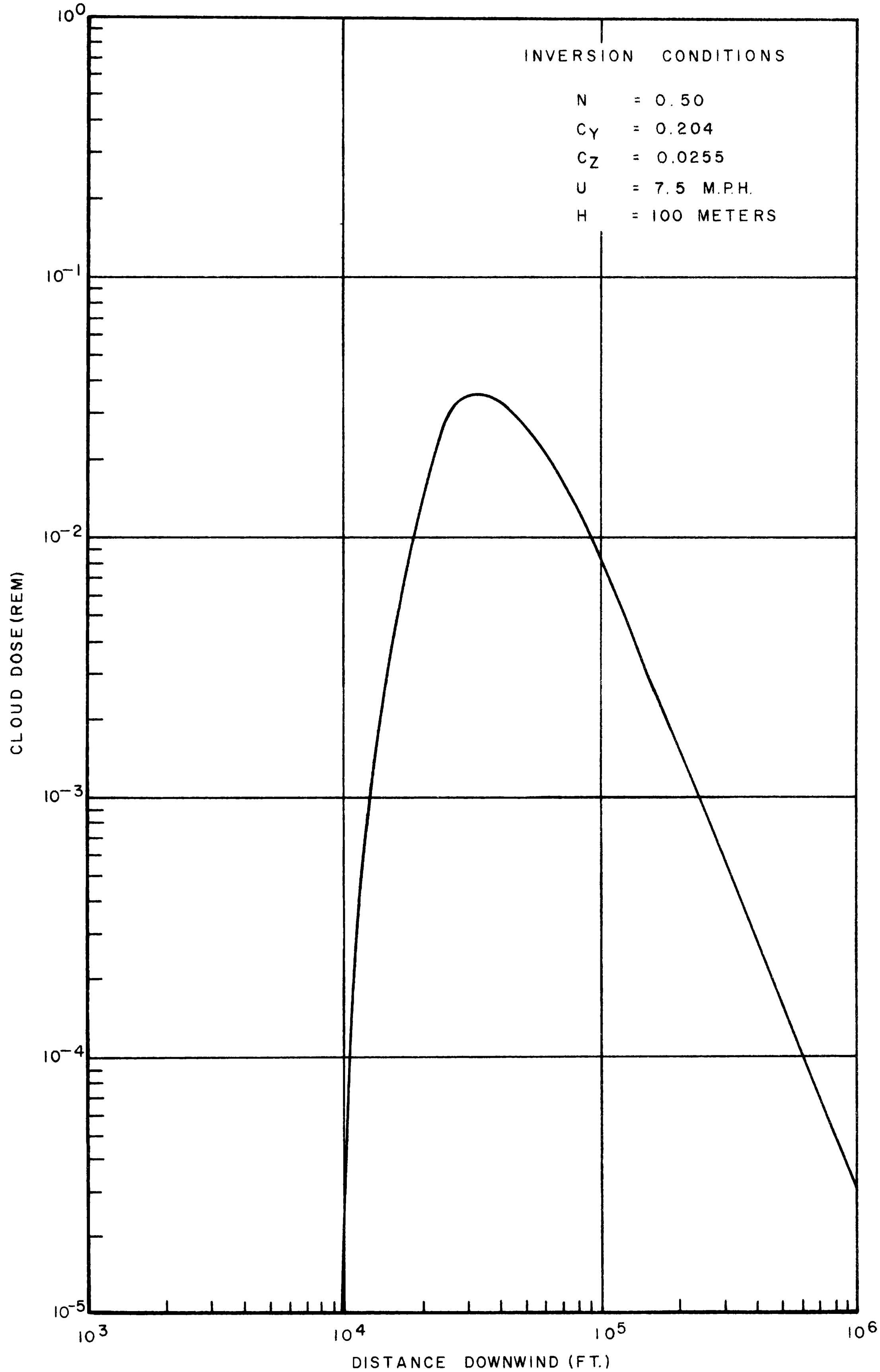


Fig. 61 - Cloud Dose For 170 Mw-sec Power Excursion Inversion Condition, 7.5 mph Wind

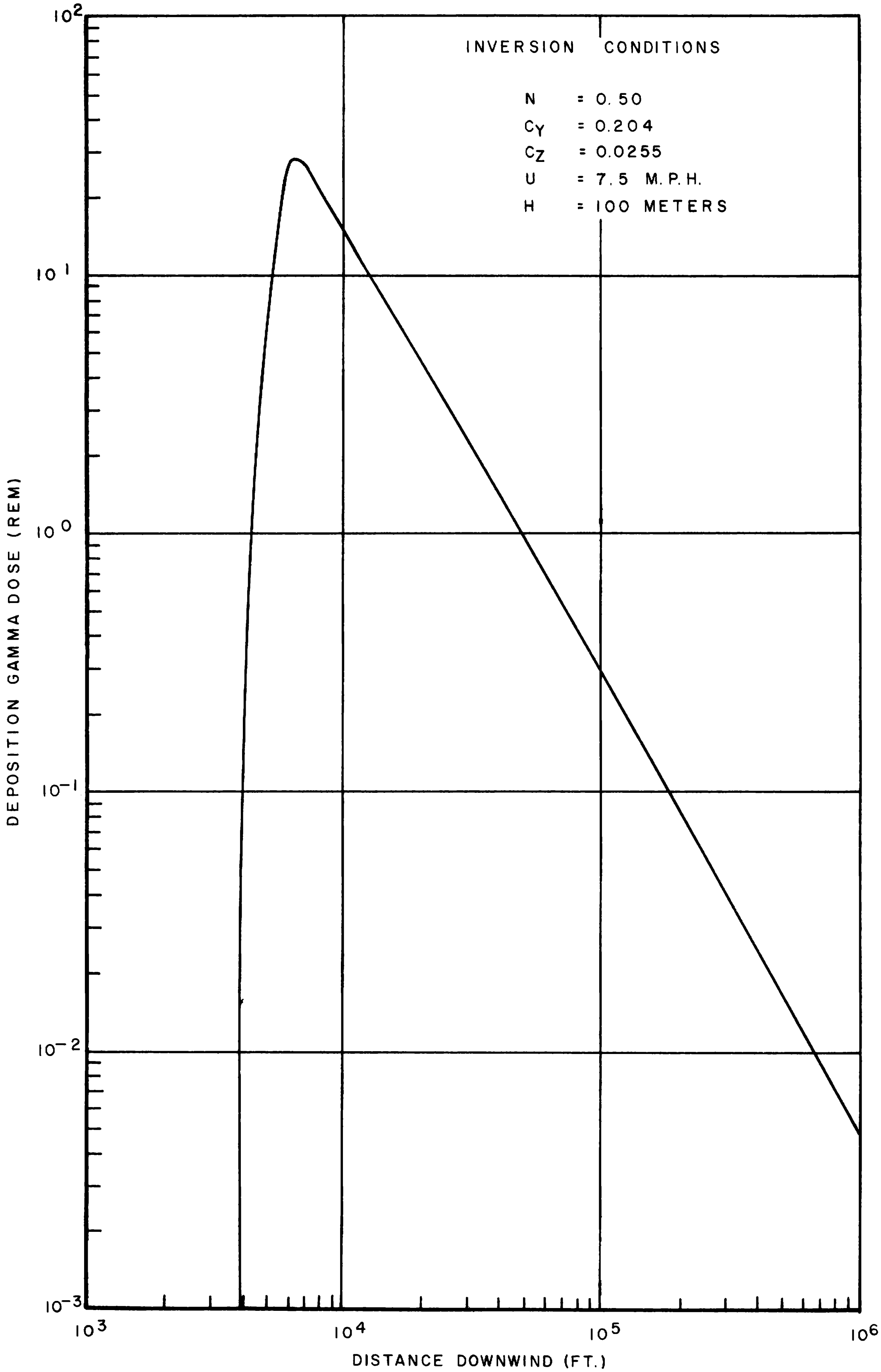


Fig. 62 - Deposition Gamma Dose For 170 Mw-sec Power Excursion
Inversion Condition, 7.5 mph Wind

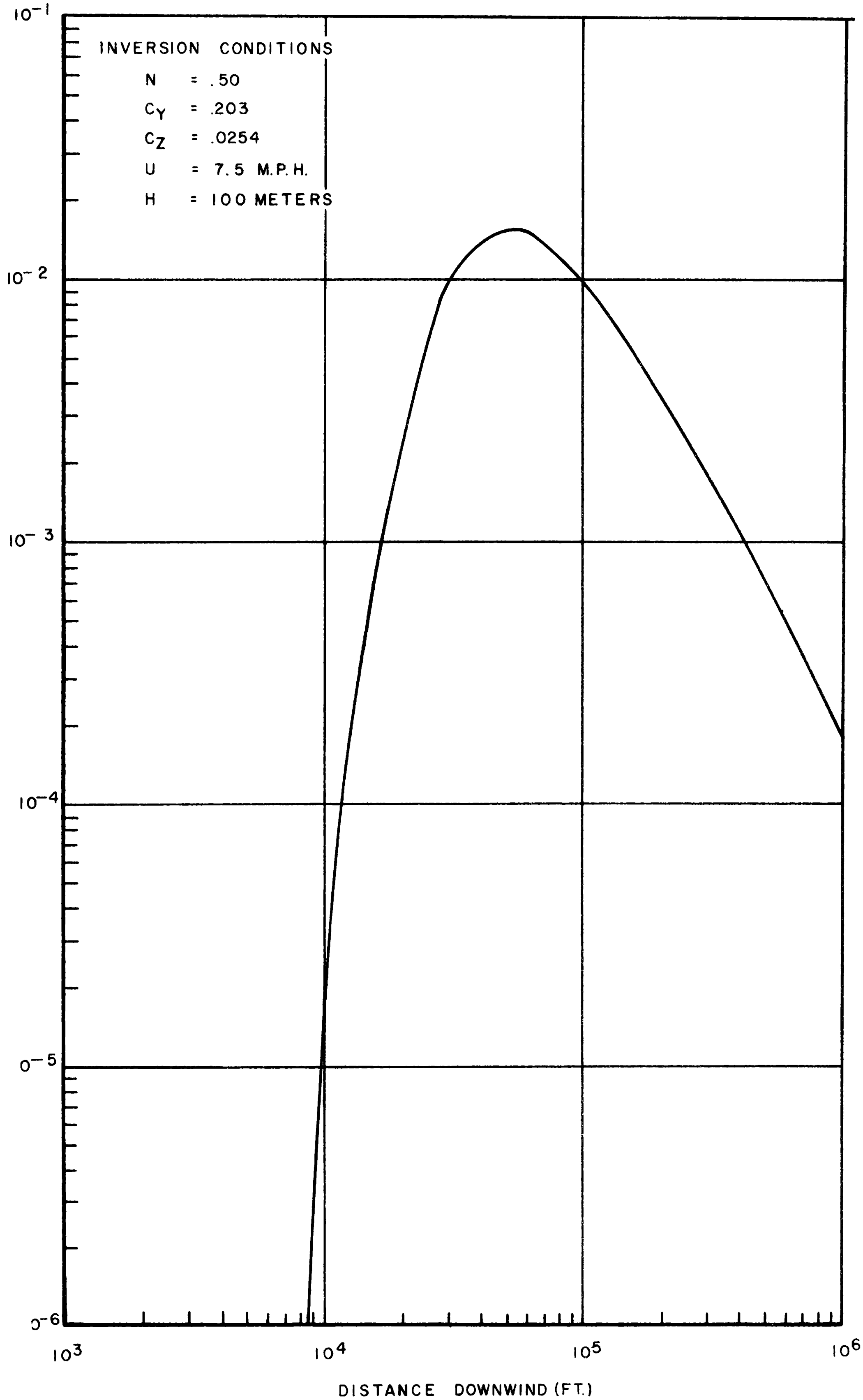


Fig. 63 - Thyroid Dose For 170 Mw-sec Power Excursion Inversion Condition, 7.5 mph Wind

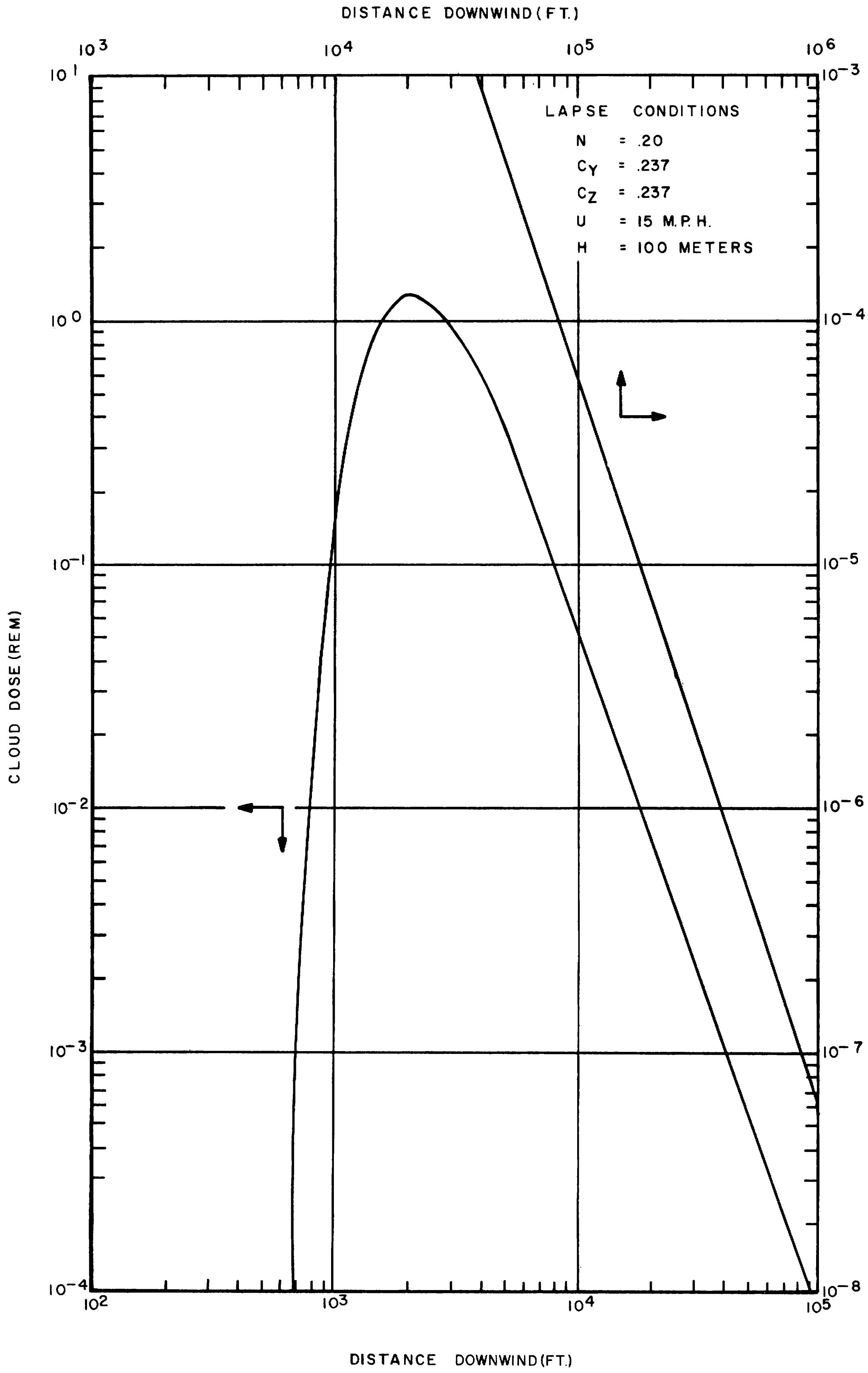


Fig. 64 - Cloud Dose For 170 Mw-sec Power Excursion
Lapse Condition, 15 mph Wind

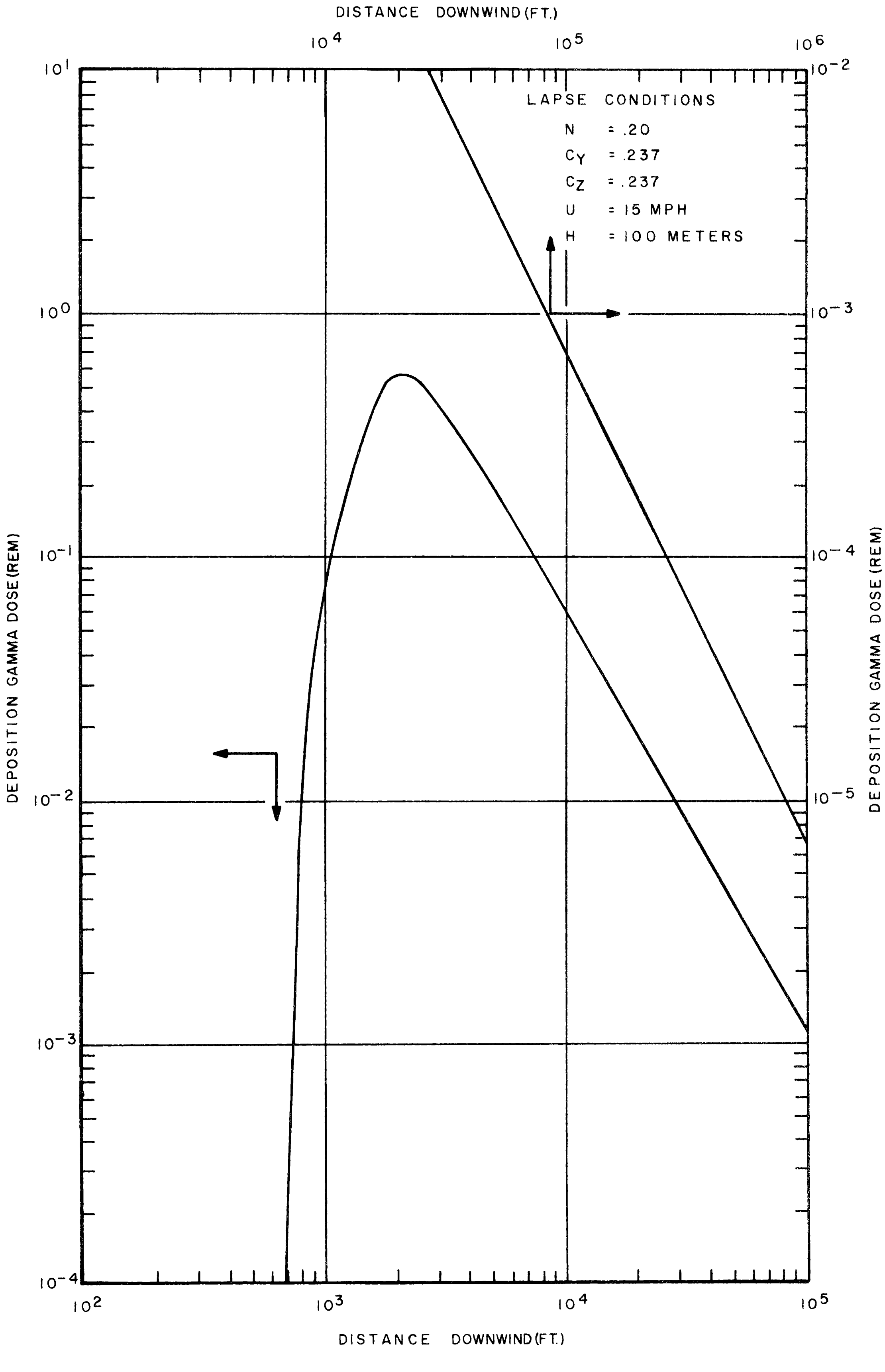


Fig. 65 - Deposition Gamma Dose For 170 Mw-sec Power Excursion Lapse Condition, 15 mph Wind

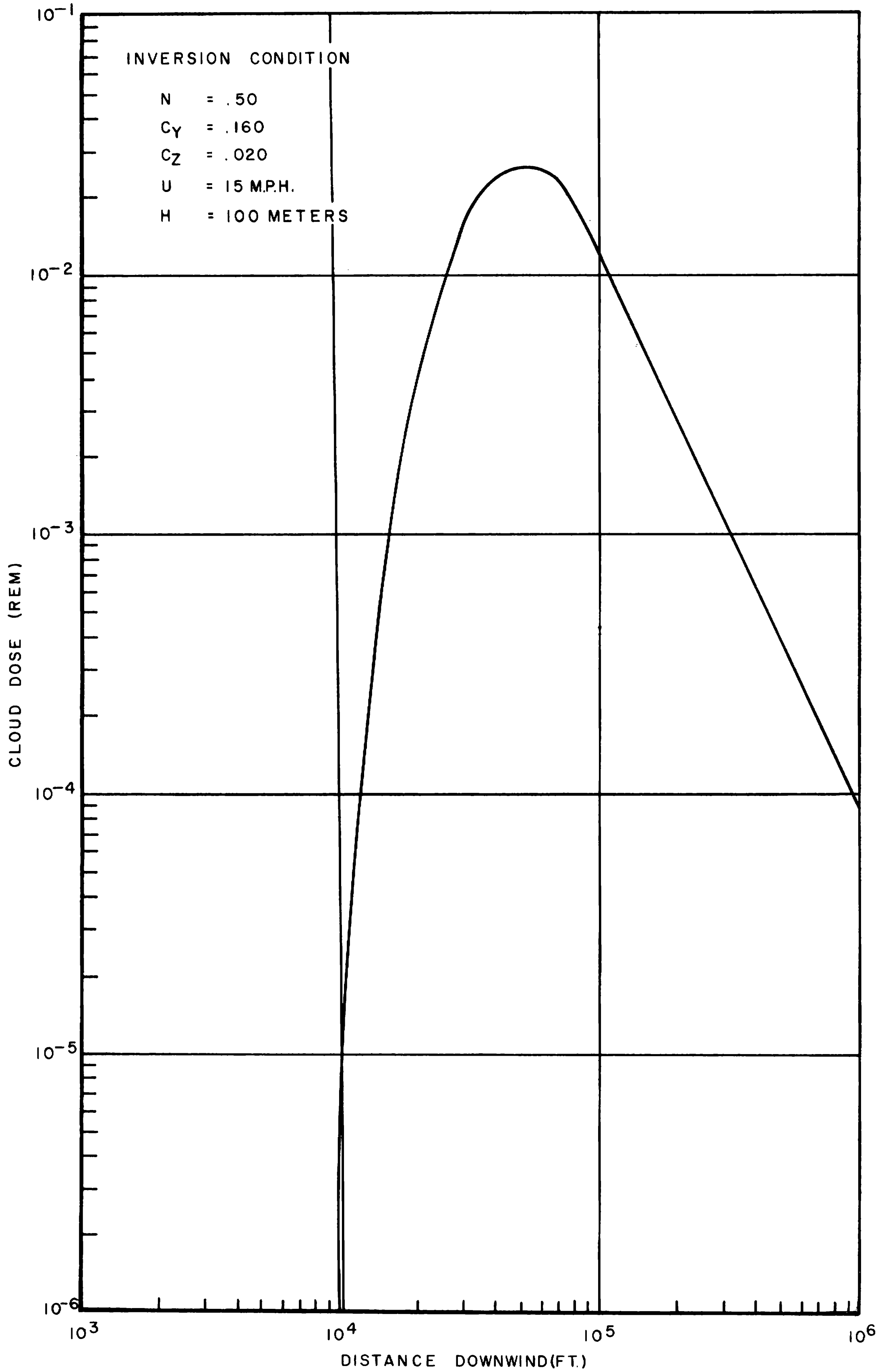


Fig. 66 - Cloud Dose For 170 Mw-sec Power Excursion
Inversion Condition, 15 mph Wind

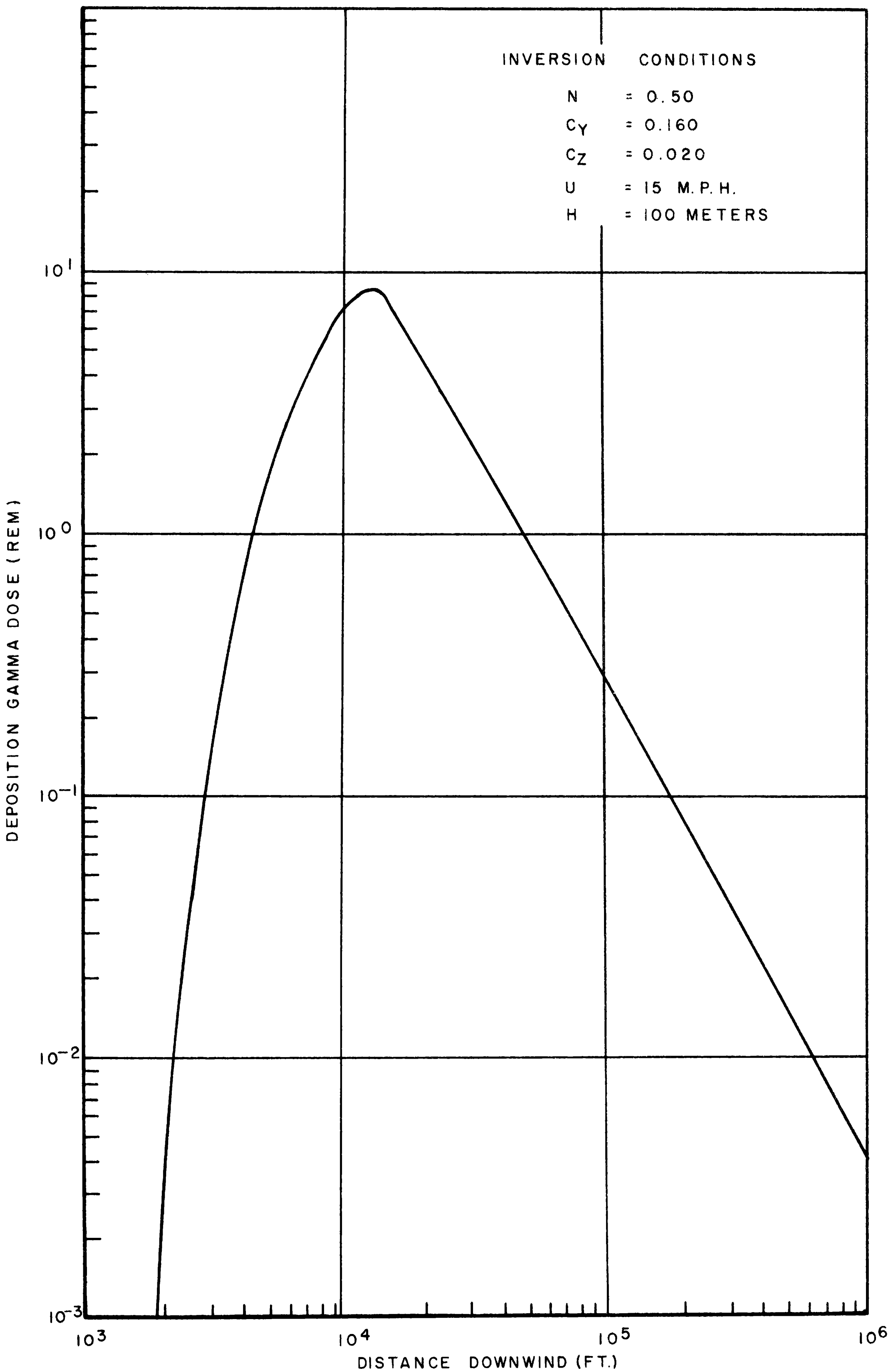


Fig. 67 - Deposition Gamma Dose For 170 Mw-sec Power Excursion
Inversion Condition, 15 mph Wind

TABLE V-5

TOTAL INTEGRATED DOSAGE FROM CLOUD AND FALLOUT UNDER NO METEOROLOGICAL CONTROL

Town	Distance from IET (miles)	Total Integrated Dosage (mrem) Lapse Conditions $\bar{\mu}$ = 7.5 mph	Lapse Conditions $\bar{\mu}$ = 15 mph	Inversion Conditions $\bar{\mu}$ = 7.5 mph	Inversion Conditions $\bar{\mu}$ = 15 mph
Monteview	12	4.05	2.90	920	645
Terreton	13	2.63	2.44	578	553
Mud Lake	15	1.88	1.86	433	428
Howe	16	1.77	1.74	392	377
Winsper	21	1.18	0.64	246	240
Roberts	30	0.56	0.32	133	130
Small	30	0.56	0.32	133	130
Dubois	33	0.50	0.25	112	109
Atomic City	33	0.50	0.25	112	109
Arco	34	0.47	0.22	107	104
Menan	37	0.39	0.19	98	87
Idaho Falls	39	0.35	0.18	86	81

Under meteorological control the total integrated dosage to the general population is well below the 0.17 rem limit as recommended by the Federal Radiation Council[29]. However, under no meteorological control and for towns lying within 16 miles downwind of IET the dosage received by the general public is approximately four times the maximum recommended value of 0.17 rem. As seen in Table V-5, for distances greater than 21 miles, the total exposure is well below the acceptable limit.

The indirect biological effect of ground deposition has been considered for the rural areas to the north and northeast of IET. The biological effect considered was the radiation dose to a child's thyroid gland which resulted from ingestion of contaminated milk. For the purpose of the ingestion calculation it was assumed the milk was obtained from a cow which had been grazing in the immediate area of the nearest site boundary to IET. This boundary, under the specified meteorological conditions, is ~ 6.5 miles northeast of the test area. The dosage to a child's thyroid, based on the above conditions and following equation, was calculated to be 340 mrem. This is considered to be a conservative value since depletion of the cloud by scavenging and radioactive decay was neglected; however, the 340 mrem dose is still below the allowable limit of .5 rem as recommended by the Federal Radiation Council[26].

$$D = \int_0^t \frac{(5.92 \times 10^2)(x) v_g f_w f_i f_m m f_w \bar{E}\, e^{-\lambda_e t}\, dt}{(v_m)(w)}$$

D = Thyroid dose (rem)

x = Cloud concentration (curie-sec/m^3)

v_g = Deposition velocity (0.02 m/sec)

f_w = Vegetation area weight factor (0.02 m^2/gm)

f_i = Animal intake factor (4×10^4 gm/day)

f_m = Fraction of daily intake to milk (0.1)

v_m = Daily mass of milk (2×10^4 gm/day)

m = Milk density (1 gm/cm^3)

f_w = Fraction of ingested material reaching the thyroid

$\bar{E}$ = Effective energy absorbed in the thyroid (mev/dis)

λ_e = Effective decay constant (sec^{-1})

t = The time over which the dose rate is integrated (sec)

w = Mass of the thyroid gland (gm)

b. Beryllium Contamination

The beryllium contamination was computed using Sutton's diffusion equation and assuming that 1% of the beryllium in the core was broken into grain size, approximately 35 microns. It was further assumed that the beryllium (40.4 grams) was released as a cloud at a height of 100 meters with a wind speed of 7.5 mph. The results were computed for two separate atmospheric conditions, lapse and inversion.

For the lapse condition, the beryllium concentration is 0.45 $\mu g/m^3$ at a distance of 7000 ft from the release point and decreases very rapidly with distance, as shown in Fig. 59. By the time the nearest site boundary is reached, 29,000 ft, the beryllium concentration is essentially zero.

In the case of inversion, the beryllium concentration is 10 $\mu g/m^3$ at 20,000 ft from the release point, and is decreasing rapidly with distance, as shown in Fig. 60.

In both cases, the concentrations in air are well below AEC recommended limits long before the site boundaries are reached, and in both cases the values are decreasing rapidly with distance.

E. Post-Destructive Hazards

Surface contamination, airborne contamination, and direct radiation in the IET and TSF areas were considered in the post-destructive analysis. Time limitations for personnel in the above areas were estimated. However, under no circumstances will a person be admitted to the area in question without proper Health Physics approval. Reentry procedures following the destructive test are discussed in Section VI.

1. Radiological Hazards

a. Initial Engineering Test (IET) Facility

(1) Radiation. The radiation hazards resulting from fission product decay following the destructive test have been evaluated for the following cases:

(1) Direct radiation from fission products when all activity is retained on the test pad.

(2) Direct radiation from fission product fallout when all activity is released to the atmosphere.

The gamma dose rate as a function of distance from the building, assuming all fission products are retained on the test pad, is presented in Fig. 68. This analysis is based on the conservative assumption that the fission products form a line source just inside the building. The source is assumed to have no self-shielding or external shielding except air.

The predicted dose rates and access time limits for the various work areas at one hour and 24 hours after shutdown are presented in Table V-6.

On the basis of this analysis, continuous access to the area inside the security fence surrounding IET, except inside the Control and Equipment Building, cannot be permitted for at least 24 hours following the destructive test.

After one week, access to the building can be permitted for approximately 16 minutes before personnel receive 300 mrem.

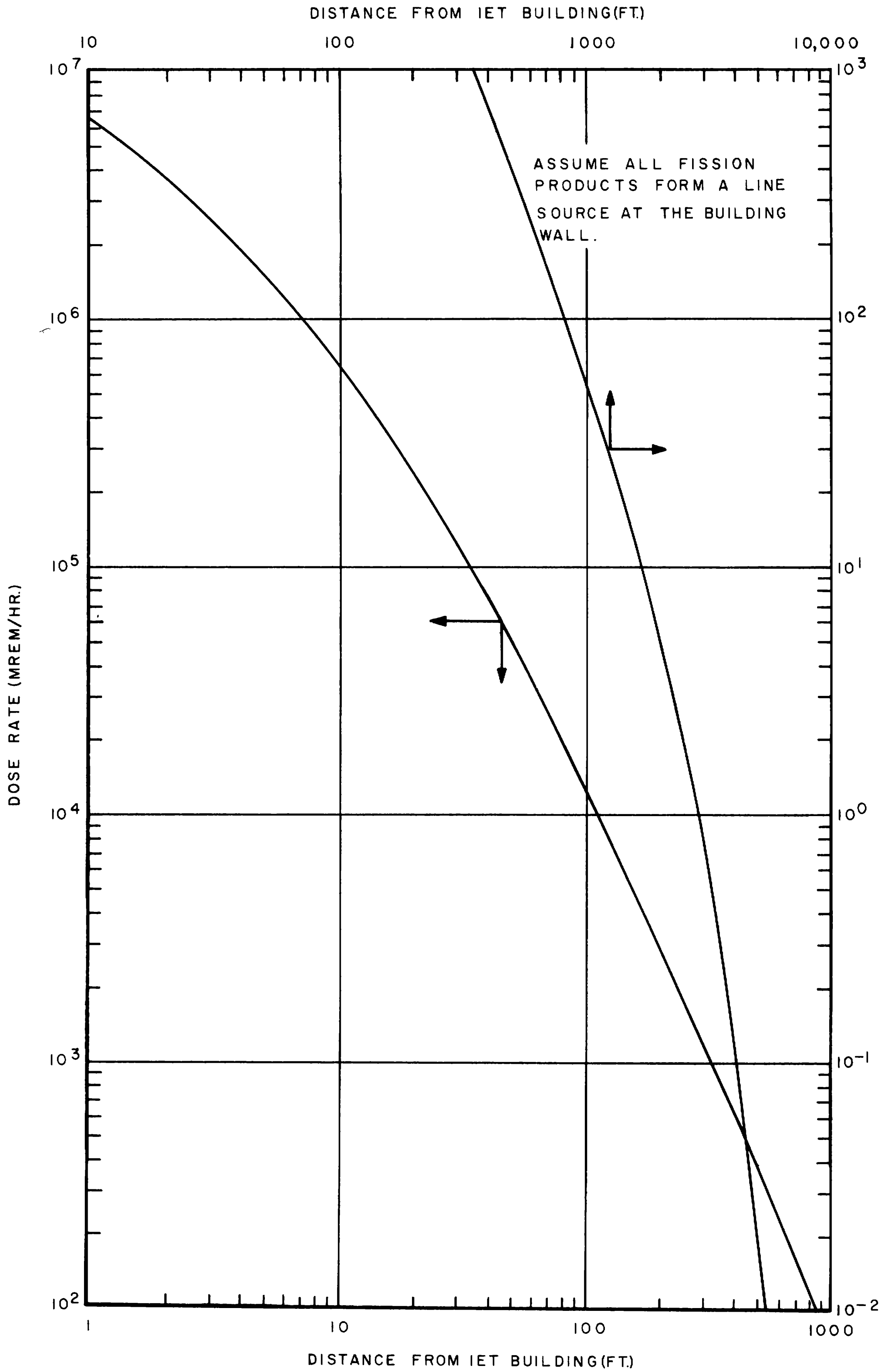


Fig. 68 - Gamma Dose Rate One Hour After Power Excursion

TABLE V-6

RADIATION LEVELS FOLLOWING DESTRUCTIVE TEST

Location	Distance From IET Building (ft)	Dose Rate (mrem/hr) 1 Hour	Dose Rate (mrem/hr) 24 Hours	Access Time Limits 1 Hour	Access Time Limits 24 Hours
Fenced area around the TSF	> 4000	< 1	< 1	None	None
End of tunnel at IET	~ 1400	20	< 1	15 hrs/wk	None
Security fence around IET	~ 170	3.7×10^{3}	56	5 min/wk	5.3 hrs/wk
Inside reactor building*		9.0×10^{5}	1.3×10^{4}	No access permitted	1.4 min/wk
IET Control and Equipment Building		< 0.1	< 0.1	None	None
Coupling Station		< 0.1	< 0.1	None	None

*An infinite plane source geometry was assumed inside the building.

As the building will not be over the pad at the time of the destructive test, it is expected that a large percent of the fission products will be admitted to the atmosphere and removed from the local area. The remaining activity is expected to be scattered over a larger area than when enclosed in the building. Thus, the dose rate, because of this wider distribution, will not exceed the predicted dose rate (9.0×10^5 mrem/hr) inside the building. Here also, extreme caution will be exercised when approaching the area.

The maximum external dose rate from fission product fallout will occur in the event all the fission products are released to the atmosphere and distributed downwind from the test pad. The dose rate in the area of maximum fallout will be approximately 3 mrem/hr at 24 hours following the release. A more complete evaluation is presented in Section V-D-1.

The total gamma energy (mev) emitted by the fission products as a function of operating time (t_o) and shutdown time (t_s) is[21]:

$$\text{Total Power} = 3.1 \times 10^{10}\ P\ G\ (t_o, t_s)\ \frac{\text{mev}}{\text{sec}}$$

where: P = reactor power in watts

$G\ (t_o, t_s) = G\ (\infty, t_s)\ -\ G\ (\infty, t_o + t_s)$

$G\ (t_o, t_s)$ = total gamma energy as a function of operating and shutdown times.

When: (t_o, t_s) = such as a burst, the total power emission is:

$$\text{Total power} \left(\frac{\text{mev}}{\text{sec}}\right) = 3.1 \times 10^{10}\ P\ t_o\ a\ G\ (t_o, t_s)$$

where: $a = 0.2019$ for $1.5 \times 10^2 \leq t_s \leq 10^6$ sec

$(P)\ (t_o)$ = integrated power excursion (100 Mw)

t_s = shutdown time (3.6×10^3 sec)

The gamma energy spectrum of t_s was also obtained from WAPD-R(F)-38[21].

To determine the gamma dose rate from fission product decay, the source geometry was assumed to be a line source for dose point distances less than 500 ft from the building and a point source for distances greater than 500 feet.

$$\phi = B \frac{S_L}{2\pi\ a} \quad F\ (\theta,b) \text{ for a line source}$$

where: $S_L = \frac{S_o}{L} = \gamma\text{'s/cm}$

S_o = gamma source strength of energy E (γ's/sec)

L = length of building (1.83×10^3 cm)

B = dose buildup factor for water

b = μt for air

$\theta = \tan^{-1} \frac{1/2L}{a}$

a = distance from building to dose point

The point source technique is the same as that described in Section V-C.

The radiation level inside the building was determined by assuming that the fission products were evenly distributed over the floor of the building. An infinite plane source geometry was used.

$$\theta = \frac{S_a}{2} \quad E_1(b_1)$$

where:

$S_a = \frac{S_o}{A} \quad \gamma\text{'s/cm}^2$

A = area of building ($2.52 \times 10^6 \text{cm}^2$)

$E_1(b_1) \rightarrow 4$ as $b_1 \rightarrow 0$

(2) Surface Contamination. Assuming 100% of the fission products are released to the atmosphere, radiological contamination of the test pad and downwind area will cause a radiation hazard sufficient to restrict access to the immediate area for at least one day following the destructive test. After the necessary delay time, access to the area will be permitted under proper Health Physics surveillance.

(3) Airborne Contamination. Before personnel enter the test area, airborne contamination from the radioactive cloud will have had sufficient time to diffuse into the atmosphere. However, airborne contamination can be expected as a result of personnel or

vehicles moving in the area and disturbing the surface contamination. Therefore, respirators will be available for use during cleanup of the area.

b. Transportation to Examination Area

(1) Contamination Spread. The spreading of radioactive materials along the railroad right-of-way is a possible hazard associated with transporting the reactor debris to the Examination Area. To reduce and maintain the spread of materials to a minimum, the dolly carrying the radioactive material will be covered to prevent the material from being blown or jarred from the dolly. In addition, loose particles that may have been scattered during the destructive test will be removed from those parts of the dolly which are not covered.

As an added precaution, a distance of 500 ft on each side of the tracks will be an exclusion area until it has been surveyed and decontaminated.

(2) Radiation. The major radiation hazard associated with transporting the reactor to the Examination Area is the direct gamma radiation from the fission product decay. This has been evaluated for the maximum radiological condition assuming all the fission products, following the destructive test, have been reassembled on the dolly with no shielding except air.

The gamma dose rate as a function of distance from the source, (a point source) is presented in Fig. 69. The analysis is made for 24 hours after shutdown since it is felt that this is a reasonable time after shutdown at which personnel would be allowed access to the area for the purpose of preparing the reactor for movement. The analytical techniques are the same as those used to determine the gamma dose rate following the destructive test.

The gamma dose rates at various locations along the dolly track while transporting the reactor to the Examination Area are presented in Table V-7.

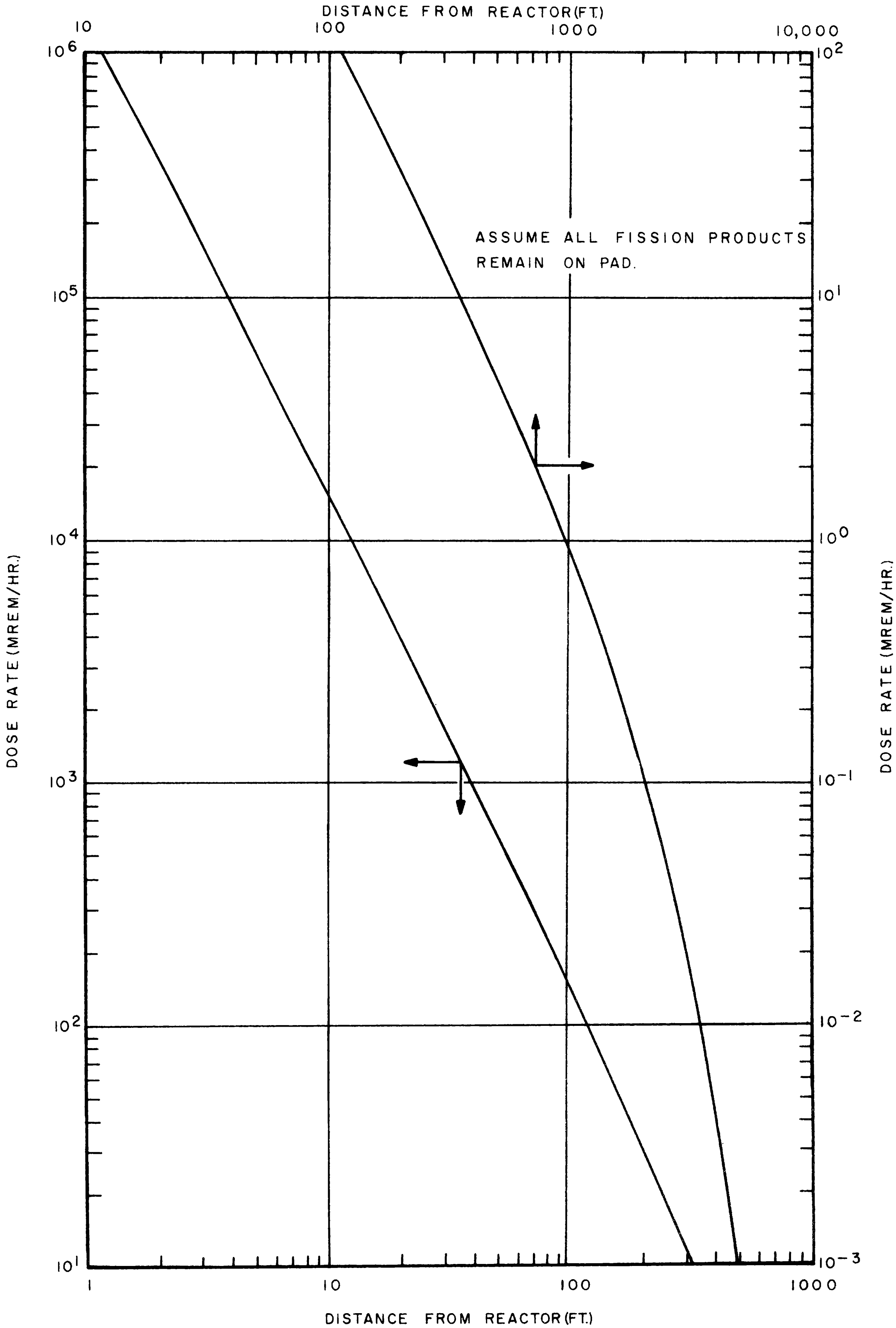


Fig. 69 - Gamma Dose Rate 24 Hours After Power Excursion

TABLE V-7

RADIATION LEVELS WHILE TRANSPORTING A REACTOR TO TSF FOLLOWING DESTRUCTIVE TEST

Location	Approx. Dist. from Track (ft)	Dose Rate (mrem/hr)	Access Time Limits
Point on Taft Road nearest the track	900	< 2	None
Administration Building	1000	< 1	None
Nearest point to security fence around Administration Building - not shielded by the earth embankment	500	4.7	None
Shielded Control Room for Hot Cell (5 ft of concrete)	50	< 1	None
Unshielded part of Bldg. 607	60 (min.)	400 (max.)	3/4 hr
Shielded locomotive		< 1	None

When the reactor is being transported to the Examination Area, a distance of 500 ft on each side of the 4-rail dolly track, except behind the earth embankment, will be an exclusion area. In the event of high radiation levels, personnel inside the Examination Area (Bldg. 607), may be evacuated to a position behind the earth embankment until the reactor is inside the large hot-cell room. Aside from the areas mentioned, all other areas will be considered as unlimited access areas.

c. Examination Area

(1) Contamination. The hot shop in the Examination Area is designed to handle highly radioactive and contaminated materials. Thus, surface contamination of reactor components does not present an uncontrollable hazard.

(2) Direct Radiation. The gamma radiation level through the hot shop walls, with the "hot" core inside, is calculated to be less than 1 mrem/hr.

The analysis is based on the assumption that the remains of the core will be moved into the hot shop 24 hours after the destructive test.

2. <u>Secondary Criticality</u>

Possible secondary criticality hazards during or as a result of the cleaup operation are: accumulation of a dry critical mass, accumulation of a critical solution in the drain system, and accumulation of a critical mass on the return trip to the Examination Area.

It is expected that the explosion in the final destructive test will dismantle the reactor and disperse debris over a large area. The proposed cleanup schedule will include the collection of debris outside the test pad, and collection of debris larger than a few cubic inches on the pad. After replacing the building, the test pad will be vacuumed and the remaining material washed into a drain system leading to a catch tank.

The dry cleanup of the Test Cell will be accomplished by the use of an electromechanical robot under the command and TV observation of an operator located at the robot control console in the IET control room.

The possibility of a critical configuration during the dry cleanup of fuel, moderator, and reflector either by loading a critical mass in a waste container or vacuum cleaner or by pushing masses into critical configurations on the reactor dolly has been considered. These hazards will be avoided by providing critical safe containers for the debris by limiting the size of vacuum cleaner receptacle, and by loading the debris on the reactor dolly in critical safe configurations.

Following the dry cleanup operation the remains of the reactor and the debris collected will be moved back to the Examination Area on the reactor dolly. During transportation there are possibilities of criticality due to movement of debris into critical configurations or the addition of water from rain or snow to the reactor remains and debris. These possibilities will be eliminated by securing debris to the dolly and providing a waterproof cover for the dolly.

The wet cleanup of the test pad will be accomplished by washing fine debris into the drain system. The drain system is made up of a drain trench, a 10 in. pipe, a fine mesh particulate filter, and a 15,000 gallon catch tank. Situations which have been studied as possible criticality hazards in the drain system are the clogging of the 10 in. pipe to form a cylindrical fuel-water mixture and the accumulation of fuel in the filter basket in a water bath. Cadmium has been used to line the drain trench, surround the drain pipes, and plate the particulate filter basket (see Fig. 70). Calculations on the IBM-650 using the DMM Code indicate that the drain system with this addition of poison forms a critically safe configuration for the planned cleanup operation.

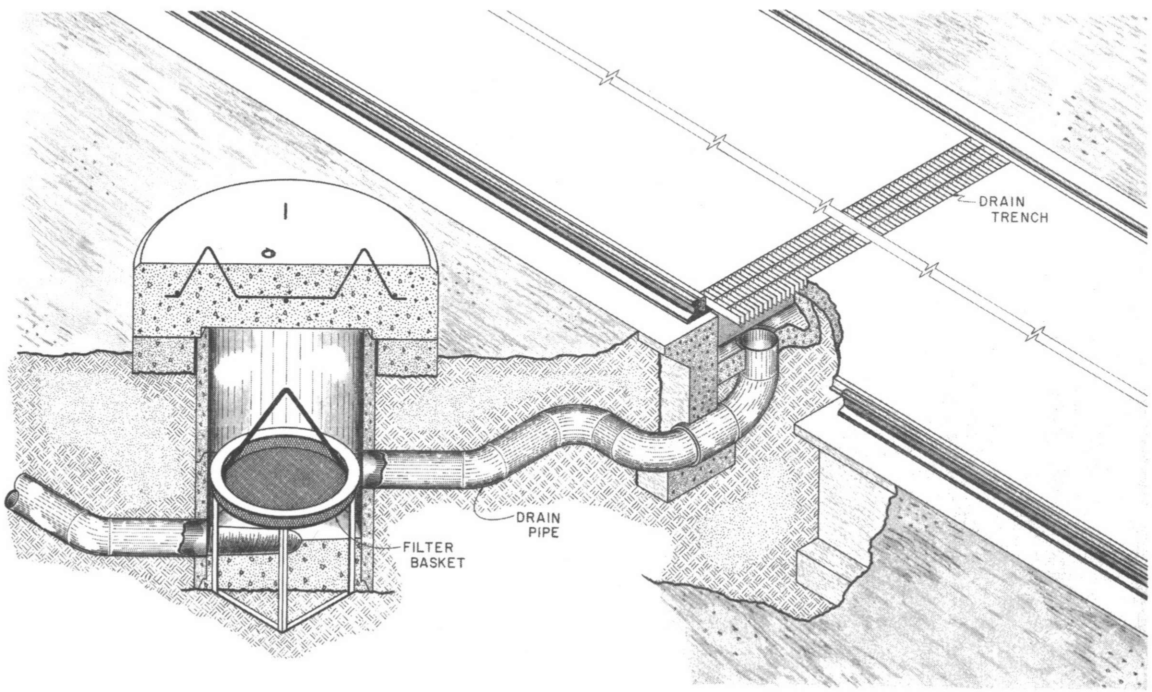


Fig. 70 - Contaminated Waste Trench and Strainer

F. Conclusions

The hazards associated with the safety test program of the SNAPTRAN 2/10A-1 reactor have been considered in this section. Some important points in the discussion are summarized below.

(1) Mechanical controls and detailed operating procedures have been established for the handling of both new and irradiated fuel during transportation, loading and storage with the intent of assuring the safety of such operations.

(2) Mechanical controls and detailed operating procedures have been established for the handling of all reflector-moderator materials in the vicinity of the reactor with the intention of assuring the safety of such operations.

(3) System design, multiple channel instrumentation, operational interlocks, automatic period and level scrams, and failure indication monitors have been incorporated and operating procedures have been established with the intention of obviating operator error or system failure and to provide backup protection in the event of such error or failure.

(4) During nuclear operation of the reactor, personnel movement into and from the test area will be controlled. The test area is defined as that area enclosed by an obstruction fence surrounding the IET facility at a distance of about 1-1/2 miles. Access to the area will be controlled at the fence gate adjacent to the Technical Support Facilities (TSF).

(5) Personnel remaining inside the test area during nuclear operation will be located in the underground, earth-shielded Control and Equipment Building. Personnel will be excluded from the Test Cell. The dose rate in the Control and Equipment Building will be 0.01 mrem/hr or less for the conditions stated in (4) above.

(6) For those tests in which there is no fission product release to the atmosphere, the estimated dose rates in the Test Cell

and in the immediate area surrounding the building require that access to the area during the first few hours after shutdown be limited. Following all tests re-entry to the test area will be subject to health physics control.

(7) For tests in which fission product release is expected, the tests will proceed under meteorological control as defined by the AEC-ID Health and Safety Division.

(8) The destructive test, which corresponds to the maximum credible incident, results in a maximum total integrated dose of 2.2 mrem to on-site personnel during the power transient. This dose is received in the IET control room. The post-test dose rate in the control room will be less than 1 mrem/hr. These calculations are based on an estimated value of 170 Mw-sec for the energy release in this test.

(9) The maximum off-site dose will occur at the site boundary ($\sim$ 6.5 miles). For the destructive test it is estimated that the total direct biological integrated dose will be less than 15 mrem. The total indirect biological integrated dose will be less than 0.350 rem to a child's throid assuming consumption of milk obtained from a cow grazing at the site boundary. These calculations are based on an estimated value of 170 Mw-sec for the energy release in this test.

It is concluded that no significant hazard exists to operating personnel or to the general public. It should also be noted that the calculations of the downwind dose for a given energy release employ conservative assumptions with respect to each of the contributing factors, hence the dose values presented overestimate the severity of the consequences.

VI. APPENDICES

A. Operating Philosophy and Test Procedures

1. Introduction

The SNAPTRAN 2/10A-1 program objectives require that certain tests be conducted in which the reactor undergoes severe nuclear excursions, including a modeling of the maximum credible incident. The operating philosophy and test procedures within the framework of the STEP organization which assure the safe conduct of these tests are discussed below.

2. Organization

a. AED Organization

The responsibility for conducting the SNAPTRAN 2/10A-1 experimental program has been assigned to the Atomic Energy Division of Phillips Petroleum Company. The four major subdivisions of the AED are Administration, Engineering, Operations, and Technical. The Reactor Projects Branch, which is a part of the Technical activity, has the responsibility for all reactor safety testing within the Division. That portion of the safety testing which involves the SNAPTRAN 2/10A-1 program will be carried out by the STEP Project. The relationship of the STEP Project to other branches of the AED is shown in Fig. 71.

b. STEP Organization

The STEP Organization is shown in Fig. 72. The organization consists of four sections: Engineering, Nuclear Test, Experiments, and Data Reduction and Analysis, each of which is in turn divided into two or more groups. The functions and responsibilities of the four sections are summarized below.

(1) Engineering Section. This section is responsible for providing engineering service to the STEP organization. These services include: assistance in planning, design, and conduct of engineering type experiments; the design of new facilities, modification

PHILLIPS PETROLEUM COMPANY

ATOMIC ENERGY DIVISION

ORGANIZATION OF THE ATOMIC ENERGY DIVISION

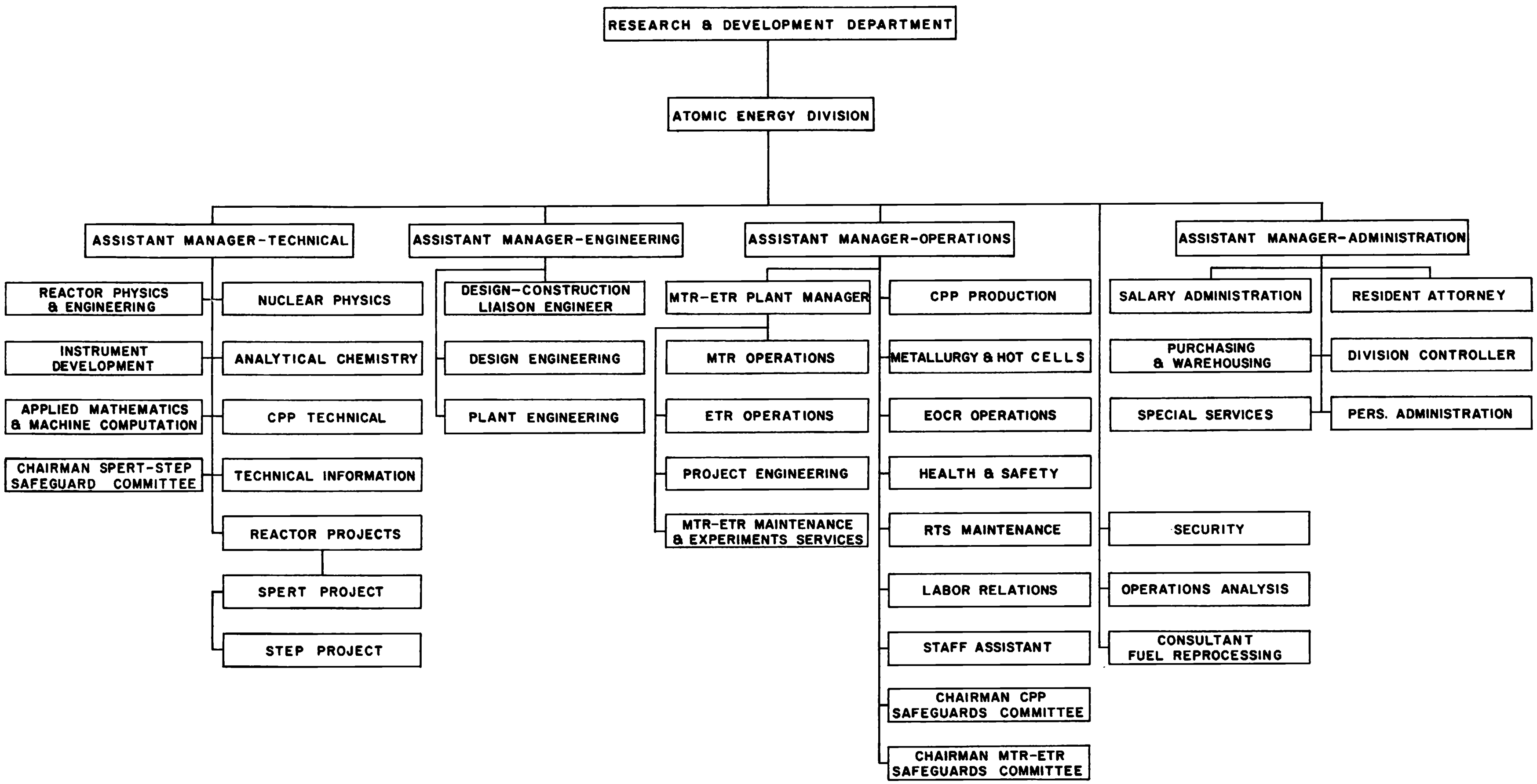


Fig. 71 - AED Organization

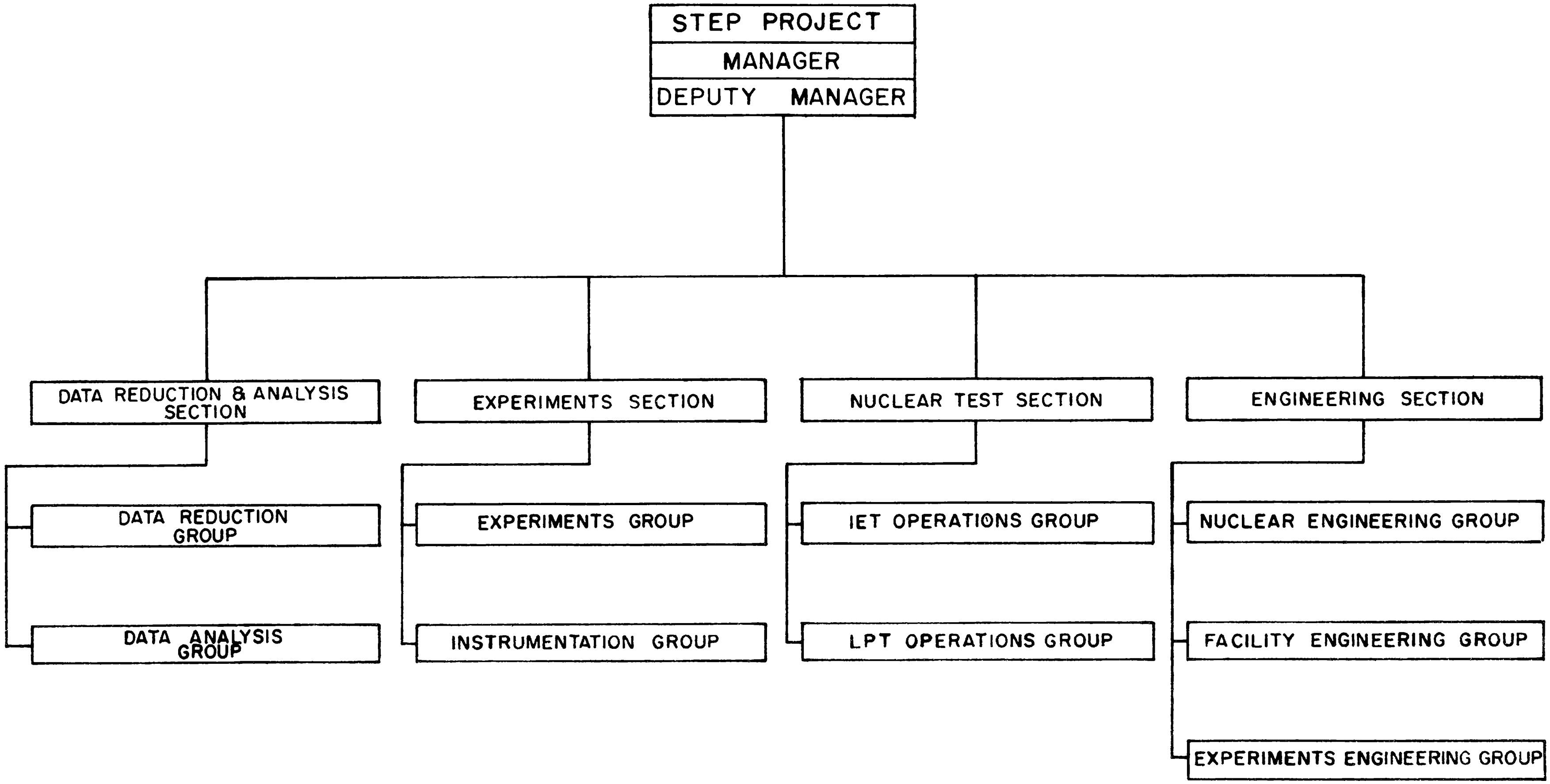


Fig. 72 - STEP Organization

of existing facilities, construction liaison, systems acceptance testing, and plant engineering; and nuclear engineering, including core and containment design, radiological evaluation, and hazards analysis.

(2) Nuclear Test Section. This section is responsible for carrying out all reactor and plant operations and for the coordination of all maintenance activities in the respective facilities.

(3) Experiments Section. This section is responsible for planning, initiation, and routine analysis of all experiments to be performed. The section is also responsible for nuclear surveillance of all operations in the IET and during transportation to and from the hot shop. In addition, the section is responsible for the design, installation, calibration, and maintenance of all operational and experimental instrumentation and data processing equipment.

(4) Data Reduction and Analysis Section. This section is responsible for the reduction, handling, and storage of experimental data. In addition, the section is responsible for the analysis and interpretation of the experimental data and for development of analytical models and calculational techniques which will assist in correlating experimental data with the physical theory.

c. Supporting Organizations

Supporting services are supplied to the STEP Project by other branches of the AED. Other branches supplying service include: Reactor Physics and Engineering, Applied Mathematics and Machine Computation, Instrument Development, Analytical, Engineering, Health and Safety, and Maintenance. In addition, a number of special committees appointed by the Division Manager are responsible for assuring the safe conduct of all operations by reviewing policies and procedures and evaluating hazards attendant to the operations. These include the following:

(1) the Safeguard Review Committee, which is responsible for determining that all operating policies and procedures are safe, current, and complete,

(2) the Nuclear Safety Committee, which is responsible for review of, for providing consultation services in, and for searching out areas in which there could be a criticality hazard, and

(3) the SPERT-STEP Safeguard Committee, which is responsible for review and approval of all reactor core loadings and control systems, operational procedures, and experimental programs from a nuclear safety viewpoint.

3. Operating Philosophy

Since the objectives of the SNAPTRAN 2/10A-1 experimental safety program necessitates operation of the reactor under conditions normally considered hazardous, administrative controls must be relied upon to minimize the probability of nuclear incidents, to insure the safety of STEP personnel and the NRTS, and to eliminate hazard to the general public.

Safety of operating personnel is assured by the requirement that all personnel in the test area be in the control and equipment building and no closer to the reactor than the control room during any nuclear operation. Safety of other than operating personnel is assured by the requirement that meteorological control be exercised during all tests in which fission product release can be reasonably expected.

Protection of the reactor system from excursions which could cause premature damage to the reactor is assured by an interlock system, by automatic period and level scrams to be used during precision control drum positioning operations, by scrams actuated by the sequence timer, or by scrams actuated at a pre-set temperature, energy level, or other parameters.

4. Planning and Approvals

The method of operation and scope of the tests to be performed on the SNAPTRAN 2/10A-1 reactor necessitates thorough planning and extensive review of the proposed procedures before approval is obtained for any test. General operating policy and procedures employed by the STEP Project are reviewed once each year by the Safeguard Review Committee.

The Experiments Section Chief is responsible for the preparation of a written Test Series Proposal (TSP). This proposal will include a statement of the objectives of the test series and its relation to the overall program, the expected results, any potential hazards to be expected, and the approximate time schedule. Technical information and assistance will be provided by Atomics International in the preparation of the TSP.

The TSP will first be reviewed in detail by the STEP senior staff. Following this review a presentation of the TSP will be made to the SPERT-STEP Safeguard Committee.

After the SPERT-STEP Safeguard Committee's approval has been obtained, the TSP will be forwarded to the Assistant Manager, Technical, for approval. Following approval by the Assistant Manager, Technical, the copies will be transmitted to Idaho Operations Office for review and approval.

Prior to conducting a test, the Experiments Section Chief will call a planning meeting. The attendance of Experiments Group Leader, Nuclear Test Section Chief and Group Leader, and any additional persons designated by the Experiments Section Chief, will be required. The TSP will be reviewed and detailed plans for performance of the tests will be made.

It is then the responsibility of the Nuclear Test Section Chief to prepare the detailed test procedures that are to be followed. He will also be responsible for informing all operating personnel of the test objectives, the detailed procedures to be followed, the expected results, and any unusual procedures which may be required as a consequence of the test results. Each person will be made aware of his individual responsibility and his working assignment for the test series.

In addition to the specific operating procedures necessary for the performance of a particular test, certain basic nuclear test procedures will apply during the entire SNAPTRAN 2/10A-1 program. These basic procedures are discussed below.

5. Nuclear Test Procedures

a. Administrative Control

Administrative control in the STEP program is outlined in the Standard Practices Manual, a written reference containing the basic rules and instructions regulating the safe and efficient operation of the STEP facilities.

In addition to the Standard Practices Manual certain specific instructions and criteria covering particular facets of the operation are contained in Operating Manuals. Each person is responsible for becoming thoroughly familiar with the instructions pertaining to his job which are described in the Standard Practices and Operating Manuals.

The STEP Nuclear Test Section is responsible for all nuclear operations including static physics measurements and kinetics tests, and for all non-nuclear operations in the IET including plant modifications, maintenance, and fuel handling. During those periods when the reactor is in transport between TSF and IET the Nuclear Test Section is also responsible for all operations involving the reactor.

b. Work Procedures with Reactor Shutdown

Reactor physics considerations dictate that several procedures be followed concerning work in the reactor building and work on or very close to the reactor. The Nuclear Test Group Leader or his designated representative, and a Health Physicist will be present in the reactor building when any other personnel are present in the building. No portable reflector-moderator material will be present in the building without the approval of the Nuclear Test Group Leader and no large amounts of reflector-moderator material will be present without the approval of the Experiments Section Chief and the Nuclear Test Section Chief. During any operations which significantly affect the reactivity of the system, a surveillant physicist from the Experiments Section will be present. No personnel will be allowed within a barrier placed at a radius of six feet from the reactor unless adequate low level neutron detection instrumentation is operating and approval of the Nuclear Test Group Leader is obtained. The placing of any material within the barrier must also have the approval of the Nuclear Test Group Leader. The reactor operator will be informed of any change in material around the reactor.

c. Preparations for Reactor Operations

Prior to reactor operation, the Nuclear Test Group will be responsible for completing check lists to verify operability of operational instrumentation, experimental instrumentation, and experimental equipment. Operational instrumentation will include, as a minimum, neutron detection from at least two pulse neutron counters, one linear power recorder, and one log power recorder.

The Nuclear Test Group will also complete check lists to verify the operability of process instrumentation and equipment. When all check lists have been completed and the Nuclear Test Group Leader has verified the completion of plant and reactor preparations, an entry will be made in the console log.

d. Routine Evacuation of the Test Area

All personnel entering the test area are required to report to the security area guard house within the test area. The guard will maintain a record of all people within the area.

Prior to any nuclear operation, the Nuclear Test Group Leader will initiate a routine evacuation of the test area in the following steps:

(1) The Reactor Operator will actuate the evacuation horn at periodic intervals for not less than twenty minutes. All those people not directly concerned with the immediate operation of the test will evacuate through the TSF area gate, notifying the security area guard. During those tests in which fission product release can reasonably be expected, a check of the area lying downwind of the reactor building between the test area and site boundary will be made in cooperation with ID Health and Safety. The Phillips Health Physics Supervisor will be notified when the downwind area has been cleared of personnel.

(2) Twenty minutes after the initial test area evacuation order, the Reactor Operator will announce over the public address system the order for all personnel inside the security fence (security area) to proceed to the control and equipment building. This will include all those people in the test cell and coupling station.

(3) The Health Physicist will leave the test cell last, checking shielding doors, doors in the coupling station, and verifying that the passageways are cleared of personnel.

(4) The Health Physicist will then ascertain the names of all personnel remaining in the control and equipment building.

(5) The Health Physicist will then leave the control and equipment building by way of the coupling station tunnel, locking the tunnel door behind him with the Health Physics key.

(6) The Health Physicist will then return through the passageways to the test cell verifying that no personnel are present.

(7) The Health Physicist will then check the security area, including all other outside buildings, to verify that all personnel have been evacuated.

(8) The Health Physicist, upon completion of the security area check, will proceed to the security area guard house and verify that the list of people in the control and equipment building checks with the record maintained by the guard. He will also verify that the test area has been cleared.

(9) The Health Physicist will then proceed to the TSF gate in the test area obstruction fence and verify with the guard at this point that no personnel are unaccounted for in the test area. He will then place a barricade with a flashing light at the obstruction fence gate.

(10) The Health Physicist will leave a copy of his personnel check list with the guard at the obstruction fence gate.

(11) The Health Physicist will then return to the security area and take the guard for this area into the control and equipment building.

(12) The Health Physicist will then receive verification from the Health Physics Supervisor that the area downwind of the reactor building between the test area and the site boundary has been cleared of personnel if fission product release is expected.

(13) The Health Physicist will report to the Nuclear Test Goup Leader in the control room and transfer the Health Physics key for the console power switch to the Reactor Operator.

e. Reactor Operation

When all pre-operational procedures have been completed, including the routine test area evacuation, the reactor is ready for startup. During operation the Nuclear Test Group Leader is the Responsible Supervisor in the control room. A Reactor Operator, an Assistant Reactor Operator and two instrument technicians must also be present in the control room. All other personnel must have the specific approval of the Responsible Supervisor to remain in the control room during the test. Permission to proceed will be given by the Nuclear Test Group Leader. Before the reactor is brought critical, checks of the scram mechanisms will be made.

As the reactor is being brought to critical, the drums will not be moved more than an amount equal to $0.50 reactivity, which is equivalent to a 6 second period, without stopping drum movement long enough to ascertain the criticality state of the reactor. If any indication of a hazardous or potentially hazardous condition exists, the reactor will be scrammed and the condition will be investigated and corrected. It is the responsibility of each member of the STEP organization to scram the reactor if he believes that a potentially hazardous condition exists.

In the event of a scram from malfunction or indication of hazardous condition, the Nuclear Test Section Chief must approve the resumption of nuclear operation.

f. Re-Entry Procedures

When instrumentation indicates that the reactor is shut down and, to the best judgment of the Nuclear Test Group Leader, that the reactor is subcritical and no foreseeable events will cause it to go critical, re-entry operations will be permitted.

Re-entry into the test cell will be under the direction of the Nuclear Test Group Leader with the advice of the Health Physics Supervisor. The following equipment will be available at all times to aid the Health Physicist in evaluating re-entry hazards:

(1) The remote area monitoring system will consist of eleven ion-chambers which will continuously record the intensity of the radiation field at their respective locations. The locations and sensitivity ranges of these ion-chambers are as follows:

(a) Railroad flatcar --------------- 100 mr/hr to 1000 r/hr
(b) The furnace enclosure of the test cell ------------------ 0.1 mr/hr to 100 r/hr
(c) Personnel door to test cell ---- 0.1 mr/hr to 100 r/hr
(d) The coupling station ----------- 0.01 mr/hr to 10 r/hr
(e) Service room below coupling station ------------------------ 0.01 mr/hr to 10 r/hr
(f) Control room escape tunnel ----- 0.1 mr/hr to 100 mr/hr
(g) Control room ------------------- 0.1 mr/hr to 100 mr/hr
(h) Data reduction room ------------ 0.1 mr/hr to 100 mr/hr
(i) Equipment room ----------------- 0.1 mr/hr to 100 mr/hr
(j) Vehicle tunnel entrance -------- 10 mr/hr to 10 r/hr
(k) Entrance to stairs leading from the underground parking area --------------------------- 10 mr/hr to 10 r/hr

The ratemeters and recorders for these ion-chambers are located in the fire equipment room of the IET.

(2) One constant air monitor will be located in the IET equipment room near the filters of the building ventilation system. This "CAM" can be quickly modified to sample air from the filter bank chamber.

The other constant air monitor will be located below the coupling station and will be sampling air from the test cell during all tests. At other times it will be located in the test cell building.

(3) One portal monitor will be located at the end of the tunnel to the test cell near the Health Physics office.

(4) A laboratory counter will be located in the Health Physics office for analyzing smears of alpha-particle and beta-particle activity.

(5) A scintillation type well-counter will be located in the Health Physics office for counting such items as water samples in which a low background count rate is necessary.

(6) A stack monitor readout panel will be located in the fire equipment room. This monitoring system samples air at the 80 foot level of the stack. Measurements are made of gross particulate activity and gross gaseous activity, which includes the iodines collected by a separate charcoal trap sampling system. The collection and detection equipment for this system is located in Building 713 below the stack.

(7) Five portable G-M type radiation monitors (0 to 20 mr/hr).

(8) Five low-range Juno portable radiation monitors (0 to 5 r/hr).

(9) Five high-range Juno radiation monitors (0 to 25 r/hr).

(10) Two Jordan portable radiation monitors (0 to 500 r/hr).

(11) Seven "Cutie-Pie" portable radiation monitors (0 to 2.5 r/hr).

(12) Three thermal neutron portable radiation monitors.

(13) One fast neutron portable radiation monitor.

(14) Fifty pencil dosimeters (0 to 250 mr).

(15) Five pencil dosimeters (0 to 5 r).

(16) Three alpha-particle portable detectors.

(17) Three nuclear accident dosimeters.

The extensive site-monitoring program for the SNAP tests will also provide data for deciding upon the proper re-entry procedure. This program is discussed in Section III-F.

The precautions to be observed prior to entry into the test cell will depend primarily upon four factors:

(1) a determination by the Nuclear Test Section Group Leader that the reactor is critically safe,

(2) a visual observance of the test cell and reactor by means of the periscopes and television cameras,

(3) the data measured and recorded by radiation monitoring instruments and beryllium sampling equipment, and

(4) a pre-entry survey of the area by the Health Physics Section.

Following each test (static, kinetic, and destructive), the Health Physics Supervisor at the IET will evaluate the radiological hazards existing based on information received in parts 3 and 4 above. A decision will be made on re-entry time limits and protective clothing, respiratory protection and personal dosimetery requirements. After consultation with the Nuclear Test Section Group Leader to insure that the reactor is critically safe, re-entry will be allowed, but only when accompanied by a Health Physicist. Phillips Petroleum Company Radiation Protection Guides will be followed at all times.

When the radiation and safety hazards have been evaluated and it has been resolved that personnel may enter the area, the first re-entry team will proceed toward the test cell. This team will consist of two Health Physicists and one person from the Nuclear Test Section. The team will be clothed in plastic suits, with self-contained breathing apparatus. The Health Physicists will be equipped with portable radiation survey instruments and a portable radio unit. As the team approaches the test cell they will radio the radiation readings to the control room. The radiation reading in rooms and passageways leading to the test cell will be recorded on a map in the control room. Smears will also be taken of these areas, and the beryllium filters below the coupling station will be recovered if exposure time permits. The team will then remove their plastic protective clothing and return to the Health Physics office for a personal survey.

The roving manipulator will be used to make a radiation survey of the test cell and surrounding outside area. It will use one of the removable Tracerlab ion chambers to survey the area near the flatcar.

The data obtained by the first re-entry team and the roving manipulator will then be used to determine when and how cleanup procedures may be undertaken. A second re-entry team consisting of two Health Physicists will be stationed at the TSF test area gate. After this team has been notified by radio that the reactor is critically safe, they will survey the road leading to the IET for contamination. After reporting their results, this team will operate as one of the teams of the site monitoring group and will collect samples on the south side of the monitoring grid.

g. Transfer of the Test Package with Fuel in the Core

When the test package and test dolly have been prepared for transfer, including drums locked out in their scrammed position, if possible, the Nuclear Test Group Leader will notify the Section Chief in charge of the hot shop that the reactor is ready for movement. At that time he will also inform the Section Chief of the radiation levels surrounding the test package. The Nuclear Test Group Leader will then initiate a routine evacuation of the test area if he feels a potentially hazaradous situation could exist during transfer of the test package.

During transfer of the package the locomotive will contain the locomotive operator, a health physicist, and a surveillant physicist from the Experiments Section. The Nuclear Test Group Leader will then inform the dispatcher that the package is ready for movement and if partial evacuation of the TSF area has been necessary, the dispatcher will notify the Nuclear Test Group Leader that such evacuation has been effected.

Once the test dolly has reached the examination area, the reactor test package responsibility will be assumed by the Metallurgy and Hot

Cells Branch. However, prior to the commencement of any operation in the hot cell, a qualified individual or group will be assigned the full responsibility for nuclear safety.

h. Examination Area Operations

While in the examination area, any operations on or near the test package, when there is fuel in the reactor, will be supervised by the qualified individual or group of individuals assigned the responsibility for nuclear safety. These operations include loading or unloading fuel from the reactor vessel, handling or placing reflector and/or moderating materials within six feet of the reactor, and removal from or addition of any component to the test package.

During any operation on the test package, including loading or unloading of fuel from the reactor vessel, at least two neutron-pulse counting systems will be in operation and the detectors will be placed adjacent to the reactor and care will be taken to prevent the insertion of shielding material between the reactor and the detectors. During all operations which could conceivably change the reactivity of the system, a neutron source will be placed in close proximity to the reactor vessel. A jig will be used to hold the reactor vessel, the neutron-pulse counting chambers, and the source when any operation involving fuel is performed on the reactor vessel while not mounted in the test package.

A full-scale model of the SNAPTRAN 2/10A-1 reactor test package has been fabricated for use in developing techniques and procedures to be followed during remote handling of the test package or any of its components.

i. Health Physics Support

The duties of the Health Physics support staff are as follows:

(1) Acquisition of necessary radiation monitoring and measuring instruments. Coordination of the installation and maintenance of this equipment plus the performance of a routine calibration procedure.

(2) Measurement, control and reporting to the AED of radiological discharges to the environs.

(3) Maintenance of exposure records of STEP personnel and notification of project supervision concerning significant exposures.

(4) Provision of protective materials and decontamination equipment.

(5) Supervision of the disposal of radioactive wastes from the STEP areas.

(6) Control of on-site and off-site radioactive shipments from the STEP areas.

(7) Control of radioactive sources within the STEP areas.

(8) Responsibility for the monitoring of beryllium contamination in cooperation with the Phillips' Industrial Hygienist.

(9) Assist in the evacuation of personnel from STEP areas when a serious hazard develops. Determine when re-entry is safe, and what precautions, if any, must be taken upon returning to the evacuated area.

(10) Consultation with STEP supervisors and engineers concerning existing or potential radiological hazards associated with reactor handling and testing.

Additional Health Physicists will be temporarily assigned to the STEP Project for the destructive test. With the help of these men, the regular STEP Health Physics staff will perform the following duties:

(1) Design and coordinate the installation of a site-monitoring grid to obtain data on the release of fission products from the destructive test. Determine which instruments will provide the most reliable and informative data.

(2) Administer radiological and physical safety control over all participating personnel who will be on or near the test grid. In conjunction with the AEC aerial monitoring team, see that the entire test area is cleared of personnel prior to the destructive test.

(3) Activate grid equipment prior to the destructive test, recover samples after the test, and deliver the samples to proper laboratories for analysis.

(4) Establish a Field Access Control Center for assuring that the pre-release setup of equipment has been completed, for clearing the area, for dispatching sample recovery teams, for checking participants in and out of the sampling sector, for decontaminating personnel, and for establishing radio contact with all participants.

(5) Determine when re-entry into the immediate reactor area is safe, and what protective apparel and precautions must be used.

B. Emergency Action Plans

1. Phillips Petroleum Company Basic Plan

a. Emergency Action Planning Committee

This committee is responsible for formulation and at least a semiannual review of plans dealing with emergencies arising within or affecting areas operated by Phillips Petroleum Company (PPCo), plans for assistance to other NRTS plants in emergencies, and plans dealing with national emergencies. This responsibility includes the procurement of emergency equipment, the organization and training of emergency teams, and the education of individual employees.

b. Plant Emergency Action Groups

Under direction of the PPCo Emergency Action Planning Committee, this group is responsible for:

(1) preparation and revision of detailed emergency plans for specific areas,

(2) maintenance and control of emergency equipment and recommendations concerning acquisition and specifications of additional emergency equipment needed, and

(3) establishing and assuring training of emergency teams and wardens, and providing educational material for and training of assigned plant personnel.

c. PPCo Control Group

Acting under the senior supervisor present, this group, after being duly convened, shall be responsible for the executive direction of all activities during emergencies at PPCo-operated plant areas. In event of national emergencies, or emergencies at the NRTS which envelope or endanger other AEC installations, this group will co-ordinate PPCo activities under the direction of the ID Control Group. The senior Operations supervisor in the CPP and MTR area, the senior

Maintenance supervisor in CFA, senior Engineering supervisor at STEP, and the senior Technical supervisor at SPERT, who will become part of this control group, are designated as the individuals in charge, in their respective areas, of Phillips Petroleum Company activities in event of emergency, until relieved by higher authority. These supervisors are responsible, in their respective areas, for determining immediate action necessary, such as evacuation, re-entry, etc., and for immediate notification to the PPCo Control Group and the ID Control Group (through the CFA Security Dispatcher, Ext. 2345).

d. <u>All PPCo Supervisors</u>

All supervisors are responsible for: (1) being familiar with the provisions outlined in these plans, (2) instructing employees in these provisions, (3) accounting for employees upon evacuation, (4) rendering all assistance possible as directed by the PPCo Control Group, (5) maintaining controls to prevent sabotage and/or damage of vital equipment, and (6) providing emergency assistance to non-Phillips Petroleum Company areas under conditions listed below:

(a) Immediate emergency assistance to save lives or attempt to save lives of personnel may be made without prior approval providing the Project Manager, the Deputy Manager, and the respective Assistant Manager are notified as promptly as possible.

(b) Other assistance will require prior approval of the Project Manager, the Deputy Manager, or respective Assistant Manager (excluding Transportation Section which will dispatch buses during regular work hours for plant evacuation, then notify one of the above individuals).

e. <u>Branch and Special Group Assignments</u>

(1) <u>Wardens</u>. As designated by supervision to the Plant Emergency Action groups.

Wardens will be responsible for knowing all details of the emergency plans and possible hazards to be avoided in evacuation of their areas. During an evacuation, wardens are responsible for determining, to the greatest extent possible, that all personnel have left their respective area. If possible, without jeopardizing personal safety, they should assure that appropriate operating equipment is shut down and classified documents and safes are secured. During off-shift hours the warden's responsibility will be assumed by the senior supervisor on duty or his designated alternate.

(2) Emergency and Special Re-entry Teams. Teams appointed by the Plant Emergency Action Group are to be continually prepared to function as directed by the Control Group. During an evacuation of the IET, they will assemble at the IET turnaround room or the nearest control point to the area. If directed, they also shall be available for emergency assistance at other plant areas.

(3) Health and Safety Branch. The Safety Engineer in each area will serve as chairman of the Plant Emergency Action Group. The Branch Manager is a member of the PPCo Emergency Action Planning Committee. If requested by ID and approved by the PPCo Health Physics, personnel will be mobilized and deployed to assist ID Health Physics.

(4) Security Branch. Security Branch Manager is designated Chairman, PPCo Emergency Action Planning Committee.

Guards at main entry points are responsible for:

(1) relaying emergency notification received by radio or telephone,

(2) opening main gates to permit exit of employees,

(3) establishing roadblocks to prevent unauthorized re-entry, and

(4) assisting ID Security with roadblock and/or traffic control.

The PPCo Guard Supervisor will assume responsibility for PPCo activities in CFA during off-shifts until relieved by a member of the Control Group.

(5) CFA Maintenance. The Transportation Section is responsible for providing emergency transportation for specific NRTS areas during regular working hours, and for mobilizing and providing transportation as directed. They will maintain a current record of buses needed for plant areas during regular work hours in order to facilitate dispatching. This section will provide maintenance for permanently assigned emergency buses at certain NRTS plant areas.

The TAN Maintenance Branch will provide available heavy equipment, operators, and incidental drivers as needed.

The Communications and Radio and Alarm Shop sections will provide liaison with Mountain States Telephone and Telegraph Company for additional communications needed, and provide emergency equipment, installation, and repairs.

f. Responsibility for Action and Notification

Upon determining that an emergency condition exists in a plant area requiring evacuation, the Supervisor-in-Charge or his designated alternate will:

(1) operate or give instructions to operate siren warning equipment (in CFA call Ext. 2345, at TAN call Ext. 6258),

(2) order crash shutdown of equipment if deemed necessary,

(3) notify NRTS Radio Communications Control Center (Ext. 2345), and furnish all known details as to the scope and location of the situation,

(4) notify the Transportation Dispatcher (Ext. 2313) if deemed necessary during regular working hours, and

(5) in emergencies requiring partial or total evacuation and/or a crash shutdown of equipment, he shall notify his immediate supervisor or the next higher supervisor. In any event the Project Manager (R. L. Doan) or the Deputy Manager (J. P. Lyon) will be called. (If deemed necessary the PPCo Control Group will be contacted).

g. Relocation Centers

Relocation centers for Control Group activities in event of a site-wide or national emergency or local TAN emergencies will be as designated by the ID Control Group. The relocation center and control point for emergency operation when the IET is affected will be at the nearest location noted below to the IET area as permitted by radiological hazards:

(1) main guard gate for Building 607, or

(2) turnaround room at IET.

h. Emergency Situations

Plant-incurred incidents, natural disasters, or enemy action may endanger the lives and safety of employees or property at NRTS. In making preparations, requesting assistance, making decisions, and in training exercises the following factors must be considered:

(1) Definitions of Plant Incidents

Class I. A hazardous incident occurring in any single area involving considerable property or personnel and which can be controlled by personnel in the area. Standby assistance of the NRTS Fire Department, medical personnel, Health and Safety, or other emergency assistance groups may be requested. Evacuation of part of the affected area for a short period of time is to be considered a Class I incident.

Class II. A hazardous incident occurring in a single area and of sufficient proportions to require the active assistance of emergency groups such as the NRTS Fire Department, medical personnel, Health and Safety, or other persons and emergency equipment not routinely assigned to the affected area. An incident which leads to the evacuation of all of the affected area or which may spread to other areas is to be considered a Class II incident.

Class III. A hazardous incident which threatens or endangers life, health, safety, and/or property in two or more areas, the entire NRTS, or communities or areas of residence outside the NRTS boundaries.

(2) Natural Disasters. Acts of nature involving flood, earthquakes, or the like could result in extensive damage or threat to safety of employees.

(3) Enemy Action. In considering the scope of possible national emergencies and enemy action it must be assumed that:

(1) strategic warning may not be received and tactical range warning lead-time may be short or nonexistent,

(2) enemy attack may involve biological or chemical agents, nuclear weapons in the multimegaton range, or any combination of the three,

(3) sabotage may occur, and

(4) any part or all of the NRTS may be the target of enemy action.

Decision to evacuate or take shelter, in event of actual enemy attack or in event tactical warning lead-time does not allow decisions by control groups, will be the responsibility of the Supervisor-in-Charge in each area. It is to be assumed that if strategic or tactical lead-time permits, total evacuation of the site will be effected. Evacuation may be ordered by the Manager, Idaho Operations Office, USAEC.

(4) CONELRAD Procedures. CONELRAD procedures, established for the alerting and control of all radio stations (including SECODE stations) operated at the NRTS by the AEC or Contractors, will be implemented by Communications Control Center at CFA.

(a) Stations at the NRTS will, upon receipt of an alert, confine transmissions to those required by emergency conditions affecting the safety of life or property.

(b) During any emergency transmission, station identification or other information identifying the station or its location will not be transmitted. All stations will use tactical call signs.

i. Relationship with AEC and Other Contractors

(1) ID Control Group. This group is appointed by the Manager, ID-AEC, to advise and assist in executive direction during periods of emergencies at NRTS. Direction of emergency action at an area at the NRTS may revert to this group if severity of the situation warrants. It is assumed that this would only occur when responsible authorities at that area are either incapacitated or unavailable for a prolonged period of time.

(2) Assistance to Other Areas. Emergency assistance to areas at the NRTS, other than those under PPCo operation, by employees and emergency teams will be provided upon request after receipt of approval by PPCo Project Manager, Deputy Manager, or an Assistant Manager; however, in event assistance must be rendered immediately to save or attempt to save lives of personnel, the above supervisors are to be notified at the earliest time possible.

(3) Civil Defense. In event of threatened or actual enemy action, decisions for actions at the NRTS will be based on information received from Federal and State Civil Defense agencies through the AEC-ID. If evacuation to employee residences is accomplished, further survival efforts are the responsibility of the individual and the local Civil Defense organization. In the interest of safety of all employees, PPCo encourages participation in home survival planning as outlined in Government publications.

j. Test Exercises

(1) The Plant Area Emergency Action Planning Groups are responsible for arranging and executing a test exercise every six months.

(2) Notification of these tests must be made to the PPCo Security Branch Manager in order that organizations concerned within AEC-ID may be advised.

(3) Evacuation alarm systems should be tested at intervals not to exceed two months. Notification of these tests will be made to the CFA Communications Dispatcher (Ext. 2345), and the CFA Fire Department (Ext. 2212). Results of these tests should be reported to the Chairman, Plant Emergency Action Planning Group for further action such as repair or modification.

2. <u>STEP Emergency Action Plan - IET Facility</u>

a. <u>Declaration of Emergency</u>

(1) Authority to declare an emergency is delegated to:

Reactor Projects Branch Manager

STEP Project Manager

STEP Engineering Section Chief

Nuclear Test Section Chief

Responsible Supervisor in control room during reactor operation

Responsible Health Physicist assigned to the reactor area.

(2) In the event that any employee believes an emergency has occurred, he shall immediately notify one of the aforementioned persons delegated to declare an emergency and will, until relieved by higher supervision, commence notification of the following, making use of the communication system:

Senior Supervisor present at IET control center

STEP Health Physics office

STEP reactor control room

All personnel in STEP Building 602

Guard Post 701 and IET Guard Post (Guard Post 701 will notify other TAN facility contractors)

If an employee is relieved by higher supervision, the relieving supervisor will commence notification of the above named persons.

b. <u>Responsibilities</u>

(1) The Senior Supervisor present at the IET control room will be responsible for:

(a) reporting to the person who observed the event or to the person who declared the emergency and <u>then taking charge of all activities</u>,

(b) determining if any personnel were in the area of the occurrence,

(c) determining if any present danger exists to personnel or property and the potential danger that may arise as a consequence of the event,

(d) determining the severity of the situation and if partial or complete evacuation is necessary,

(e) ordering crash shutdown of all operating equipment, if necessary,

(f) notifying the NRTS Radio Communications Control Center (Ext. 2345), and furnishing the following information if deemed necessary:

(1) Type of occurrence (fire, explosion, radiation release, etc.). Estimate of the class of emergency. (Ref. Basic Plan-Section B)

(2) Location (area, building number, and location within building).

(3) Employee's name reporting the situation and where the AED Control Group will convene.

(g) notifying the Transportation Dispatcher if deemed necessary (Ext. 2313).

(h) notifying the first member of the AED Control Group who can be reached. The first person contacted is responsible for contacting all of the other members of the Control Group, informing them of the situation and instructing them as to where the group will convene. In notifying the first member of the AED Control Group, STEP-IET Senior Supervisor should make it clearly understood that he is depending on this member to contact the other members of the Group specified in the Phillips Petroleum Company Emergency Action Plans.

(i) assembly of STEP Senior Staff and secretary,

(j) assigning a person to the reactor control room to maintain communication with personnel in the reactor area,

(k) recognition of senior person in charge of the reactor area,

(l) determining the location and identity of all persons in the IET area,

(m) disposition of personnel involved, the route to be followed and the terminal point in the case of complete or partial evacuation of reactor area. Disposition of personnel also includes such special assignments of personnel (rescue, etc.) as may be dictated by circumstances,

(n) insuring surveillance of meteorological and radiation conditions in the IET control room area, and

(o) documentation of events and ensuing activities through photographic, sound recording, or other means.

(2) The responsible supervisor in the control room has the responsibility for:

(a) safe shutdown of the reactor as rapidly as possible,

(b) notifying all personnel in the reactor area of the emergency and determining the senior person in the reactor area who will be in charge in that area until relieved by higher supervision,

(c) providing at least one person in the control room to maintain communication with the reactor area and with the senior supervisor,

(d) use of all monitoring instrumentation and recorders available in the control room.

(3) The senior person in the IET reactor area will be responsible for:

(a) directing the evacuation of the reactor area if the incident results in an evacuation signal or order,

(b) assembling all personnel in the reactor area within the reactor building and holding all personnel at this location until instructed by higher supervision to evacuate the area or until radiation levels in the reactor area warrant evacuation.

(c) safe shutdown of the plant as rapidly as possible and in a manner which will minimize hazard to the plant in case evacuation is ordered,

(d) monitoring radiation levels at the reactor building and ordering evacuation of the area if the levels warrant it,

(e) removal of portable radiation monitoring equipment from the reactor area when evacuation takes place,

(f) placing in service any radiation detectors and recorders in the reactor area with sufficient supply of paper and ink,

(g) providing sufficient transportation to evacuate reactor area, and

(h) specifying evacuation route and terminal point when the evacuation order is initiated at the reactor area.

(4) The Security Guard on duty at Post 701 or IET Guard Post has the responsibility for:

(a) allowing no one to enter or leave the STEP area unless on official business related to the emergency,

(b) maintaining a record of all entries and departures to or from reactor areas and the control center following notification of an emergency, and

(c) providing on request information on the identity and total number of personnel in the STEP area.

Every individual in the STEP-TSF area not specifically assigned to an emergency activity is to proceed to his desk or normal work location within the TSF area. The use of telephones for calls not directly the business of the emergency is absolutely forbidden.

3. STEP-IET Evacuation Horn System

a. Operational Information

(1) The "all evacuate" signals are on the ADT supervised circuit and automatically alarm the AEC Fire Department at TAN. The ADT alarm wheel which energizes the alarm circuit is manually spring wound and must be reset after each alarm. Resetting is the responsibility of the AEC Fire Department.

(2) All evacuation switches are of the non-return type and will continue actuation until thrown to the "off" position. Therefore, horns can be turned off only at the actuating station. This will require making a re-entry following evacuation with the evacuation horn possibly still sounding.

(3) It should be emphasized that an "alert" is not necessarily followed by an "evacuation"; neither is an "evacuation" necessarily preceded by an "alert".

(4) Any person who inadvertently actuates a horn must immediately effect an "all clear" signal and inform the control room so that an "all clear" announcement can be made.

b. Alert

(1) The alert signal is an approximate one-minute continuous sounding of the horn with non-varying intensity.

(2) Upon receipt of an alert, employees are to take cover inside a building and if deemed necessary, take shelter in specific parts of the building within areas as directed by supervision.

(3) All operating equipment is to be shut down and personnel will safeguard classified documents in their possession in preparation for evacuation.

c. Evacuation of IET Area

(1) The evacuation signal is an oscillating horn at 1 second on and 1 second off repeated for three minutes.

(2) Upon receipt of an evacuation signal but no other information, all employees are to proceed to the turnaround room and enter the evacuation vehicles as directed.

(3) Personnel bringing vehicles from the reactor area following an "all evacuate" signal will take the vehicle out the guard gate, or automobile ramp, where further instructions will be given. Personnel evacuating from the reactor area will remain with the reactor vehicles.

d. All Clear

(1) The all clear signal is three (3), fifteen (15) second continuous horn signals separated by five (5) second intervals of silence. The "all clear" will be sounded by manual operation of the siren control on the "alert" position.

(2) Upon receipt of an "all clear" following an alert, employees are to resume normal assignments.

(3) Upon receipt of an "all clear" following evacuation, re-entry to the reactor areas will be made in the usual manner.

e. System Tests

(1) The STEP-IET evacuation alarm system should be tested at intervals not to exceed two months in co-operation with security personnel. Notification of these tests will be made to the CFA Communications Dispatcher (Ext. 2345), and the CFA Fire Department (Ext 2212). Results of these tests will be reported to the Chairman, STEP Empergency Action Planning Group for further action, such as repair or modification.

(2) Prior to a nuclear destructive test, there will be a practice evacuation through the escape hatch to the locomotive. This test evacuation shall be under the direction of the Nuclear Test Section Chief.

(3) The STEP-IET Senior Engineer is responsible for requesting Security to test the operation of the evacuation alarm system. Prior to performing the test, an announcement will be made stating,

"This is a test of the evacuation alarm system. This is a test." The Nuclear Test Section Chief must also be assured that all personnel in the control area are aware of the test.

(4) The "all clear" will follow every test and will be acknowledged by the reactor area via its control room.

f. Evacuation Transportation

(1) A minimum of two 6-passenger vehicles will be stationed in the turnaround room at IET during a nuclear test for emergency evacuation. The conditions stated under Fighting Fires - item 3 shall apply.

(2) The locomotive will be available for emergency evacuation from IET through the underground hatchway during a nuclear test.

(3) At the TSF area there will be three evacuation buses available for the evacuation of personnel from the hot shop, administration area, and those from the locomotive.

g. Emergency Transportation Drivers

(1) A minimum of four (4) personnel working in the IET building during tests shall be assigned to drive the vehicles from the turnaround room and operate the locomotive. Those that operate the locomotive shall operate the locomotive once each month for refresher training. This training will be directed by the Engineering Section Chief, or his designated alternate.

h. Wardens

(1) Wardens are responsible for knowing thoroughly all details of the STEP evacuation plan. When an evacuation alarm is sounded, they are responsible for supplying such checking and direction as may be required to see that the evacuation in their particular area proceeds smoothly and in accordance with provisions of the plan. Before leaving their assigned area, the wardens will check to see that all personnel are evacuated, that appropriate operating equipment is shut down, and that classified documents and safes are properly safeguarded,

if possible, without jeopardizing their own safety. In the event that the assigned warden is not in his area when an evacuation alarm sounds, the supervisor or senior employee on duty, or alternate designated by him, will assume the warden duties. Evacuation wardens will be designated by the STEP Engineering Section Chief for the various STEP areas and the names of the wardens will be plainly posted in each area.

(2) The foregoing warden procedure applies only to regular daytime working hours. During off shifts it will be the responsibility of the senior supervisor present to direct evacuation of the STEP area and provide for the protection of classified materials and other equipment as required in this evacuation plan.

4. <u>STEP-IET Emergency Re-Entry Plan</u>

a. <u>Authority and Responsibility</u>

The authority and responsibility for assignment of personnel to the emergency re-entry teams shall reside with the STEP Project Manager. Makeup of the re-entry team shall include the following as a minimum:

(1) a senior STEP supervisor who acts as planning director,

(2) the Nuclear Test Group Leader,

(3) one STEP Health Physicist,

(4) one surveillant physicist, and

(5) one reactor technician.

The existing Fire Brigade will supplement the re-entry team to handle fire fighting and emergency first aid as necessary.

The planning director of the emergency re-entry team will be responsible for providing the team with all available information necessary to make the re-entry. This includes plant photographs, drawings, and any other pertinent information.

b. <u>Training</u>

The STEP Safety Engineer will be responsible to the following for special annual training in the proper use of emergency equipment and other related information concerning re-entry and rescue:

(1) the Nuclear Test Section Group Leader in charge of reactor facility,

(2) all STEP Health Physics personnel,

(3) a minimum of four assigned "surveillant physicists", and

(4) all reactor technicians.

c. <u>Emergency Equipment Available</u>

(1) <u>Protective Clothing</u>

(a) 100 coveralls

(b) 100 cloth shoe covers

(c) 50 cotton gloves

(d) 25 latex gloves

(e) 24 rubber gloves

(f) 20 cotton caps

(g) 10 lab coats

(h) 20 rubber boots

(i) 5 plastic suits and head covers

(j) 12 Comfo respirators

(k) 3 - 30 minute press demand Scott Air-Packs

(l) 12 hard hats

(2) <u>First Aid</u>

(a) 3 first aid kits

(b) 3 burn kits

(c) 1 decontamination kit

(d) 1 MSA cabinet

(e) 3 stretchers with blankets

(3) Fire Protection (NaK)

(a) 2 Met-L-X 30# Ansul extinguishers

(b) 1 Met-L-X 300# unit mounted on test dolly

(c) 2 shovels

(d) 2 full length leather coats

(e) 2 leather leggings

(f) 2 elbow length leather gloves

(g) 2 full face shields

(h) 2 - 50# cans Met-L-X powder

(4) Radiation Instruments

(a) Ample supply portable radiation instruments

(b) 3 Nuclear accident dosimeters

(5) Facility Drawings. The STEP Design Engineering Section Chief is responsible for assuring maintenance of current drawings of each reactor area for emergency re-entry use.

5. Fire Safety and Prevention

a. General Rules

(1) Personnel shall not evacuate a building in which there is a fire unless there is a definite fire hazard in their particular area.

(2) Persons not engaged in fighting the fire shall remain away from the area.

(3) Employees should be familiar with the location of fire alarm boxes, the method of operation of such boxes and the meaning of the fire alarm signals for the STEP area.

(4) Employees should become familiar with the location and operation of fire extinguishers in their general working area.

(5) Near each fire extinguisher is a marker indicating the extinguisher's limitations. Fire extinguishers must be used in accordance with these limitations.

(6) A fire extinguisher which has been partly or wholly discharged, regardless of how it was discharged, must be reported to the Nuclear Test Group Leader.

b. Reporting Fires

(1) Fire may be reported by actuating the nearest Fire Alarm Box (indicated by the red light) or, as a second choice, by telephoning 2211 or 6213 (give location of the building, location of the fire within the building, and the type of fire - electrical, wood, etc.). The individual actuating the alarm will remain or have someone remain at the alarm box, or nearby, to direct the first fire fighting crew to the fire. It shall also be his responsibility to inform the STEP Project Manager of the details of the fire as soon as possible.

(2) Instructions on filling out formal reports of a fire may be found in the Phillips AED Administrative Instruction Bulletin, AIB-6.06.

(a) At all times when personnel are working in a reactor area, a vehicle must be available to these personnel for evacuation. If the only vehicle in a reactor area must be used to transport individuals to the TSF area, and personnel are to remain in the reactor area, the vehicle driver must not leave the vehicle and must return it immediately to the reactor area.

c. Fighting Fires

(1) Every employee is expected to fight a fire which develops in his work area to prevent its spreading and to control it if possible.

(2) If a fire should occur during normal hours, the area supervisor, plant engineer, and all available Fire Brigade members shall report to the scene of the fire immediately. The Nuclear Test

Section Group Leader at the reactor building will be responsible for the safe shutdown of equipment if, in his opinion, shutdown is required. He shall also be responsible for the safeguarding of all equipment, fuel, etc., which may be in danger of damage by the fire. The senior supervisor in attendance will be in charge of the Fire Brigade and he will coordinate activity between the Fire Department and the Brigade.

(3) The Brigade members will have first priority of available government transportation in the STEP areas in event of a fire.

(4) There is a possibility of radioactive contamination in areas affected by fire; therefore, it is necessary to restrict the entry of the Fire Brigade and the AEC Fire Department during normal working hours unless they are accompanied by or have permission of a Health Physicist.

(5) Should a fire occur in or near any fuel storage vault, extreme care must be taken to assure that: (a) no water is used within the vault or in such a manner that the vault may become flooded, and (b) proper safeguarding of fire fighting or other personnel in the area is accomplished to prevent inhalation of radioactive fumes.

d. <u>Fire Prevention</u>

(1) Good housekeeping shall be required in all areas as a primary means of fire prevention.

(2) Oily and greasy rags shall be disposed of in appropriately marked covered metal waste cans. At no time will volumes of waste combustibles be left in work areas over night.

(3) Burning of rubbish is not permitted in the STEP areas.

(4) Portable heating devices including hot plates, heaters, and other small appliances should be placed so that they are protected from or located a suitable distance from wood or other combusible surfaces. Utmost caution should be employed to make certain no such appliances are left on overnight.

(5) Instruments which are left on overnight shall be tagged with full instructions for instrument shutoff. This will assist the Security Guard in case the instrument should catch fire.

e. Sprinkler System

Should a sprinkler head be broken or fail when there is no fire, the Nuclear Test Section Group Leader should be immediately notified so he can turn off the post indicator valve on the sprinkler system water supply to preclude excessive water damage.

Maintenance and repair of the sprinkler system is the responsibility of the AEC Fire Department.

C. Fire Brigade

1. Personnel Assignment

The Fire Brigade will consist of four persons for each shift assigned by the Nuclear Test Station Chief and two Health Physics personnel assigned by the Health Physics Supervisor. The persons assigned by the Nuclear Test Section Chief will be personnel normally located at the Reactor Facility.

2. Brigade Responsibilities

a. The specific assignment of the Fire Brigade is to extinguish a fire or confine its spreading until the Fire Department arrives and then assist the Fire Department as needed.

b. In the event of a fire during normal work hours, all available members of the Fire Brigade will immediately report to the scene of the fire and will be responsible to the senior STEP supervisor present.

c. Fire Brigade members will not attempt to fight fires in any electrical substation or transformer area until the AEC Fire Department arrives.

d. In the event of a fire, Brigade members will have first priority of available government transportation in the STEP area.

e. Brigade members will be responsible for providing emergency first aid assistance as required either as a consequence of fire or in conjunction with emergency re-entry activities.

3. Training of Brigade Members

The STEP Safety Engineer will be responsible for calling regular (monthly) meetings of the Brigade for instructing the members in their responsibilities.

D. Fire Alarms

1. Response to Fire Alarms

a. The area supervisor will immediately report to the scene of the fire.

b. Fire Brigade members will immediately report to the scene of fire.

c. Persons not engaged in fighting the fire shall remain away from that area.

2. Use of Fire Alarms

The instructions for actuation of the fire alarms appear simply and plainly on the front of each alarm box. Any persons needing assistance in fighting a fire may actuate the fire alarm signal.

3. Maintenance and Testing

a. It is the responsibility of the STEP Senior Engineering Supervisor to request that the AEC Fire Department test and maintain all fire alarms in the STEP areas.

b. Whenever feasible, all personnel will be notified of a scheduled fire alarm test prior to the test and at the time of the test.

REFERENCES

1. "Preliminary SNAP 10A-SNAPSHOT Safety Report", NAA-SR-6684 (Secret), February, 1962.

2. R. L. Brehm, "Summary Hazards Report and Operations Manual for SNAP Critical Assemblies 4A and 4C (SCA-4A-4C)", NAA-SR-MEMO-7011, January, 1962.

3. "Local Climatological Data ANP Area", U. S. Department of Commerce, Weather Bureau 1953-1961.

4. C. C. Gamertsfelder, "HTRE Hazards Report", APEX-180, December, 1956.

5. R. L. Nace, Morris Deutsch, P. T. Voegeli, "Geography, Geology and Water Resources of the National Reactor Testing Station, Idaho", IDO-22033-USGS, USAEC 1956.

6. G. H. DeMarrais, "The Engineering Climatology of the National Reactor Testing Station", IDO-12004, November, 1958.

7. "Population Distribution of NRTS", 1960 Census, USAEC Site Survey Office, Central Facilities, NRTS, Idaho.

8. "Off Site Population Distribution", Life Atlas, 1960 Census Survey, Rand and McNally, 1961.

9. G. A. DeMarrais, Norman F. Islitzer, "Diffusion Climatology of the National Reactor Testing Station", IDO-12015, April, 1960.

10. Private Communication with Weather Bureau concerning temperature variation at NRTS.

11. R. L. Nace, J. W. Stewart, W. C. Walton, "Geography, Geology, and Water Resources of the NRTS, Idaho", IDO-22034-USGS, USAEC, 1959.

12. "Seismic Probability Map of the United States", 1949, United States Coast and Geodetic Survey.

13. "Nuclear Safety Guide", TID-7016, Rev. 1, 1961.

14. R. L. Detterman, personal communication.

15. J. R. Huffman, personal communication.

16. F. H. S. Clark, Jr. III, "Analysis of S-2-DR Design", NAA-SR-5890.

17. NAA-SR-3700 (title classified).

18. M. A. Cook, "The Science of High Explosives", Reinhold Publishing Company, New York, 1958.

19. T. Rockwell III, "Radiation Shielding", TID-7004, March, 1956.

20. F. E. Obenshain and A. Foderaro, "Energy from Fission Product Decay", WAPD-P-652, January, 1956.

21. K. Shure, "Fission Product Decay Energy", WAPD-R(F)-38, May, 1961.

22. M. W. Hulin, ANS Transactions, Vol. 4, No. 1, June, 1961.

23. H. S. Van Ordstrand, "Current Concepts of Beryllium Poisoning", Annals of Internal Medicine, 35, Pages 1203-1217 (1951) December.

24. C. W. Schwenzfeier, Jr., "The Metal Beryllium", (Ed. by D. W. White, Jr., and J. E. Burke), American Society for Metals, Cleveland, 1955.

25. J. D. Buddery and G. E. Darwin, "Beryllium", Butterworth Publications Limited, 1960.

26. "Background Material for the Development of Radiation Protection Standards", Report #2, Federal Radiation Council, September 1961.

27. H. T. Lee and R. M. Brick, "Deformation of Beryllium Single Crystals at 25°C to 500°C", Final Report, NP-4949.

28. Hortmann, Nagy, Jacobson, "Explosive Characteristics of Titanium, Zirconium, Thorium, Uranium and Their Hydrides", Bureau of Mines, Report 4835, December 1951.

29. "Background Material for the Development of Radiation Protection Standards", Report #1, Federal Radiation Council, May 13, 1960.

30. "Meteorology and Atomic Energy", AECU-3066, United States Department of Commerce, July 1955.

31. C. Rogers McCullough, "Safety Aspects of Nuclear Reactors", D. Von Nostrand Company, Inc., New York, 1957.

32. "SNAP 2 Safety Evaluation Analysis", NAA-SR-4714, January 1960.

33. R. C. Bolles, N. E. Ballou, "Calculated Activities and Abundances of U^{235} Fission Products", USNRDL-456, August 30, 1960.

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OMIC ENERGY DIVISION