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

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

W. J. Neal, Editor

PHILLIPS
PETROLEUM
COMPANY

ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION

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Atomic Energy Division

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Idaho Operations Office

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ABSTRACT

This report presents the results of safety analyses performed on the SNAPTRAN 2/10A-1 and SNAPTRAN 2/10A-2 reactors. These reactors will be utilized to experimentally investigate the transient power behavior of a SNAP 10A/2 reactor in both the nondestructive and destructive region. Also included are descriptions of the test program, the reactor test packages, and the test site and facilities.

Calculations are presented to indicate the analytical techniques used to determine the transient behavior of the reactor for various reactivity additions and to determine the radiation doses to be expected for various conditions. The calculations show that an unplanned destructive excursion with an energy release of 220 MW-sec yields a maximum off-site dose of 6.3×10^{-2} -rem to a child's thyroid due to inhalation of the released fission products. From the results of the analysis, it has been concluded that the SNAPTRAN 2/10A-1 and SNAPTRAN 2/10A-2 test programs can be carried out without undue hazard to the general public.

SUMMARY

Kinetics testing of a reactor of the SNAP 10A/2 type will be conducted at the Test Area North (TAN) of the National Reactor Testing Station by Phillips Petroleum Company. This testing, which is a part of the Atomic Energy Commission's nuclear safety program, is to determine the response of the beryllium reflected SNAP 10A/2 reactor to large positive reactivity insertions. The testing program includes nondamage step and impulse tests initiated from ambient and from elevated temperatures, partial damage tests, and a destructive test. This report contains a description of the reactors, the test facilities, the experimental program and an evaluation of the hazards attendant to the conduct of this program.

The testing program has been divided in two parts, the nondestructive, SNAPTRAN 2/10A-1 program and the destructive SNAPTRAN 2/10A-2 program. The SNAPTRAN 2/10A-1 program includes static physics measurements essential to interpretation of the kinetic test results and kinetic tests aimed at measuring the dynamic coefficients of the reactor. The SNAPTRAN 2/10A-2 program consists of a nuclear excursion approaching the maximum possible for extending the range over which the kinetics information is obtained and to aid in assessing the consequences of a destructive excursion with a SNAP 10A/2 reactor. The reactors for performing these tests were designed and built by Atomics International using flight reactor components where possible. The primary difference between the flight reactors and the test reactors is the redesign of the control drum drive mechanisms to permit rapid reactivity insertions.

The potential hazards of conducting the experimental program have been evaluated. 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 nondestructive kinetics tests, and the consequences of the destructive test.

For the case where no fission product release occurs the maximum radiation dose to personnel is negligible. For the case of the planned destructive test, assuming an energy release of 200 MW-sec which is somewhat greater than expected, the maximum dose of 4.8×10^{-2} rem occurs to personnel in the control room and is again negligible. The off-site whole body and thyroid doses are also well below the recommended limits stated in AEC Standards for Radiation Protection, Chapter 0524.

For the unplanned destructive excursion, assuming an energy release of 220 MW-sec under weak lapse meteorological conditions, a maximum off-site dose of 6.3×10^{-2} rem occurs to a child's thyroid due to inhalation of the released fission products. This dose is well within the recommended limits stated in AEC Standards for Radiation Protection, Chapter 0524. It is therefore concluded that the SNAPTRAN 2/10A-1 and -2 test programs can be carried out without undue hazard to the general public.

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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 and Technology. That portion of the aerospace safety program directed towards studying the kinetic behavior and certain of the consequences of nuclear accidents of SNAP type reactors has been designated SNAPTRAN. Consideration has been given to the kinds of nuclear accidents which can conceivably occur, and two general categories have been found. These are: (a) nuclear excursions occurring in an atmospheric environment during assembly, transportation, installation of the reactor on the launch vehicle, or launch abort [1], and (b) nuclear excursions resulting from immersion of the reactor in water or wet earth during the above series of events. The SNAPTRAN 2/10A-1 and SNAPTRAN 2/10A-2 testing programs are aimed at providing information regarding the beryllium reflected reactor. The water immersion test, designated SNAPTRAN 2/10A-3, was completed in 1964.

Both nondestructive (SNAPTRAN 2/10A-1) and destructive (SNAPTRAN 2/10A-2) kinetics tests will be conducted. 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 at various initial temperatures up to approximately 1400°F. The destructive test will involve the addition of a sufficient amount of reactivity to destroy the reactor.

The reactor systems for performing the SNAPTRAN 2/10A-1 and -2 studies were designed by Atomics International and incorporated flight system design where possible with modifications to allow rapid insertions of reactivity. The SNAPTRAN 2/10A-1 reactor was fabricated, assembled, and acceptance tested by Atomics International and Phillips Petroleum Company personnel at Canoga Park, California. It was shipped to the NRTS where Phillips Petroleum Company assumed the operational responsibility for conduct of the tests. Technical direction of the test program for the initial critical experiment was provided by Atomics International personnel at the NRTS. Technical direction of all other phases of the test program was provided by Phillips Petroleum Company with technical consultation provided by Atomics International. Conduct of the critical experiment relied 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 relies on the experience and techniques developed by Phillips Petroleum Company in the performance of reactor transient tests with the SNAPTRAN 2/10A-3 and the Spert reactors.

This report includes a brief description of the test program, a description of the site, facilities, and reactor, and the results of the safety analysis conducted.

II. EXPERIMENTAL PROGRAM

The SNAPTRAN Experimental Program has been designed to provide: (a) information which provides insight into the nature and magnitude of the consequences of accidental water immersion of a SNAP 10A/2 reactor, considered to be the maximum credible accident pertinent to SNAP 10A/2 and reactors of similar type, and (b) information which permits the reasonable prediction of the dynamic reactor behavior as it influences those consequences. The specific designations for the programs are SNAPTRAN 2/10A-1 for the kinetic studies, SNAPTRAN 2/10A-2 for the destructive test with the beryllium reflected reactor, and SNAPTRAN 2/10A-3 for the water immersion destructive tests.

Information pertaining to the water immersion test has already been obtained and reported [2]. The SNAPTRAN 2/10A-1 and -2 experimental programs will be performed on two modified SNAP 10A/2 reactors. The modifications consists primarily of the addition of control drum drive mechanisms designed to move the drums very rapidly.

An experimental program has been formulated to explore several areas of reactor dynamics. The response of the reactor will be analyzed to determine the reactivity feedback function and its dependence on absolute temperature and rate of energy deposition. In addition, the effect of the reflector on neutron generation time will be analyzed with respect to its influence on the kinetic response of the reactor to large reactivity insertions.

1. PROGRAM OBJECTIVES

The objective of the SNAPTRAN 2/10A-1 and -2 programs is to provide information which will allow dependable prediction of the dynamic reactor behavior in the range where this behavior influences the consequences of a nuclear accident. The experimental program is divided into two parts: (a) nondamage kinetic testing and (b) testing which results in partial and complete destruction of the reactors.

The nondamage portion of the program, utilizing the SNAPTRAN 2/10A-1 reactor, will be directed at determining the negative reactivity feedback mechanisms and their dependence on absolute temperature and rate of energy deposition. The analyses associated with determining the fundamental feedback producing phenomena will be concentrated toward understanding the various effects of the bound hydrogen, present in the zirconium-fuel lattice, as a moderating mechanism. Also, the reactivity associated with reflector savings will be analyzed with respect to its dependence on neutron energy spectrum and reactor period.

That portion of the testing program dealing with partially and totally destructive transients will be aimed at providing information concerning the effects which influence nuclear and radiological consequences. These effects include

phenomena such as pressure buildup, particle excursion, and thermal expansion. This part of the program will utilize the SNAPTRAN 2/10A-1 reactor for the controlled minor damage tests and the SNAPTRAN 2/10A-2 for the full scale destructive test.

Step and impulse reactivity insertions will be used to satisfy the particular segments of the information requirements. The step inputs provide good resolution of reactor period versus reactivity input and temperature dependent reactivity feedback; however, the mechanical limit of drum motion reversal for scram initiation produces a high energy release to reactivity input ratio and, therefore, limits the magnitude of the reactivity insertion before damage occurs. The impulse reactivity input allows a much higher excess reactivity to be inserted, thus emphasizing the energy deposition rate and reflector kinetics effects on the reactor dynamics. However, a more complex analysis of the data is required. These two modes of reactivity input coupled with a variable initial isothermal temperature allows for the degree of flexibility required to fulfill the program objectives.

2. TEST PROGRAM

In order to achieve the above objectives in the time available, an integrated program was required in which the sequence of tests was ordered to permit optimum use of the reactor in a manner which eliminated need for hot shop work on the system, minimized the down time for system alterations required between various categories of tests, and minimized the radiation exposure of operating personnel.

The tests were divided into three basic categories; nonfuel damage, limited fuel damage, and destructive test. The first category contained six parts for adequately describing the experimental behavior of the reactor: (a) control system mechanical performance, (b) static physics tests, (c) ambient temperature step tests, (d) ambient temperature impulse tests, (e) elevated temperature step tests, and (f) elevated temperature impulse tests.

2.1 Nonfuel Damage Tests

2.11 Mechanical Check-out. Since the SNAPTRAN 2/10A-1 reactor is unique in that reactivity control for static operation, scramming, and transient initiation is effected by use of integral control units, the movable reflector segments, the control hardware, and control system logic are extremely complex. Extensive performance testing was, therefore, required to ensure against unplanned reactivity insertions.

The mechanical check-out included repeatability and positioning precision of the drums, drum scram time, and drum insertion time for all four drums using the transient reactivity insertion mechanisms. Performance of the attendant logic circuits, pneumatic systems, drive motors, etc, was also checked. This initial check-out was repeated just prior to the elevated temperature tests at various temperatures up to about 1000°F to determine if control system performance was not altered when operated in a high temperature environment.

2.12 Static Physics Tests. The nuclear testing began with a loading to critical [3] using the subcritical multiplication rate technique to estimate the critical number of fuel rods. Loading continued until the maximum fuel loading was reached and the excess reactivity was adjusted to that desired by the addition of beryllium reflector shims.

The following static physics measurements were made at ambient temperature:

- (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 and fuel,
- (4) measurement of flux distribution, and
- (5) power calibration of neutron and gamma detectors.

The elevated temperature static physics tests will include many of the same measurements. In addition, the isothermal temperature defect will be determined for the combinations of a hot reactor-cold reflector, and hot reactor-hot reflector. The temperature defect will be determined in several increments between ambient temperature and about 1200°F.

2.13 Ambient Temperature Step Tests. The kinetics testing commenced with a series of step reactivity insertion tests which resulted in asymptotic periods in the order of seconds. The series proceeded to reactivity insertions which resulted in asymptotic periods as short as 1.7 msec. The termination of this series was based on a temperature below which no bound hydrogen was released. The tests in this series were initiated with the reactor subcritical at a power level of about one milliwatt.

The data from these tests are being evaluated to provide information regarding the prompt negative reactivity coefficients. In addition, the dynamic behavior of this reactor system shows good agreement with predictions from analytical models.

2.14 Ambient Temperature Impulse Tests. In order to extend the experimental determination of the negative reactivity coefficient contributors to shorter effective periods while maintaining nondamage producing energy release, the reactor will be excited by a series of impulse insertions of reactivity. For these tests, drum drive mechanisms are provided which permit two drums to be coupled and to be rotated through the reactivity region for the purpose of generating approximately a cosine shaped reactivity-time function. It is anticipated that these tests will provide information concerning the effect of the reflector on neutron generation time. In addition, a series of single-drum impulse tests will be performed from several steady power levels to determine the feedback coefficient in a manner which is independent of the reactivity time history throughout the impulse.

2.15 Elevated Temperature Transient Tests. The hot-spot to average core temperature ratio will limit the average core temperature for the non-damage, ambient temperature tests to about 800°F. However, by heating

isothermally, the temperature coefficient may be determined to temperatures approaching 1200°F.

This test series will commence with mechanical check-out at temperatures ranging from ambient to about 1000°F and will be followed by a limited number of static physics tests to determine effect of temperature on the reactivity worth of the drums. The transient tests will be carried out using step and/or impulse reactivity insertions.

2.2 Limited Fuel Damage Tests

In order to determine the dynamic coefficients of the reactor for fuel hot-spot temperatures above about 1500°F, and to gain information on the energy deposition rate dependence for the threshold of hydrogen evolution and fuel disintegration, several tests will be conducted in which damage will be sustained by the reactor core. It is intended that this damage be limited to a cluster of rods in the center of the core in which the fuel clad may be severely deformed as a result of hydrogen pressure buildup. It is expected that the control system will not be damaged. It has been shown in out-of-pile tests that pressures of about 2000 psi can be contained before fuel clad rupture occurs when the clad is at room temperature.

2.3 Destructive Test

A second machine designated SNAPTRAN 2/10A-2, will be used in the destructive test and will be altered functionally from the SNAPTRAN 2/10A-1 machine by providing a mechanism for stopping the impulse drums at 0° (the maximum reactivity position). Other modifications may include addition of special mirrors and additional fixtures for support of disassembly measurement detectors and lighting fixtures for photographic purposes.

The reactor core, as for the kinetics tests, will be without NaK. This configuration permits a more interpretable extrapolation of the nondamage kinetics test and allows an upper limit energy release to be generated since the mechanical tamping effected by the NaK will not be available to otherwise act earlier (and at a lower energy) in displacing the beryllium reflector.

After an operational check and reactivity calibration of control drums to verify similarity to the SNAPTRAN 2/10A-1 machine, several low energy step transient tests will be run as a power calibration of the nuclear detectors and to verify operability of the instrumentation.

Final preparations will then be made for radiological and photographic coverage. Following a series of successful dry runs and pending meteorological conditions consistent with the required wind velocity, direction, and predictability, the destructive test will be performed from ambient conditions.

2.4 Post-Test Examination

Upon completion of the destructive test, an area survey will be made to determine the extent of contamination spread and the distribution of reactor core and structural components. All sizable fragments of fuel and other reactor components will be collected and moved with the reactor remains to the examination area. An extensive meteorological, radio-chemical, metallurgical, and chemical analysis will be made to aid in determining the nuclear, thermal,

chemical, and mechanical effects that existed during the excursion. It is expected that the information gained from this test will aid in understanding the mechanism of hydrogen release, fuel disintegration, and fission product release.

III. TEST PACKAGE DESCRIPTION

The SNAPTRAN 2/10A-1 and -2 test packages (Figures 1 and 2), modified SNAP 2/10A flight machines, have 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 modification 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 heat source. Figure 3 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

Fig. 1 SNAPTRAN 2/10A-1 reactor assembly.

Fig. 2 SNAPTRAN 2/10A-1 reactor pictorial.

being provided with scram capability. For normal reactor control the drums are driven, individually or grouped, by a continuously variable motor drive capable of rotating the drums at rates from $0.05^\circ/\text{sec}$ to $1^\circ/\text{sec}$. The impulse drums can be driven through an arc of 450° , 225° on either side of the maximum reactivity position (0°). The step drums can be driven through an arc of 180° clockwise to 0° and counterclockwise to 180° . 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 shaft opposite the drums.

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° away. The step drums have mechanical stops that prevent drum rotation beyond 0° . Each of the step drums is driven by a pneumatic-mechanical system capable of rotating the drums 45° in 0.060 second. In a two-drum step insertion, the two drives are linked together with a chain and sprocket assembly to assure coordination of movement. The scram mechanisms for the step drums are mechanical and are designed to rotate the drums from 0 to 135° in 0.110 second.

Fig. 3 Test package model in IET.

An impulse test will be accomplished by rotating one or two drums through the full 450° of travel; however, only for $\approx 270^\circ$ (135° on each side of 0°) will the drums be in the reactivity region. The system is designed to rotate the drums through this 270° arc in 0.030 second by pneumatic-mechanical means. During the two drum impulse tests, the drums are linked together with a chain and sprocket system to assure coordinated 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 reactivity region, -135 to 0°, being traversed in 0.015 second. A hydraulic system will be added prior to the destructive test to stop drum rotation at about 0°. The impulse scram is pneumatic-mechanical and is designed to rotate the drums from 0 to +135° in 0.120 second and from the dead spot (-20°) to -135° in 0.140 second.

Following the transient test series, the SNAPTRAN 2/10A-1 test package will be moved to the examination area and disassembled. The SNAPTRAN 2/10A-2 test package, to be used for the destructive test, will then be assembled on the test dolly.

1. STRUCTURE

The structural framework for the test package is made from five-inch 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 pins welded to the deck. The mounting pins are designed to prevent the package from turning over during transit or maintenance operations, 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 controls are located on the console. The rotational motor-brake assembly is used to invert the rotational framework in order to permit removal or replacement of the control drum drive assemblies. 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 temperatures, 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 12.5 kW electrical heaters which are used to heat the Be reflector and to maintain reactor temperature during the elevated temperature tests.

For the tests above ambient temperature, the reactor temperature will be raised by a heated stream of N_2 gas and by the door heaters (Figure 4). 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 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 , 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 stored in three liquid N_2 storage tanks until required for a test. The tanks are pressurized by air heating in a recirculation line and the tanks are protected against overpressure by rupture discs and relief valves.

The liquid N_2 flows from storage to the vaporizer where it is vaporized and heated to about 70°F and 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 passes 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.

The N_2 heater, which is controlled by a saturable reactor, 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.

2. REACTOR

2.1 Core

A cross section of the reactor is given in Figure 5. The core is a right cylinder which consists of 37 fuel rods arranged in a triangular array on 1.260-inch centers to form a hexagon eight inches 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 weight percent of 93 percent enriched uranium in the form of 1.250-inch-OD x 12.35-inch-long cylindrical rods. The fuel-moderator density is 6.08 g/cm³, and its volume is 520 in.³ (8540 cm³). 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 with a wall thickness of 0.015 inch. A gap of approximately 0.001 inch exists between the cladding and fuel-moderator. The ultimate strength of the fuel-moderator alloy

Fig. 4 Nitrogen heating system.

Fig. 5 Reactor cross section.

open at both ends to allow photographic coverage. 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 in the test building ventilation system. During the initial static and transient tests, the void areas in the vessel will be filled with nitrogen, and for the destructive test, the reactor will be open to the atmosphere.

2.3 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 (Figure 5). The nominal thickness of the reflector is 1.875 inches; however, up to five 0.125-inch shims can be added to increase the thickness to 2.5 inches.

2.4 Control Drums and Drives

The control drums are the same as those used on the SNAP 2/10A flight machine. A drum is composed of a sector of a right circular cylinder of 3-1/2-inch radius and has a maximum nominal thickness of 1-3/4 inches (Figure 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.

Copper 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.

is about 15,000 psi, and the bursting pressure of the Hastelloy N cladding is about 1500 psi at elemented temperatures. Caps and grid plate indexing pins are welded to the ends of each fuel rod. 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.

2.2 Vessel

The reactor vessel (Figure 6) is cylindrical in shape, 8.94-inch OD, with a 0.032-inch 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 be

Fig. 6 Vessel pictorial.

Normal positioning of the drums will be accomplished by rotating the drums to their proper positions with an electromechanical system consisting of a geared drive motor, a gear reducer, an electromagnetic tooth clutch, and the drive shaft (Figure 7). The motor is a permanent magnet, dc motor having speed control through supply voltage control. Armature speed at a drum speed of $1^\circ/\text{sec}$ is 20,000 rpm. Armature disassembly is likely to occur at approximately 30,000 rpm due to centrifugal forces, so the maximum drum insertion rate is limited to a maximum of $1.5^\circ/\text{sec/drum}$. 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

Fig. 7 Stepper drum and drive assembly.

scrammed. The tooth clutch is used to provide a 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 a torque of 300 ft-lbs, 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 1-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 to 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 two inches, 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 N₂ 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 N₂. 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 inch 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-inch clearance between the tooth clutch and the lip of the clamp is used to ensure 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 minimize a malfunction due to a valve failure. Two pressure switches 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 venting the cylinder. 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 were scrambled without the damper cylinder being pressurized. The driving cylinder and damping cylinder are rated for 2200 psi. Driving pressure will be ≈ 350 psi and the damping cylinder pressure will be ≈ 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 (Figure 8) includes the driving-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 reactivity worth. 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 -225 and -135° used to accelerate the drum and the dead-band between $+135$ and $+225^\circ$ 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 $5\frac{1}{2}$ inches in diameter. The piston is driven to one end of the cylinder by the drive motor. 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 wall

Fig. 8 Impulse drum and drive assembly.

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. Compression of this trapped N_2 cushions and stops the piston at the opposite end of the cylinder. A motor-operated valve in the exhaust line is positioned to bleed the pressure around to the back side of the piston so that the pressures are equalized on both sides of the piston 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. The pressure used to drive the impulse piston is 1000 psi. The acceleration on the drum due to the 1000 psi driving pressure 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 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. In addition, a hydraulic load cell is used as a backup device to prevent the impulse drum from being driven back into the reactor after the completion of a transient. The sprag clutch and the hydraulic load cell must be disengaged to permit the impulse drums to be inserted by the drive motors; power is required to change them from engaged to disengaged and vice versa.

Scramming of the 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 scam the drums, the tooth clutch is released, and the spring drives the rack to return the drums to the full-out position. To scam the step drums after a step reactivity input, the driving cylinder pressure is vented. The step drums can only be driven in the positive sector and always scam in the counter-clockwise direction.

The scam mechanism for the impulse drums is considerably more complex because it operates in both sectors. It must not scam the drum back through the maximum reactivity position from the negative sector, and it must not exert a counter force to the drive piston driving an impulse reactivity insertion. 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 both sides, corresponding to the positive and negative sectors. 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 scam the drum. The scam 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 scrambled.

There are two limitations associated with this mechanism: the cam dead spot and the need for invalidating the scram during an impulse 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 20° beyond the maximum reactivity position in the negative sector. This minimizes the possibility of the drums 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. 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. The pressure required to lift the rollers off the cams is 350 psi. 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.

2.5 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 switch indications. Digital indication is obtained from a servo-synchro system which has an overall accuracy of $\pm 0.1^\circ$. The readout from this system is located on the control console. The analog indication is obtained from a potentiometer which has an overall accuracy of $\pm 1^\circ$. The signal from the potentiometers is recorded on a panel mounted adjacent to the console.

The drum positions are also indicated by lights actuated by cam operated switches. On all drums the maximum reactivity position (0°) and the minimum reactivity position (180°) are indicated. In addition, indications are given to show whether the impulse drums are in positive or negative sectors, and also to show if the impulse drums are in the sectors from -135 to -225°.

2.6 Reactor Characteristics

The reactor characteristics are summarized in Table III-1.

TABLE III-1
SNAPTRAN 2/10A-1 REACTOR CHARACTERISTICS

<u>Core Composition</u>	
Fuel Elements	
Number	37
Fuel diameter (inches)	1.212
Cladding material	Hastelloy N
Cladding diameter, OD (inches)	1.25
Cladding thickness (inches)	0.015

TABLE III-1 (Cont.)

Fuel alloy, wt% U in Zr	10
Degree of hydriding, $N_H(10^{22} \text{ a/cc})$	6.5
Active fuel length (inches)	12.25
Internal Reflectors	
Composition	Be
Cladding	None
External Reflectors	
Composition	Be
Cladding	None
Nominal thickness (inches)	1.875
Reflector Control Elements	
Number	4
Material	Be
Nominal thickness (inches)	1.875 minimum 3.629 maximum
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
<u>Nuclear Parameters</u>	
Maximum uranium loading, Kg U-235	5.00
Mean fission energy (eV)	0.18
Prompt (fuel) temperature coefficient ($\phi/^{\circ}\text{F}$)	-0.12 ± 0.04
Mean prompt neutron lifetime, 10^{-6} sec	6.5
Initial cold excess reactivity (β)	≈ 5.00
Total control drum worth (β)	≈ 9.00
Effective delayed neutron fraction	0.008

3. CONTROL SYSTEM

3.1 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 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 programming 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.

3.2 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 sides of a square. The two center sections, as shown in Figure 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 start-up, 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. 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.

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 program sequence timer 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, is not used. A cabinet containing two closed circuit television monitors and the television system controls is positioned adjacent to the end section of the console.

The rectangular section adjacent to the left corner contains two logarithmic picoammeter amplifiers with meters and trip circuits to monitor reactor period and level and to scram the reactor at preset period or power levels. The left end section contains two pulse counting scalers from which precise neutron counts may be obtained during start-ups and for determining criticality.

Adjacent on the left end section is a cabinet containing a pressurized ink strip chart recorder with ten channels, eight of which are capable of one percent

Fig. 9 Control panel layout and schematic.

accuracy and 4 ms full-scale rise time. The other two provide time reference markers. This "events recorder" has 12 chart speeds ranging from 3 mm/min to 20 cm/sec. Four channels are used for recording drum position as indicated by the shaft-mounted potentiometers; four more channels are 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, programmed scrams, or normal operation scrams.

Key operated switches are provided on the console to control: (a) power to the console, (b) arming of the stepper drives, (c) lockout of scram on the impulse drives, (d) arming of impulse drives, (e) starting of program sequence timer, (f) driving of impulse drums past the fully inserted position into the negative region, and (g) 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, permission necessitating from health and safety supervision as well as nuclear operation supervision before the control system can be energized.

Except for the red mushroom-head scram button and the keyswitches, all lights and switches on the control console are 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.

The lights indicate closed or open circuits by being on or off except when 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 comes on white) and while the circuit is "operative" during the program sequences 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

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 pressurize the drum drive cylinders.

3.3 Console Power

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.

400 cps, ac power is required for the servo-synchro digital drum position indicating system. This is provided by a solid state inverter from the 28 volt dc console power.

The diagram consists of two parts. The top part shows a battery charger circuit. It starts with a +20 V source connected to a switch. The switch is labeled "BATTERY CHARGER OFF (CONSOLE)". The circuit then goes through a battery (B) labeled "BATTERY CHARGER OFF (BATTERY ROOM)" and a DC- terminal. The bottom part shows a motor generator circuit. It starts with a 400 V, 30 A source connected to a motor generator (M). The motor generator is connected to an electric operated breaker (C) labeled "ELECT. OPERATED BREAKER". The breaker is connected to a 24 V source, which is connected to a console power keyswitch.

Fig. 10 Console power supply.

to commercial power when the generator is not operating, when the reactor is shut down, or when a failure occurs.

3.5 Scram Systems

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

Current for the tooth clutches is supplied by the "scram bus". The scram circuitry (Figure 11) connecting the scram bus 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 bus 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

Fig. 11 Scram-fire logic control system schematic.

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 two portable scram buttons are located 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 of the scram. The sigma scram relay, which may be replaced directly by an amplifier when fast response is demanded, accepts scram signals from sources such as power level and period safety trip instrumentation or a program sequence timer. Two "Log N" period and level amplifiers with appropriate trip circuits are provided for safety scrams. The trip levels for these circuits will be set to operate at a power not to exceed 1100 watts or on a period of 10 seconds or shorter. These restrictions, effective during normal static operation, will be bypassed automatically during the transient tests by the program sequence timer used to initiate the tests.

Each tooth clutch is controlled by a selector circuit containing a holding relay and a sigma relay similar to the master scram circuit. The tooth clutches are energized individually by momentary console push buttons which shunt the holding relay contacts. A single tooth clutch can be dropped out by depressing the "tooth clutch off" switch, and all clutches may be dropped out simultaneously by scrambling, by turning off console power, or by a signal to the sigma relay.

No interlocks are provided which would 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.

3.6 Drum Motor Drive Control

In order to effect drum motion (Figure 12) from the motor drives, the appropriate tooth clutches must be engaged and the scram circuit made operative. A single pistol grip, drum insert-withdraw 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 drum insert-withdraw switch. These motor selector relays, which have double-throw contacts, serve a dynamic braking function. The relays 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, drum insert-withdraw switch, interlock and control relays, and motor armature current ammeters comprise the basic motor drive circuitry.

A number of interlocks, which apply to all drums, are provided between the drum insert-withdraw switch and the motor selector switches in the drum insert circuit. Thus no drum insertion can take place if interlock relays do not indicate that contamination monitors, start-up channels, events recorder, and logarithmic operational power recorder are operating, stepper dampers are operating, and impulse scram springs are not locked out. Also, indication from any "arming" or "armed" relays or coupling clutch relays will prevent

drum insertion. Motor selector relays for the stepper drives operate directly from the insert-withdraw switch through the motor selector switches, rack travel limit switch relays, and the above mentioned insertion interlocks. Motor selector relays for the impulse drives require, in addition, limit indicating relays to stop the drives at the full-in and full-out positions and a keyswitch to allow driving beyond the full-in position together with a selector reversing switch to cause the selector relays to drive the motors in the proper sense (clockwise or counterclockwise) in accordance with drum location to give insertion or withdrawal. Also, interlocks prevent both impulse drives being driven onto the dead spots on the scram cams at the same time.

The motor selector relay circuitry just described uses 28 volt dc console power. In order to provide drive speed control, the motors are driven through the selector relays from a variable transformer and full wave bridge rectifiers.

The motor selector relays are interlocked with one another to provide dynamic braking and to minimize the hazard of welded relay contacts. Also, in addition to the interlocks in the motor selector relay circuits which prevent all insertion if any drum is armed, interlocks in the motor power circuits prevent any driving at all of individually armed drums.

3.7 Pneumatic Drum Drive Control

3.71 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-2.4). 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 the resulting increase in nuclear energy release might cause damage to the reactor. Hence, to provide interference from the scram mechanisms, 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.

All of the pneumatic control valves for 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 bus or the scram bus to maintain the impulse drives in either the armed or disarmed state.

In order to set up for an impulse test, the impulse drum(s) 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 and "reverse" button(s). In normal static test operation, "drum in" limit switches act as limits on drum insertion by motor drive. The withdraw circuits ordinarily permit rotation back through 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 and "reverse" button(s). 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 actuated. Tooth clutches must be energized on the drum(s) to be armed. Indication of tooth clutch energization 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 Figure 13). 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 racks.

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 impulse damper valve(s) is then opened and the sprag clutch(es) and load cell(s) switched to the operative mode, in preparation for deceleration and restraining of the impulse drum(s) at the end of travel (+ 225°). Interlocks prevent these steps until scram lockout is accomplished.

With the above interlocks satisfied, actuation of the impulse arming keyswitch 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. A normally closed test switch for each charge valve and a normally open test switch for each vent valve are included in the arming circuit to provide for an operational check of the valves.

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

3.72 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(s) 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 bus, 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 a scram signal 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 Figure 14. As in the case of the impulse test, the step drums must be in the cocked position 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 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.

3.8 Sequence Timer

To provide the precise resolution required in performing very short period transient tests, an electronic sequence timing system is provided. The system consists of two independent, 5-channel, solid state, electronic counter-timers, each with its own power supply. Maximum time cycle of a single channel is 999.99 seconds. Longer cycles may be obtained by sequencing two or more channels. Minimum resolution of program settings is 100 microseconds, with nominal accuracy of setting being 10 microseconds, or one cycle of the crystal oscillator used as the master clock in the system.

Fig. 14 Step arm logic control system schematic.

The critical function of the timer in performing transient tests is programming the sequence of events which includes the following:

- (1) bypassing the safety amplifiers, which otherwise would abort the test the moment the reactor attained a transient period
- (2) removing the tooth clutch clamps from armed drums
- (3) initiating the transient by releasing the tooth clutches of armed drums
- (4) scrambling the reactor by releasing remaining tooth clutches and venting pneumatic drive pressure
- (5) re-establishing safety amplifier control.

In addition, other important functions are served such as starting high speed cameras, starting recording oscillographs, and providing event marking signals.

Failure to scram at the correct time following a step transient could allow an energy release of destructive magnitude to occur. To maximize reliability in the control of such vital functions, the timing system has been designed in two independent sections so that two independent channels can be assigned in parallel to such a function when necessary.

Control of circuits by the timer is implemented through silicon controlled rectifiers triggered by the counters. Thus the delay time of electromechanical relays is eliminated.

Test abort is not an explicit function of the timer, but of external interlock circuitry. The voltage on the "fire" circuit closed by the timer derives from the "relay" circuit which depends upon the drums remaining properly armed. Thus, no "fire" voltage is available if arming pressure drops. Also, if a tooth clutch clamp malfunctions in the sequence the "fire" voltage is routed to a disarm circuit instead of firing the transient. Manual abort switches are also provided both on the control console and at the magnetic tape recorder racks.

4. OPERATIONAL INSTRUMENTATION

For effective surveillance of the nuclear status of the reactor, the operator will have control of and indication from: (a) four channels of neutron pulse counting equipment covering source level to 5×10^{-2} W, (b) one channel of linear power recording equipment covering 10^{-6} to 10^2 W, and (c) three channels of log power equipment covering 10^{-3} to 10^4 W; one channel is recorded and the other two channels have automatic period and level scrams for safety purposes. The operational instrumentation is schematically represented in Figure 15.

Fig. 15 Operational instrumentation.

The detectors for two start-up channels will be placed against the fixed beryllium flats on the reactor (inside the thermal box) where the variable shielding effect from rotating drum movement will be minimized, and the detectors for the other two channels will be mounted on the reactor support structure for data correlation when high temperature tests are conducted. The two detectors inside the thermal box will be removed during high temperature tests.

The detectors for the linear and log power channels will be mounted on the reactor support structure.

A one curie source was placed directly under the reactor for the loading experiments where the source radiation would be seen through the reactor by the start-up channels. This source was also used for reactor operation once the full core loading was achieved until the reactor itself provided a source of neutrons adequate for safe start-up.

5. EXPERIMENTAL INSTRUMENTATION

The reactor will be instrumented within the capabilities of existing instrumentation technology to provide the maximum amount of information for interpreting and assessing reactor kinetic behavior and the phenomena of a destructive power excursion. To ensure measurement capability, stringent requirements are being imposed upon the pressure sensors, temperature measuring devices, and other detectors with respect to adequate response characteristics as well as physical characteristics conducive to placement within the reactor and/or its environment.

The measurements to be made during the nondamage 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) external vessel temperature
- (8) vessel strain; hoop and longitudinal
- (9) fixed beryllium reflector temperature and strain.

For the destructive test the following parameters will be measured:

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

The recording instrumentation for the above measurements is schematically represented in Figure 16.

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

To enable an evaluation of the radiological hazards, the proposed instrumentation distribution (Figures 17 through 20) 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. 16 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

KEY

- FILM BADGES, FALLOUT PLATES & POCKET DOSIMETERS
- * RADIOLOGICAL SURVEILLANCE TOWER
- ◇ POLYMERIZATION, CHEMICAL AND/OR GLASS DOSIMETERS
- ▲ HIGH VOLUME AIR SAMPLERS
- △ REMOTE AREA MONITOR
- FILM BADGE DROP
- NUCLEAR ACCIDENT DOSIMETERS

Fig. 17 Radiological grid 2 and 10 meter arcs.

- FILM BADGES, FALLOUT PLATES & POCKET DOSIMETERS
- ◐ FILM BADGES & POCKET DOSIMETERS
- ∞ GRASS SAMPLERS
- + PARTICLE SIZING SAMPLERS
- ◊ FISSION GAS DETECTORS
- ◻ ANTHRACENE CRYSTALS
- NUCLEAR ACCIDENT DOSIMETERS
- ▲ HIGH VOLUME AIR SAMPLERS
- △ REMOTE AREA MONITORS
- CONSTANT AIR MONITORS
- ◊ POLYMERIZATION, CHEMICAL AND/OR GLASS DOSIMETERS
- ◻ FILM BADGE DROPS
- ★ IODINE SCRUBBERS
- ▽ WELL TYPE FILM BADGES
- SAMPLING RADIOLOGICAL TOWERS

Fig. 18 Radiological grid 25 and 50 meter arcs.

- FILM BADGES, FALLOUT PLATES
& POCKET DOSIMETERS
- NUCLEAR ACCIDENT DOSIMETERS
- ▲-HIGH VOLUME AIR SAMPLERS
- △-REMOTE AREA MONITORS
- FISSION GAS DETECTORS
- + -PARTICLE SIZING SAMPLERS
- FILM BADGE DROPS
- FILM BADGES & POCKET DOS-
IMETERS
- α-GRASS SAMPLERS

Fig. 19 Radiological grid 100, 200, and 300 meter arcs.

- FILM BADGES, FALLOUT PLATES
& POCKET DOSIMETERS
- FILM BADGES & POCKET
DOSIMETERS
- △-REMOTE AREA MONITORS
- ▲-HIGH VOLUME AIR SAMPLERS
- α-GRASS SAMPLERS

Fig. 20 Radiological grid 500, 900, and 1500 meter arcs.

- (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 isotopic identification as a function of position on the grid
- (13) particulate activity in the cloud and isotopic identification as a function of position on the grid
- (14) noble fission gas concentrations and isotopic distribution in the cloud
- (15) cloud concentration and distribution of radioisotopes according to particle size
- (16) amount of beryllium release.

Phillips Petroleum Company personnel will be responsible for radiological measurements for distances up to ≈ 2.3 miles downwind of IET. The AEC-ID Health and Safety Branch will be responsible for measurements and evaluation of the radiological parameters for distances greater than 2.3 miles.

IV. DESCRIPTION OF SITE AND FACILITIES

1. SITE DESCRIPTION

1.1 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 Figure 21. 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 [4]. 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 and -2 reactor test programs will be conducted in the IET facility.

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

The surface of the area is nearly flat with an elevation of 4800 feet at the IET. Surface drainage is good and studies have shown there are no important impermeable boundaries existing within the TAN area [5].

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 feet high and contains wind direction and temperature measuring equipment.

1.2 Geography

1.21 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 (Figure 22). From north to south it is 34 miles long, and from east to west it is 29 miles wide [6].

The site is at an average elevation of 5000 feet above sea level with a one percent slope from southwest to northeast. As seen in Figure 23, which is a topographic cross section along the X', X' and Y', Y' axis of Figure 22, the elevation is lower along the Snake River to the east except for a gentle 600 foot rise midway between the site and the river.

The mountains, seen in Figure 22, encircle the Snake River Plain and rise to elevations of 10,000 and 11,000 feet 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 [6]. 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 [7].

[illegible]

45

TABLE IV-1

TAN FACILITIES

Location	Distance from IET	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 and Maintenance Area (AMA)	1.3 miles	Southwest	158

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

(2) Off Site. The towns with unlisted populations in Table IV-3 are unincorporated and no census figures are available. For locations of towns see Figure 24.

1.3 Meteorology

The meteorological conditions at NRTS are described in detail in previous reports [7, 10]. These reports are used as a basis for the following data:

1.31 Temperature [11]. The maximum recorded temperature at the Weather Bureau Office (WBO) at NRTS was 101°F in July 1960, and a minimum temperature 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.

1.32 Precipitation. Precipitation for the NRTS in general averages 7.69 inches 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 inches in June 1954, and that of snow, 8.6 inches in January 1957. For the TAN area in particular, the annual amount of moisture is 7.35 inches, with a maximum rainfall per 24 hours of 1.33 inches recorded in July 1953.

1.33 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

Fig. 22 NRTS and southeast Idaho.

Fig. 23 Topographic cross section through NRTS.

TABLE IV-2

POPULATION DISTRIBUTION OF NRTS^[8]

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		

TABLE IV-3

OFF SITE POPULATION DISTRIBUTION[9]

<u>Location</u>	<u>Distance from TAN in Miles</u>	<u>Population</u>	<u>Direction from TAN</u>
Mud Lake	10-15	187	East
Montevieu	10-15	[a]	NE
Berenice	10-15		West
Terreton	10-15	[a]	East
Howe	10-20	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
Mean	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 (Cont.)

OFF SITE POPULATION DISTRIBUTION^[9]

<u>Location</u>	<u>Distance from TAN in Miles</u>	<u>Population</u>	<u>Direction from TAN</u>
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

[a] The rural population in the area surrounding Mud Lake, Montevideo, and Terreton is approximately 1,000.

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 percent 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 Figure 30 of IDO-12015 [10], inversions may be expected 92 to 98 percent 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 percent 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 [10] lead to the following conclusions:

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

Fig. 24 National Reactor Testing Station and vicinity.

winter seasons the probability of an inversion of such duration increases considerably.

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

1.34 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 (shown in Figures 25 and 26). 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 percent was noted between the value for the diffusion coefficient, C , as listed in the HTRE hazards report [5], and the values listed in IDO-12015 [10]. The values listed in IDO-12015, and as shown in Tables 30 and 31 of that report, are used in Sutton's diffusion equations since these values were found by the NRTS Weather Bureau Office to be more accurate than those listed in the HTRE report.

1.4 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 covered by waterborne and windborne topsoil, under which there is a considerable depth of gravel ranging in size from fine sand to three inch aggregate [6].

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 [12].

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 [12].

The altitude of the water table ranges from 4580 feet above sea level in the northern part of the station to about 4400 feet near the southwest corner. The water table in the TAN area is at a depth of 200 feet 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 [5].

1.5 Seismology

The NRTS site is located in a region which "The Pacific Coast Uniform Building Code", (1949), designated as a zone-two area [13]. Although many recorded earthquakes have been felt in Idaho, none were of sufficient intensity

Fig. 25 TAN 20-foot level wind roses, 1953 through 1956.

Fig. 26 TAN 150-foot level wind roses, May 1956 through April 1957.

to cause more than minor damage to buildings. Of the fourteen 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 [6].

Although the lava plain of the Snake River is geologically young, the surrounding mountains are mostly of great age [6]. 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.

2. FACILITIES DESCRIPTION

The facilities used in the SNAPTRAN 2/10A-1 and -2 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.

2.1 IET Building

2.11 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 (Figure 27), restricts access to within approximately one mile of the facility. The immediate IET area (Figure 28) 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

Fig. 27 Vicinity map of TAN area.

Fig. 28 Plot plan of IET area.

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.

2.12 Control and Equipment Building. A plan view of the control and equipment building is given in Figure 29. The building is a poured concrete structure approximately 50 feet wide x 200 feet long, having two-foot-thick walls and a three-foot-thick ceiling. The ceiling is two feet below grade and is covered by mounded earth approximately 14 feet deep. The control and equipment building and the test cell are separated by 15 feet of earth at the closest distance.

Fig. 29 IET control and equipment building.

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 x 38 feet 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 Figure 30.

The large room directly east of the control room measures 28 by 44 feet 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 x 31 feet, 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 as 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 x 106 foot equipment room which houses air compressors, freon refrigerating units, air washers, and water treating equipment. The 21 x 20 foot 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 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 x 24 foot electrical switchgear room contains the protective equipment and breakers for commercial and emergency power distribution. The 29 x 24 foot diesel room houses a 600 kW emergency power diesel generator.

A four-foot-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 eight-foot-wide door between the building and the parking area allows small truck access for repair and replacement of equipment.

2.13 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 Figure 31. 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 feet from the test pad. When in place over the

Fig. 30 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

pad, about 43 x 60 feet 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 feet long x 22 feet wide with an 8 foot 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 feet long x 1 foot high and can be closed by means of lead shielding doors. As the coupler on the rail dolly is only about eight feet 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.

The periscopes for remote viewing from the control room are located on either side of the coupling station and are housed in a 10-inch pipe surrounded

Fig. 31 IET test cell.

by 7-inch-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.

2.14 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-inch line from the wells and storage in the TSF area. Flow rates of 1600 gpm can be obtained for four-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-inch raw water supply line and distributed by an 8-inch 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 four-hour period.

The IET control and equipment building is provided with hose stations, deluge spray systems, portable CO₂ 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 CO₂, three 10 lb CO₂, 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. CO₂ fire extinguishers are strategically located in the test cell.

The tank building, serving as the nitrogen unloading and vaporizing station, has two CO₂ extinguishers. 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 percent makeup air which is filtered through a two-inch coarse glass filter and two 1/2-inch layers of fine spun glass which provide a filtering efficiency of 99.4 percent based on particle count of atmospheric dust. Actual tests on particles of 0.9 micron DOP were conducted which indicated the effectiveness of the filters to be at least 99.7 percent.

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. This is due to 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 both manual and 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-inch line. Maximum flow through this system has been calculated at 630 gpm. A hot waste, coarse-filter sump is installed in the 10-inch 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-inch 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-inch line from the decontamination area and one 2-inch line from the trench in front of the coupling station drain into the 10-inch line terminating in the hot waste tank. The portion of the 10-inch 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 Idaho Chemical Processing Plant (ICPP) or to the evaporator in the TSF area
- (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.

2.2 Examination Area

The examination area, designated 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 (the hot shop, hot cells, metallurgical and chemical laboratories, etc) are of primary importance and will be used to perform remote maintenance on the reactor package, and to perform chemical, physical, and metallurgical examination of the reactor fuel during the kinetic tests and following 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 Figures 32, 33, and 34.

2.21 Hot Shop

(1) Construction. The hot shop is an area 160 feet long x 51 feet wide x 67 feet 6 inches 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 x 12-1/2 feet), 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.

Each turntable has an overall diameter of 17 feet 6 inches 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.

Fig. 32 TAN building 607 -- first floor plan.

Fig. 33 TAN building 607 -- second floor plan.

Fig. 34 TAN building 607 -- third floor plan.

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 feet 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 300 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 feet 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 feet from the floor in an "up" position; 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-inch tubes passing through the hot shop wall above one of the hot shop windows. These manipulators are designed for a 6-foot 10-inch wall thickness, for work loads in the order of four pounds 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 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-foot 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 six-inch 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 six-inch stainless steel pipe which also goes to disposal building 616. The two six-inch drain lines leading from the hot shop connect to a common sump. The waste is then pumped the 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-inch 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 inch of water pressure in the warm services area and a negative 1/4 inch 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.

2.22 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.

2.23 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 operation 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 three-foot 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 facilitate 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 three-inch 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 CO₂ cylinders manifolded to four discharge nozzles in the RML ceiling. This system is automatic or manual with flow limited to 1950 cfm.

2.24 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.

2.25 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 reduce 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 check-out of equipment and as a sound monitor during operations. A second intercom connects the control and setup areas with the remainder of the building.

2.26 Auxiliary Facilities. The following facilities are housed in the examination area: 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).

2.27 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 are 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 four-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 (Figure 3).

V. OPERATING PHILOSOPHY AND TEST PROCEDURES

The SNAPTRAN 2/10A-1 and -2 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.

1. ORGANIZATION

1.1 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 five major subdivisions of the AED are Administration, Engineering, Operations, Nuclear and Chemical Technology, and Nuclear Safety Technology. The Nuclear Safety Technology Organization has the responsibility for all reactor safety testing within the AED. That portion of the safety testing which involves the SNAPTRAN 2/10A-1 and -2 programs will be carried out by the STEP Project (Engineering and Test). The relationship of the STEP Project to other branches of the AED is shown in Figure 35.

1.2 STEP Organization

The STEP Organization is shown in Figure 36. 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.

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

1.22 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.

1.23 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.

1.24 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.

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 ATOMIC ENERGY DIVISION
 ORGANIZATION OF THE ATOMIC ENERGY DIVISION

Fig. 35 AED organization.

Fig. 36 STEP organization.

1.3 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.

2. OPERATING PHILOSOPHY

Since the objectives of the SNAPTRAN 2/10A-1 and -2 experimental safety program necessitates operation of the reactors under conditions normally considered hazardous, administrative controls must be relied upon to minimize the probability of nuclear incidents, to ensure 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.

3. PLANNING AND APPROVALS

The method of operation and scope of the tests to be performed on the SNAPTRAN 2/10A-1 and -2 reactors necessitate 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 approval has been obtained from the Spert-STEP Safeguard Committee, the TSP will be forwarded to the Assistant Manager, Nuclear Safety Technology for approval. Following approval by the Assistant Manager, Nuclear Safety Technology, the copies will be transmitted to AEC-ID 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 and -2 programs. These basic procedures are discussed below.

4. NUCLEAR TEST PROCEDURES

4.1 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 facts 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 nonnuclear 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.

4.2 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.

4.3 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.

4.4 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 people not directly concerned with the immediate operation of the test will evacuate through the T&E (Post 707) area gate, notifying the security 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. At this time, the guard at Post 707 will be notified that no more entry to the reactor area will be permitted.

(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 IET 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) A health physicist will leave the test cell last, closing the tunnel shielding doors, checking the shielding doors in the coupling station, and verifying that the passageways are cleared of personnel. He will then lock the access door behind him.

(4) The health physicist will move Guard Post 705 from the security area gate to the turnaround room and will then check the security area, including all outside buildings, to verify that all personnel have been evacuated.

(5) The health physicist, upon completion of the security area check, will obtain the names of all personnel remaining in the control and equipment building. He will verify, via telephone, that the list of people in the control and equipment building checks with the record maintained by the guard at Post 707 and will also verify that the test area has been cleared.

(6) 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.

(7) The health physicist will report to the responsible supervisor in the control room and transfer the health physics key for the console power switch to the reactor operator.

4.5 Reactor Operation

When all pre-operational procedures have been completed, including the routine test area evacuation, the reactor is ready for start-up. 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 six-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.

4.6 Reentry 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 sub-critical and no foreseeable events will cause it to go critical, reentry operations will be permitted.

Reentry 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 reentry 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 park-
ing area. 10 mr/hr to 10 r/hr

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

(3) One portable 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 SNAPTRAN tests will also provide data for deciding upon the proper reentry procedure. This program is discussed in Section III-6.

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
- (4) A preentry 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 items (3) and (4) above. A decision will be made on reentry time limits and protective clothing, respiratory protection, and personal dosimetry requirements. After consultation with the Nuclear Test Section Group Leader to ensure that the reactor is critically safe, reentry will be allowed, but only when accompanied by a health physicist. Phillips Petroleum Company Radiation Protection Guides will be followed at all times.

Following the destructive test, when the radiation and safety hazards have been evaluated and it has been resolved that personnel may enter the area, the first reentry 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 members 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 pad and surrounding area.

The data obtained by the first reentry team and the roving manipulator will then be used to determine when and how cleanup procedures may be undertaken. A second reentry 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.

4.7 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 reentry 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 reentry into the immediate reactor area is safe, and what protective apparel and precautions must be used.

VI. HOT SHOP OPERATIONS AND TRANSPORT OF THE REACTOR

1. TRANSPORT OF THE REACTOR

1.1 Responsibilities and Approval for Movement

To ensure the safety of personnel and the test package during transport, the reactor must be maintained in a safe nuclear state at all times. The main factors to be considered are safe configurations of the control drums and the exclusion from around the reactor of any material that might act as a neutron reflector and/or moderator. Primary responsibility for preparation of the reactor for transport resides with the Nuclear Test Section. Nuclear surveillance is provided by the Experiments Section and radiological monitoring is the responsibility of the TAN Health Physics group.

Prior to moving the test package from the IET, permission to place the reactor in the hot shop will be obtained from the Metallurgy and Hot Cells Branch, the Spert-STEP Safeguards Committee, and/or the Nuclear Safety Committee, and ID will be made cognizant of the impending move. Once the test package has reached the examination area, responsibility for it will be assumed by the Metallurgy and Hot Cells Branch.

1.2 Preparations for Transport

Procedures for movement of the test package will include (a) movement of the package during daylight hours only and with the reasonable expectation of completing the movement prior to nightfall; (b) predicted wind velocities not to exceed 25 mph; (c) no precipitation expected for at least four hours and the long range forecast prediction of no extreme weather conditions for at least 24 hours; (d) an exclusion area around the reactor to be maintained during transport. The exclusion area for direct radiation hazards will be a circle, the radius of which will be established by radiological considerations as discussed in Section VII-4. If the possibility of radioactive contamination exists, an exclusion area 500 feet on each side of the tracks will also be maintained until the area has been surveyed and decontaminated; (e) the test package will be transported using the shielded locomotive at a speed not to exceed 3 mph; and (f) a waterproof weather shield will be placed over the dolly to prevent water from coming in contact with the reactor in the event precipitation occurs.

Fuel particles and radioactive debris in the test area and on the reactor dolly following the partial damage and/or destructive test will be picked up and placed in critically safe containers which, in turn, will be placed on the dolly for movement to the hot shop. Decontamination of the dolly and the railroad tracks will be as thorough as possible prior to movement of the test package to minimize the spread of contamination.

Under normal circumstances, that is barring a component failure, the control drums are maintained in the full-out position by the action of the scram springs. However, to ensure a safe configuration of the control drums during transport of the reactor, the control drums will either be removed from the test package or pinned in the full-out position, depending on the radiation levels around the reactor existing at the time the transfer is required. In the unlikely

event that one or more of the control drums are stuck in the reactivity region, no movement of the reactor will be attempted until the situation has been corrected and the drums either removed or pinned.

Although movement of the test package will not take place unless the weather forecast is favorable, an umbrella-type weather cover has been provided for covering the reactor while in transit in the unlikely event a component failure on the test dolly or locomotive immobilizes the system between the IET and the hot shop. When locked in place over the test dolly, the cover prevents precipitation from falling on or in the reactor. The cover has been designed to withstand gusts of wind up to 75 mph.

1.3 Reactor Transport

During transport, the radiological and nuclear state of the reactor will be monitored by detectors mounted on the end of the locomotive nearest the dolly with readout in the cab of the locomotive. To assess the information from the detectors, a health physicist, a surveillant physicist, and a reactor operator will ride in the cab with the locomotive operator.

2. HOT SHOP OPERATIONS

Provisions for remote assembly and disassembly have been incorporated in the design of the SNAPTRAN 2/10A-1 and -2 reactor test packages [14] in the event the reactor becomes too radioactive for direct contact maintenance and fuel inspection. However, because of the close proximity of adjacent components, the large number of control and instrumentation cables to be connected and disconnected, and the close tolerances of mating parts, remote operations on the test package will be extremely difficult. In the event remote operations on the test package become necessary, these operations will be limited to removal of the loaded reactor vessel from the test package and unloading the fuel from the vessel. Any further operations on the package will be accomplished by direct contact.

2.1 Hot Shop Operating Procedures

The hot shop operations will fall into two classifications: (a) operations involving sufficient fuel in the reactor to constitute a critical mass, and (b) operations involving fuel in less than critical mass quantities. The first case requires review and approval of operating procedures by the Spert-STEP Safeguards Committee and the Nuclear Safety Committee and the second case requires review and approval by the Nuclear Safety Committee only. Prior to the start of any operation on the reactor in the hot shop, a qualified individual or group of individuals from the Nuclear and Chemical Technology organization will be assigned the responsibility for nuclear surveillance.

2.2 Nuclear Safety Criteria

The SNAPTRAN 2/10A-1 and -2 reactors have under-moderated, high leakage cores and as such are extremely sensitive to the presence of moderating (hydrogenous) materials within the core and to the proximity of any materials, which act as neutron reflectors, around the outside of the core. Because of this sensitivity to the presence of foreign materials, the following general safety rules

will be observed while handling the fuel: (a) all water or other hydrogenous material will be excluded from the area in which an assembly of fuel is located, (b) no material will be placed around the periphery of any assembly of fuel, and (c) all fuel will be stored in critically safe containers except during examination, at which time no more than five rods will be worked on at a time.

The critical mass of the flooded and immersed reactor (water in the core and surrounding reactor vessel) is about 3.1 Kg of U-235 ($\approx$ 24 fuel rods) and for the immersed reactor (water outside the reactor vessel only) about 4.35 Kg of U-235 ($\approx$ 34 fuel rods). The flooded reactor, in the absence of an external reflector, will not go critical. The fully loaded reactor, with the external reflector in place, will be greater than five dollars supercritical when flooded and immersed.

The fuel containers are modeled after the fuel shipping containers designed by Atomics International. Calculations have been made for these shipping containers which indicate that when loaded with five fuel rods each and placed in a slab geometry on 30-inch centers they are in an infinitely safe geometry even when immersed and flooded with water. During cleanup and examination operations, fuel outside the reactor vessel will be kept in the fuel recovery containers. The containers are in two sizes, one of which will hold five intact fuel rods and the other which will hold the equivalent of five fuel rods in the form of small particles. These containers must be kept on a 30-inch center-to-center spacing.

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 the 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. Care must be taken to prevent the insertion of any 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.

VII. SAFETY ANALYSIS

1. 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.

1.1 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 [15]. Five fuel rods will be shipped per shipping container. The container design is shown in Figure 37.

Fig. 37 Bird cage assembly for fuel shipping and storage.

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 V-1) and will not be exceeded or changed without prior written approval.

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.

1.2 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 similar to that used for new fuel. The maximum allowable amount of fissionable material per container will be reduced to an amount equivalent to five fuel rods. Transfer of the test package, which may contain damaged fuel, from IET to TSF following the destructive test is discussed in Section VI.

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

1.3 Neutron Reflecting Materials

The SNAP 10A/2 reactor is a high leakage core (≈ 40 percent 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 could 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 V.

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-2, 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.

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. Estimates of the reactivity effects of various configurations of water and beryllium reflector around the core are summarized in Table VII-1. Of the reactivity worths presented in Table VII-1, the infinite water and the fixed beryllium reflected cases have been experimentally verified as part of the SNAPTRAN program.

TABLE VII-1

SUMMARY OF REACTIVITY WORTH OF
VARIOUS REFLECTOR AND MODERATOR CONFIGURATIONS

	Bare Core	Water Filled Core	Water Reflector on End of Core	Water Reflector around Core	Fixed Beryllium Reflector Pieces in Place	Fixed Beryllium Reflectors Plus Beryllium Drums Fully Inserted
Reactivity worth (\$)	---	+\$10.00	+\$1.50	+\$23.60	+\$15.00	+\$24.00
Reactivity of core with the indicated reflector	-\$20.00	---	---	+ \$3.60	---	+ \$4.00

2. TEST PROGRAM HAZARDS

2.1 Introduction

The SNAPTRAN 2/10A-1 and -2 safety test programs include reactor kinetic tests. The factors of importance in evaluating the reactor dynamic behavior and the hazards attendant to conducting the kinetic tests have been considered. The operational and system failure problem, 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 quantities 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. In the SNAPTRAN reactor the "shutdown mechanisms" 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.

2.2 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.

2.21 Operator Error

(1) Start-up Accident. In order to prevent a start-up 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 reactor period of 10 sec or a power level of up to 1100 watts. Two channels of instrumentation capable of providing period or level scrams have been supplied. In addition, a third neutron level channel and four channels of start-up instrumentation (B-10 pulse counters) are provided. Key actuated interlocks have been included on the control console to prevent an unauthorized reactor start-up and assist in the enforcement of administrative controls by ensuring that supervision is present during start-up. 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 start-up 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 start-up accident analysis in which it was assumed that reactivity was added at the maximum ramp addition rate ($0.16\$/\text{sec}$), and that the period level scram failed to operate. This calculation used the IREKIN reactor kinetics code with a feedback coefficient of $0.0008\$/^{\circ}\text{F}$, which appears to be conservative on the basis of present experimental and theoretical correlations and an initial power of 1 MW, which is typical of starting conditions for transients. The resultant energy release was found to be less than 2 MW-sec for any total scram time less than 310 msec. The total scram time has been determined to be 250 msec or less; thus the energy release from such a start-up accident is inconsequential.

(2) 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 drums(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.

(3) 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-4, 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 keyswitch 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.

(4) 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 keyswitches under the control of the Nuclear Test Section and Health Physics. 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 keyswitch be in his possession. In addition to keyswitch control, scram actuators will be located in the test cell.

2.22 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.

(1) Failures When Reactor is Shut Down

(a) 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 the scram system 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.

(b) Drum Drive Malfunction. Unintended insertion of drums by the motor drives requires simultaneous malfunctions in (a) the two control power keyswitches, (b) the spring-return-to-neutral insert-withdraw switch, (c) the motor selector switches, (d) the scram reset switch, and (e) 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 (a) two series-connected, solenoid-operated, normally closed, inlet valves on each drive, (b) 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 (c) 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.

(2) Failures During Operation

(a) Scram System Failure. Operating procedures will include scram check-out on each drum prior to every start-up. Each drum will be driven to the position of maximum reactivity from which it will be scrambled. 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 scrambled at all times.

(b) 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.

(c) 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 start-up channels, power level channels and safety circuits, drum position indicators, and core temperature channels. In addition, redundancy is provided for all these measurements; ie, there will be four completely independent start-up 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 to 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.

2.23 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, 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 and -2 reactors, 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, 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 than these interrelated mechanisms may, therefore, appear as the reactor undergoes progressively more severe excursions.

The evaluation of the hazards attendant to conducting the kinetics tests has been based on the calculations discussed in this section. The calculations have been made for a range of step and impulse reactivity inputs using the IREKIN code [16] which solves the space independent reactor kinetics equations for power, energy, average core temperature, inverse period, and reactivity as a function of time for an arbitrary reactivity driving function. The reactor model used in IREKIN has a linear heat capacity and a feedback or prompt temperature coefficient independent of temperature.

The axial expansion contribution to the prompt temperature coefficient has been calculated to be approximately $-0.05\phi/^{\circ}\text{F}$, independent of temperature. The magnitude of the spectral contribution depends on the nature of the hydrogen binding in the zirconium-hydride lattice, which is not completely understood at present. However, calculations indicate that the spectral contribution is certainly negative. Measurements of the total prompt temperature coefficients of the S2DR machine, which has the same core composition as SNAPTRAN 2/10A-1 and -2, indicate that the magnitude of the net prompt temperature coefficient is not less than $-0.05\phi/^{\circ}\text{F}$. Figures 38 and 39 show results for an IREKIN 90¢ step with various feedback coefficient values and for a 90¢ experimental step. From these results it appears that a prompt temperature coefficient near $-0.09\phi/^{\circ}\text{F}$ gives the best correlation to the experimental results.

In the analysis of the step and impulse reactivity inputs, the prompt temperature coefficient was conservatively taken to be $-0.08\phi/^{\circ}\text{F}$ over the complete core temperature range, and no provision in the model was made for hydrogen loss. Table VII-2 presents the results of the IREKIN calculations as a tabulation of peak power, energy release to peak power, and average core temperature

Fig. 38 Power versus time from IREKIN calculations for $\$0.90$ steps with temperature feedback coefficients of 0.0008, 0.0009, and 0.0010 $\$/^{\circ}\text{F}$.

Fig. 39 Energy versus time from IREKIN calculations for $\$0.90$ steps with temperature feedback coefficients of 0.0008, 0.0009, and 0.0010 $\$/^{\circ}\text{F}$.

TABLE VII-2

RESULTS OF REACTOR TRANSIENT CALCULATIONS

Reactivity Input	Peak Power (MW)	Energy to Peak Power (MW-sec)	Average Core Temperature Rise to Peak Power (°F)
\$0.20 STEP	0.0181	2.0	190
\$0.40 STEP	0.115	3.2	294
\$0.60 STEP	0.400	4.4	390
\$0.80 STEP	1.10	4.3	381
\$1.00 STEP	4.50	0.64	64
\$1.20 STEP	304	3.0	275
\$1.65 STEP	3660	11.4	850
\$2.20 Impulse (1-Drum)	0.0037	8.0×10^{-6}	0
\$3.00 Impulse (2-Drums)	0.020	2.8×10^{-5}	0
\$3.50 Impulse (2-Drums)	3.4	0.0054	0.5
\$3.75 Impulse (2-Drums)	120	0.17	16
\$4.00 Impulse (2-Drums)	3800	5.2	440

rise at peak power for step reactivity insertion of \$0.20 to \$1.65 and for impulse insertions of \$2.20 to \$4.00 in which the starting power was 1 mW in each case. Figures 40 through 43 present power and energy versus time for several typical step and impulse transients.

The model used with the IREKIN code to describe a destructive transient (SNAPTRAN 2/10A-2) is based on core behavior during the SNAPTRAN 2/10A-3 destructive test and the SNAPTRAN 2/10A-1 kinetic testing series. In addition to the feedback model employed to simulate the nondestructive testing series, three additional effects were included. Feedback proportional to temperature was halted at various temperature limits in excess of 2000°F. A mode of disassembly resulting from a constant pressure acting radially was initiated at a temperature threshold above 2000°F. A time delay was included between the time the temperature reached the threshold and the initiation of disassembly.

The limit on temperature feedback arises as an approximation to the expected reactivity effect of hydrogen evolution. The temperature at which significant evolution can occur for periods of 250 μ sec is not well known. Therefore, the expected range has been bracketed by calculations at 2200 and 3000°F.

Fig. 40 Typical \$0.60 step IREKIN calculation.

Fig. 41 Typical \$1.20 step IREKIN calculation.

Fig. 42 Typical \$3.75 impulse IREKIN calculation.

Fig. 43 Typical \$4.00 impulse IREKIN calculation.

The disassembly behavior is assumed to result from general disintegration of the fuel. The pressures required internal to the fuel are about 20,000 psi for an explosion along dislocations to about 650,000 psi for a lattice disintegrating explosion. The temperature range indicated by these pressures is from 1950 to 2600°F. To bracket the temperature range, calculations were made for thresholds of disassembly at 2000, 2300, and 2700°F average core temperature. The rate of disassembly was assumed to result from a 20,000 psi internal pressure manifesting itself as a 3000 to 10,000 psi pressure radially accelerating the full mass of the reflector plus one-half the mass of the fuel. The worth of the radial expansion has been calculated as about 10 dollars/inch. A time delay after the attainment of the threshold of 100 μ sec was included to account for the finite propagation time of the core explosion. Figure 44 shows an example calculated for a destructive threshold of 2300°F.

The calculated total nuclear energy releases are shown in Table VII-3. The initial period in these calculations was .225 μ sec. The lifetime of 6.7 μ sec is a lower limit and leads to an upper limit on the energy release with respect to prompt neutron lifetime. The expected ranges of parameters are:

3000 psi	effective in-core pressure at disassembly	10,000 psi
2200°F	limit on temperature feedback	3000°F
2000°F	disassembly threshold	2300°F

For this set of parameters, the expected energy release is 175(+75, -50) MW-sec. If disassembly is not initiated until lattice-disintegration pressures are reached in the fuel, the upper limit on energy release is 300 MW-sec.

Fig. 44 IREKIN calculation of power, energy, and reactivity for a SNAP destruct test.

TABLE VII-3

CALCULATED DISTRUCTIVE TEST ENERGY RELEASES

$$\ell = 6.7 \text{ } \mu\text{sec}$$

Case Number	Effective In-Core Pressure	Feedback Limit	Disassembly Threshold	Peak Power (GW)	Total Energy (MW-sec)
1	10,000 psi	2200°F	2000°F	220	130
2	10,000 psi	2200°F	2300°F	260	160
3	10,000 psi	2200°F	2700°F	330	210
4	10,000 psi	3000°F	2000°F	200	120
5	10,000 psi	3000°F	2300°F	220	140
6	10,000 psi	3000°F	2700°F	250	170
7	3,000 psi	2200°F	2000°F	280	210
8	3,000 psi	2200°F	2300°F	330	250
9	3,000 psi	2200°F	2700°F	410	300
10	3,000 psi	3000°F	2000°F	225	180
11	3,000 psi	3000°F	2300°F	250	200
12	3,000 psi	3000°F	2700°F	290	235

3. RADIOLOGICAL ANALYSIS

This section presents the potential radiological hazards that may prevail during and following the SNAPTRAN 2/10A-1 and -2 test programs.

In general, the planned program will consist of three phases: (a) a series of nondamaging power transients, (b) one or more power transients resulting in partial damage to the core, and (c) a power transient resulting in core destruction. The nondamaging phase of the program will consist of power transients producing a maximum accumulated total energy release of 20 MW-sec per week for not more than 28 weeks. The partial damage phase will consist of one or more power transients involving a reactivity insertion of sufficient magnitude to damage one or more of the fuel rods. It is postulated that a maximum of seven central fuel rods will be damaged and that the total energy release will not exceed 20 MW-sec. Following the partial damage tests a new reactor with a clean core will be installed in the IET and the destructive test will be performed. This test will consist of a reactivity insertion of sufficient magnitude to cause complete destruction of the core.

The cases for which the radiological hazards have been considered, both planned and unplanned events, are as follows:

- (1) Nondamaging Tests: A series of transient tests with no release of fission products.

- (2) Partial Damage Tests: Transient tests conducted under strict meteorological control culminating in limited damage to the core including the possibility of fission product release.
- (3) Planned Destructive Test: A planned destructive excursion conducted under strict meteorological control and resulting in complete destruction of the reactor.
- (4) Unplanned Destructive Excursion: An accidental nuclear excursion resulting in complete destruction of the reactor under lapse meteorological conditions.
- (5) Examination Area Hazards: The hazards in the TSF Hot Shop which are associated with the "hot" assembled reactor, or the remains of the reactor after destruction.

It should be pointed out that in areas involving an uncertainty such as predicting precise energy releases, the fission product behavior following core destruction, etc, conservative assumptions resulting in upper limit values for the radiological hazards have been employed in the calculations. The specific assumptions are discussed in each case considered. The analytical techniques and conclusions are presented at the end of this section.

3.1 Nondamaging Tests

It is assumed that the nondamaging power transients will not involve a release of fission products to the atmosphere. Therefore, the potential hazards that have been considered in this case are from direct radiation in the form of neutrons and gamma rays emanating from the core and structural material during and following the power transients.

The radiation hazards during the transients which have been considered are those produced by the following sources: (a) prompt fission gammas and very short-lived fission product gammas, (b) prompt gammas from neutron absorption in structural materials, (c) neutrons escaping from the core, and (d) neutrons produced by the γ, n reaction in the beryllium reflector surrounding the core. Calculations indicate that sources (b) and (d) are negligible. The total dose from sources (a) and (c) at selected locations in the TAN complex are presented in Table VII-4. The calculations are based on the conservative assumptions that the core will undergo a 20 MW-sec energy release (the maximum total energy release for a one week period) in a single transient and that there is no self attenuation in the core.

Following the power transients, the potential hazards are from (a) gamma rays from fission product decay, (b) gamma rays from induced activity in the structural materials, and (c) neutrons from the γ, n reaction in the beryllium reflector. Sources (b) and (c) were determined to be negligible. The dose rates from source (a) at selected locations in the TAN complex are also presented in Table VII-4. These are the maximum dose rates postulated to prevail after the final 20 MW-sec transient of a series of one 20 MW-sec transient per week for 28 weeks.

It is of interest to note that the contribution to the total dose rate from the previous 27 transients is negligible until approximately 12 hours after the last

TABLE VII-4

TOTAL DOSE FROM FINAL 20 MW-SEC
TRANSIENT AND DOSE RATE AFTER SHUTDOWN

Location	Distance from Reactor (meters)	Neutron and Gamma Dose During Transient (rem)	Gamma Dose Rate after 28th Transient	
			Time after Shutdown (hrs)	Dose Rate (rem/hr)
Inside reactor building	5	1.0×10^4	1	22
			12	0.93
			24	0.43
			48	0.23
	10	2.2×10^3	1	5.0
			12	0.24
			24	0.11
			48	6.0×10^{-2}
IET control and equipment building (Control Room) ^[a]	12	5.0×10^{-3}	1	1.0×10^{-3}
Security fence around IET ^[b]	50	50	1	0.19
			12	7.8×10^{-3}
			24	3.6×10^{-3}
			48	1.9×10^{-3}
End of tunnel at IET	200	0.60	1	4.5×10^{-3}
TGF area	1500	1.0×10^{-5}	1	1.0×10^{-3}

[a] Shielded from reactor with earth and concrete.
[b] At the nearest point.

transient. At this time the previous transients contribute approximately five percent to the total dose rate. At 48 hours after the final transient, the previous operating history contributes approximately 35 percent to the total dose rate.

The analytical models used and the detailed assumptions made for both the neutron and gamma dose and dose rate calculations are presented in section VII-3.6.

As shown in Table VII-4, the radiation levels during and following a 20 MW-sec transient are sufficiently low to preclude excessive exposures to operating personnel in the shielded control and equipment building and to personnel at other facilities located at distances greater than 1500 meters from the IET facility. The area within 1500 meters of IET, except for the control and equipment building, is considered an exclusion area during reactor operation. Following a power transient, admittance will not be permitted within the security fence until the area has been properly surveyed and cleared by a health physicist.

3.2 Partial Damage Tests

As testing of the SNAPTRAN 2/10A-1 reactor nears completion, partial damage tests will be performed. Such tests may lead to deformation of the fuel rods, rupture of one or more rods, or reactor disassembly. In the case in which damage is limited to fuel rod deformation, fission products will not be released; therefore, the radiological hazard is due only to direct radiation. In the event of fuel rod rupture, gaseous and particulate fission products may be released to the atmosphere, thereby presenting a potential hazard from fission products in the cloud as well as from direct radiation. The potential hazards from the released fission products are considered in this section of the report. The hazards from direct radiation will be no greater than those presented in Table VII-4 for the nondamaging tests. In the event of reactor disassembly, the potential radiological hazards are proportional to the energy release and to the percentage fission product release. The radiological analysis for the partial damage case was based on the following assumptions:

- (1) Operating history of 20 MW-sec energy release for each of 28 consecutive weeks
- (2) Rupture of the seven central fuel elements occurs during the final 20 MW-sec transient
- (3) One hundred percent of the activity of the seven central elements is released to the atmosphere. This activity constitutes 25 percent of the total activity of the core [17]
- (4) Fission product release occurs under controlled meteorological conditions with a mean wind velocity of 15 mph from a sector 180° to 240° (0° = true north)
- (5) Cloud depletion due to fallout constitutes an insignificant reduction in airborne particulate activity
- (6) The deposition gamma dose occurs during a one-year exposure period.

In order to evaluate the radiological hazard resulting from each case of fission activity release, the whole body and thyroid doses are compared with the Atomic Energy Commission's radiation protection standards for individuals and population groups in uncontrolled areas. These standards are present in Table VII-5 and are taken from AEC Appendix 0524 "Standards for Radiation Protection". The limits specified for the average exposure to a population sample are the limits used for evaluating the dosages calculated for partial damage and destructive testing, while the limits for exposure to an individual are applicable for evaluating the hazards associated with an accidental excursion.

Cloud radiation dosages at the site boundary following a 20 MW-sec partial damage test are presented in Table VII-6. Deposition gamma doses are calculated for a one-year exposure period. The ingestion dose is that acquired by a child through the intake of contaminated milk. This dose is a factor of 10 greater for a child than that for an adult because of the smaller critical organ. The ingestion is considered only beyond the site boundary since controlled grazing conditions are, in effect, within the boundary. Analytical techniques employed

TABLE VII-5

STANDARDS FOR RADIATION PROTECTION

Type of Exposure	Exposure to Individual	Average Exposure to Population Sample
Whole body (cloud and deposition), gonads, or bone marrow	0.5 rem/year	0.17 rem/year
Thyroid (inhalation and ingestion) or bone	1.5 rem/year	0.5 rem/year

TABLE VII-6

CLOUD RADIATION DOSAGES
AT SITE BOUNDARY PARTIAL
DAMAGE TESTS -- 20 MW-SEC
ENERGY RELEASE

Exposure	Dose (rem)
Cloud Gamma	1.7×10^{-5}
Deposition Gamma	3.3×10^{-5}
Inhalation to Thyroid	3.5×10^{-5}
Ingestion to Thyroid	4.5×10^{-3}

and fission product activities considered in the calculations are presented in Section VII-3.6.

The whole body dose (cloud plus deposition and the thyroid dose inhalation plus ingestion) are 5.0×10^{-5} rem and 4.5×10^{-3} rem, respectively, which are well below the permissible levels of 0.17 and 0.5 rem. Therefore, it has been concluded that the partial damage tests will not present a significant radiological hazard to the general public.

Radiation exposures to site personnel will be insignificant because all tests in which partial damage is expected will be performed under controlled meteorological conditions with the wind blowing from the on-site facilities. Personnel access to those areas on-site traversed by the radioactive cloud will be prohibited until they have been cleared by Health Physics personnel.

Following the partial damage test, if radiation levels make it necessary, the reactor package will be moved to the TSF hot shop for removal of the used fuel elements. Prior to moving the reactor, proper precautionary steps will be taken to preclude the possibility of an accidental nuclear excursion enroute to the TSF area. Each control drum will either be secured in the out position by inserting the shaft lockpins or will be removed from the package. In addition, as an added precaution, the reactor will be moved while the wind direction is from the TSF area.

While the reactor is being moved, the exclusion distance from the dolly will be 50 meters. Assuming the reactor is not moved until at least 24 hours after the last transient, the maximum dose rate at the exclusion distance is postulated to be approximately 3.6×10^{-3} rem/hr as shown in Table VII-4. This is well below permissible limits.

3.3 Planned Destructive Test

The planned destructive test will be performed with a clean core containing an insignificant quantity of fission products. This test will be performed to model the maximum possible accident involving the SNAPTRAN 2/10A-2 reactor. An energy release of about 200 MW-sec, causing complete reactor disassembly and the release of radioactive materials to the atmosphere, is expected to be produced during the test. The consequences of the destructive test have been evaluated for the following conditions:

- (1) Total energy release of 200 MW-sec
- (2) The total fission product inventory is that produced by the 200 MW-sec excursion
- (3) All fuel elements in the core rupture, releasing one hundred percent of the fission product inventory under controlled (lapse) meteorological conditions.

The potential radiological hazards accompanying the planned destructive test arise from direct radiation during the power burst, direct radiation from fission product decay, and from cloud radiation caused by radioactive materials transported downwind. These hazards have been evaluated on the basis of the above-listed conditions. Hazards associated with transporting the disassembled reactor to the hot shop have also been considered as has the possibility of a second criticality. The results of the evaluation are presented and discussed in this section of the report. Examination area (hot shop) hazards are discussed in greater detail in Section VII-5.5.

3.31 Direct Radiation. The total dose from direct radiation during the 200 MW-sec destructive excursion and the dose rates at various times after destruction for various areas are shown in Table VII-7. The transient dose presented includes the dose from fission neutrons, prompt fission gammas, and short-lived fission product decay gammas. The total dose is considered to be very conservative because the effect of self-absorption from materials in the core was not considered in the calculations.

The gamma dose rates after reactor destruction are from fission product decay gammas and are based on the conservative assumption that all fission products form a line source along the edge of the test pad.

Before the destructive test is initiated, the area inside the obstruction fence will be cleared making certain that all personnel in the area are inside the underground control and equipment building. The maximum dose from the destructive transient to personnel in the control room will be approximately 0.048 rem at the point nearest the reactor. However, this result is considered to be very conservative because of the assumptions employed in the calculations.

Since the direct radiation dose rate beyond the obstruction fence is negligible, it is concluded that the planned destructive test presents no significant hazard from direct radiation to on-site personnel. The hazards from the fission products released to the atmosphere are considered in the next section.

TABLE VII-7

TOTAL DIRECT RADIATION DOSE FROM 200 MW-SEC
DESTRUCTIVE EXCURSION AND DOSE RATE AFTER DESTRUCTION

Location	Distance from Reactor (meters)	Total Transient Dose (rem)	Gamma Dose Rate after Destruction	
			Time after Destruction (hrs)	Dose Rate (rem/hr)
	10	2.2×10^4	1	72
			12	3.0
			24	1.2
			48	0.48
IET control and equipment building (Control Room) ^[a]	12	4.8×10^{-2}	1	5.0×10^{-3}
Security fence around IET ^[b]	50	5.0×10^2	1	2.6
			12	0.11
			24	4.4×10^{-2}
			48	1.8×10^{-2}
End of tunnel at IET	200	6.0	1	5.6×10^{-2}
			2	2.4×10^{-2}
Test area ^[c]	1500	1.0×10^{-1}	1	1.0×10^{-3}

[a] Shielded from reactor with earth and concrete.

[b] At the nearest point.

[c] Area outside of obstruction fence.

Following the destructive test, continuous access to the area inside the security fence, except inside the control and equipment area, cannot be permitted for at least 24 hours or until the area has been cleared by a health physicist. Access to the immediate vicinity of the test pad will be restricted until radiation surveys made by a health physicist indicate the levels are within permissible limits. Calculations indicate that after one week, access to the reactor building can be permitted for approximately 30 minutes before personnel receive 0.3 rem.

3.32 Cloud Radiation. To reduce the radiological hazards to personnel in TAN area and to off-site population, strict meteorological control will be exercised for all tests which could lead to release of radioactive products to the atmosphere. The meteorological conditions established by the Idaho Operations Office of the AEC in conjunction with the U. S. Weather Bureau, are:

- (1) Lapse (unstable) conditions to persist a minimum of three hours after the test to ensure adequate diffusion conditions
- (2) Mean wind direction in the sector between 180 and 240° with allowance of an additional 20° on each side of the sector for short term wind fluctuations
- (3) Wind speed between 10 and 30 miles per hour, the lower limit to ensure reliability of forecasting a persistent wind direction and the upper limit to ensure safe aerial monitoring
- (4) No precipitation in the area in order to maintain good sampling conditions and to prevent localized washout of radioactive material.

Under these meteorological conditions, the passage of the cloud of released materials will be away from all site facilities and will be over an uninhabited area for at least 6.5 miles, thereby keeping the potential hazard to on-site personnel and the general public at a minimum.

The radiological dosages calculated for the planned destructive test were based on the following assumptions.

- (1) One hundred percent of the fission product activity is released to the atmosphere as noble gases, halogens, and solids.
- (2) Release occurs under lapse conditions with a mean wind speed of 7 meters/second (15 mph).
- (3) Cloud depletion due to fallout is insignificant in the reduction of airborne activity.
- (4) Deposition gamma dosages are considered for a one-year exposure period.
- (5) The ingestion dose is considered for 100 percent grazing conditions during the summer months.

Cloud radiation dosages at the site boundary for the planned 200 MW-sec excursion are presented in Table VII-8. The ingestion dose to the thyroid is that for a child while the inhalation dose is that received by an adult. The ingestion dose to a child is approximately 10 times greater than that which an adult would receive.

As can be observed from Table VII-8, the whole body and thyroid doses beyond the site boundary ($>10^4$ meters) are 8.4×10^{-4} rem and 4.5×10^{-2} rem, respectively. These values are well below the permissible 0.17 rem/year and

TABLE VII-8
CLOUD RADIATION DOSAGES
AT SITE BOUNDARY FOR
PLANNED DESTRUCTIVE TEST
200 MW-SEC ENERGY RELEASE

<u>Exposure</u>	<u>Dose (rem)</u>
Cloud Gamma	6.6×10^{-4}
Deposition Gamma	1.8×10^{-4}
Inhalation to Thyroid	1.3×10^{-3}
Ingestion to Thyroid	4.4×10^{-2}

0.50 rem/year limits for uncontrolled populated areas; therefore, it is concluded that the planned destructive test will not present a radiological hazard to the general public.

Additional calculations have also been made to determine the doses as a function of distance from the IET facility. The results of the calculation are presented in Table VII-9. Various on-site and off-site locations and their respective distances from the IET test area are given in Table VII-10. The analytical techniques used in determining the dosages, and the fission product activities used as the source term in the mathematical expressions are presented in Section VII-4.6.

3.33 Transportation to Examination Area. The major radiation hazard associated with transporting the reactor debris to the examination area after the destructive test is the direct gamma radiation from fission product decay. The calculated decay dose rate is based on the assumption that all the fission products are on the dolly with no shielding except air.

The gamma dose rates at various locations along the dolly track during transport of the reactor to the examination area are presented in Table VII-11. The dose rates presented are based on the assumption that the reactor will be moved 24 hours after the destructive test.

While the reactor is being transported to the examination area, a distance of at least 50 meters on each side of the four-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 (building 607), except the operators in the shielded hot shop control room, may be evacuated to a position behind the earth embankment until the reactor is inside the hot shop. Aside from the areas mentioned, all other areas will be considered as unlimited access areas.

3.4 Secondary Criticality

Possible secondary criticality hazards during or as a result of the cleanup 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 (Mobot) under the command and TV observation of an operator located at a 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

TABLE VII-9

CLOUD RADIATION DOSES FROM PLANNED DESTRUCTIVE TEST
(200 MW-sec Energy Release)

Distance from Reactor (meters)	External Gamma Dose		Ingestion Dose to Child (rem)
	Cloud Gamma (rem)	Deposition Gamma (rem)	
5×10^1	8.3×10^2	4.4×10^0	
10^2	1.5×10^2	1.1×10^0	
5×10^2	2.4×10^0	5.1×10^{-2}	
10^3	3.9×10^{-1}	1.4×10^{-2}	
5×10^3	4.4×10^{-3}	6.5×10^{-4}	
10^4 [a]	6.6×10^{-4}	1.8×10^{-4}	4.4×10^{-2} [b]
5×10^4	5.4×10^{-6}	7.8×10^{-6}	1.0×10^{-2}
10^5	7.1×10^{-7}	1.9×10^{-6}	4.0×10^{-3}

Distance from Reactor (meters)	Inhalation Doses						
	Body (rem)	Thyroid (rem)	Muscle (rem)	Lung (rem)	Liver (rem)	Kidney (rem)	Bone (rem)
5×10^1	4.9×10^{-2}	7.8×10^0	1.2×10^{-5}	7.6×10^{-1}	1.0×10^{-1}	9.6×10^{-2}	5.0×10^{-1}
10^2	1.6×10^{-2}	2.4×10^0	4.6×10^{-6}	2.5×10^{-1}	2.9×10^{-2}	2.8×10^{-2}	1.8×10^{-1}
5×10^2	1.2×10^{-3}	1.7×10^{-1}	9.6×10^{-7}	1.9×10^{-2}	1.7×10^{-3}	1.8×10^{-3}	1.4×10^{-2}
10^3	4.1×10^{-4}	5.4×10^{-2}	4.8×10^{-7}	6.3×10^{-3}	5.3×10^{-4}	6.2×10^{-4}	4.8×10^{-3}
5×10^3	3.1×10^{-5}	4.0×10^{-3}	4.7×10^{-8}	4.8×10^{-4}	3.8×10^{-5}	6.1×10^{-5}	3.5×10^{-4}
10^4 [a]	9.7×10^{-6}	1.3×10^{-3}	1.4×10^{-8}	1.5×10^{-4}	1.2×10^{-5}	1.9×10^{-5}	1.1×10^{-4}
5×10^4	5.5×10^{-7}	8.8×10^{-5}	8.0×10^{-10}	8.4×10^{-6}	7.0×10^{-7}	1.1×10^{-6}	6.0×10^{-6}
10^5	1.4×10^{-7}	2.1×10^{-5}	2.3×10^{-10}	2.2×10^{-6}	1.9×10^{-7}	3.1×10^{-7}	1.6×10^{-6}

[a] Site boundary

[b] Controlled grazing conditions for distances of $< 10^4$ meters.

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 reactor and debris.

TABLE VII-10

DOWNWIND LOCATIONS OF RADIOLOGICAL IMPORTANCE

Location	Distance from Reactor (meters)	Distance from Reactor (miles)
<u>On-Site</u>		
TSF	2.0×10^3	1.25
Fenced area around TSF	1.5×10^3	1.0
End of tunnel IET	2.0×10^2	0.13
Security fence around IET	5.2×10^1	0.03
LCRE	2.5×10^3	1.5
NRF	2.9×10^4	18
Site Boundary	1.0×10^4	6.5
<u>Off-Site</u>		
Monteview	1.5×10^4	9.5
Mudlake	1.8×10^4	11
Terreton	2.1×10^4	13
Howe	2.6×10^4	16
Winsper	3.2×10^4	20
Atomic City	4.8×10^4	30
Small	5.1×10^4	32
Dubois	5.1×10^4	32
Arco	5.6×10^4	34
Menan	5.6×10^4	35
Idaho Falls	6.7×10^4	42

TABLE VII-11

RADIATION LEVELS WHILE TRANSPORTING
REACTOR TO TSF FOLLOWING DESTRUCTIVE TEST

Location	Approximate Distance from Track (meters)	Gamma Dose Rate (rem/hr)
	10	1.2
	50	4.4×10^{-2}
Point on Taft Road nearest the track	275	$< 3.0 \times 10^{-4}$
Administration building	300	$< 2.0 \times 10^{-4}$
Nearest point to security fence around administration building -- not shielded by the earth embankment	150	$< 3.0 \times 10^{-3}$
Unshielded part of Building 607	20	0.33
Inside shielded locomotive		$< 1.0 \times 10^{-3}$

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-inch 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-inch 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 Figure 45). 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.

3.5 Unplanned Destructive Excursion

The possibility of an accidental destructive excursion occurring while the reactor is shut down or during nontransient operation is not considered credible due to the many protective measures built into the reactor control system and the administrative controls as described in Section VII-2. However, an analysis has been performed to investigate the maximum consequences of an accidental excursion occurring during transient testing. All transient tests in which fission products could be released to the atmosphere will be conducted under strict meteorological control with the restriction that lapse conditions exist. The analysis is based on the following assumptions:

- (1) A 28-week series of transients (one 20 MW-sec transient per week) culminates with the final transient producing an

Fig. 45 Contaminated waste trench and strainer.

energy release of 220 MW-sec resulting in complete reactor destruction with 100 percent of the fission products released.

- (2) The accident occurs at the IET facility during weak lapse meteorological conditions.

3.51 Direct Radiation. The total dose from the direct radiation during a 220 MW-sec destructive excursion at the IET facility and the dose rates at various times after destruction for various areas are presented in Table VII-12. The transient dose presented includes the dose from fission neutrons, prompt fission gammas, and short-lived fission product gammas. The dose rates after destruction are from fission product decay gammas and are based on the conservative assumption that all fission products form a line source along the edge of the test pad.

Since no personnel will be inside the obstruction fence during reactor operation, except in the shielded control and equipment building, it is concluded that the 220 MW-sec excursion presents no significant hazards from direct radiation to on-site personnel.

3.52 Cloud Radiation. The major radiological hazard associated with an unplanned nuclear excursion is that due to the radioactive cloud which carries radioactive gases and particulates downwind to either on-site facilities or to off-site inhabited areas. The total dose results from external exposure to the cloud and deposited materials and from the internal dosage from radioactive materials taken into the body through inhalation and ingestion.

The calculations of dosages resulting from an unplanned destructive excursion are based on the following assumptions:

TABLE VII-12

TOTAL DIRECT RADIATION DOSE FROM 220 MW-SEC
DESTRUCTIVE EXCURSION AND DOSE RATE AFTER DESTRUCTION

Location	Distance from Reactor (meters)	Total Transient Dose (rem)	Gamma Dose Rate After Destruction	
			Time after Destruction (hrs)	Dose Rate (rem/hr)
	10	2.4×10^4	1	80
			12	3.5
			24	1.5
			48	0.62
IET control and equipment building (Control Room)[a]	≈ 12	5.3×10^{-2}	1	5.0×10^{-3}
Security fence around IET	≈ 50	5.5×10^2	1	2.9
			12	0.12
			24	5.2×10^{-2}
			48	2.5×10^{-2}
End of tunnel at IET	≈ 200	6.6	1	6.2×10^{-2}
TSF area[b]	> 1500	$< 1.0 \times 10^{-4}$	1	1.0×10^{-3}

[a] Shielded from reactor with earth and concrete.
[b] Area outside obstruction fence.

- (1) The final transient in the 28 transient series causes complete reactor disassembly with the release of 100 percent of the total fission product inventory to the atmosphere.
- (2) Release occurs under weak lapse conditions with a minimum wind velocity of 2.2 mph (1 meter/sec).
- (3) Cloud depletion due to fallout is insignificant in the reduction of airborne activity.
- (4) The deposition gamma dose is considered for a one-year exposure.
- (5) The ingestion dose is not included in the thyroid dose.

The average wind velocity which occurs under lapse conditions at the NRTS is approximately 7 meters per second^[19]; however, calculations indicate that the thyroid inhalation dose, which is the limiting downwind dose, is greater for lower wind speeds. Therefore, to make the analysis conservative, a one meter per second wind speed was assumed.

The radiological doses at the nearest site boundary ($\approx 10^4$ meters), based on the preceding assumptions, are presented in Table VII-13.

The analytical techniques and fission product activities used for the analysis of the results of an accidental excursion are presented in Section VII-3.6.

TABLE VII-13

CLOUD RADIATION DOSES AT SITE BOUNDARY
RESULTING FROM UNPLANNED 220 MW-sec EXCURSION

External Gamma Doses						
Cloud Gamma Dose (rem)			Deposition Gamma Dose (rem)			
1.9 x 10 ⁻³			3.3 x 10 ⁻³			
Inhalation Doses						
Body (rem)	Thyroid (rem)	Muscle (rem)	Lung (rem)	Liver (rem)	Kidney (rem)	Bone (rem)
2.7 x 10 ⁻⁴	4.2 x 10 ⁻²	4.1 x 10 ⁻⁷	4.2 x 10 ⁻³	3.6 x 10 ⁻⁴	5.3 x 10 ⁻⁴	2.9 x 10 ⁻³

3.53 Accidental Excursion in the Hot Shop. Prior to the planned destructive test, it is not expected that the reactor package will ever have to be moved into the hot shop when the reactor contains fuel. However, in the extremely unlikely event that such a need should arise, additional approvals will be required before such a move is undertaken, and precautionary steps will be taken which make an accidental nuclear excursion in the hot shop incredible.

3.54 Reactor Debris in the Hot Shop. After the reactor debris has been moved to the hot shop following the destructive test, the radiation levels outside of the hot shop are determined to be very low. Assuming all of the fission products form a line source on the dolly, which is moved to the examination area 24 hours after the test, the dose rate through the hot shop walls is calculated to be less than $1 \text{ by } 10^{-3} \text{ rem/hr}$. Therefore, the radiation from the reactor debris in the hot shop present no significant hazard to personnel in the examination area.

3.6 Analytical Techniques

The analytical models used to evaluate the radiological hazards given in the previous sections are presented below.

3.61 Direct Radiation from Fission Product Decay. To determine the gamma dose rate as a function of distance from the reactor after a series of non-destructive transients, a cylindrical source geometry was assumed for distances less than 10 feet and a point source for distances greater than 10 feet. The techniques used are those presented in TID-7004 [20].

Cylindrical source

$$\phi = \frac{B S_V R_O^2}{2 (a + z)} F(\theta, b_2)$$

where

ϕ = gamma flux (γ 's/cm²-sec)

B = buildup factor (dose buildup for water, point source) due to external shielding

S_V = gamma source density (γ 's/cm³-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)

$\theta = \tan^{-1} \left(\frac{h/2}{a+z} \right)$ 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 i^{th} shield material, cm⁻¹

t_i = thickness of the i^{th} shield material (cm)

μ_s = macro-cross-section (energy absorption) of the source material, cm⁻¹

$$F(\theta, b_2) = \int_0^\theta e^{-b_2 \sec \theta} d\theta$$

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 (cm³)

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

Note: The point source was assumed to be at the center of the cylinder. Attenuation in the core was neglected. The shielding materials considered for the point source and cylindrical source were the steel vessel, beryllium reflector, and air.

To determine the gamma dose rate from fission product decay after a destructive excursion, the source geometry was assumed to be a line source for distances less than 500 feet from the test pad and a point source for distances greater than 500 feet. The only attenuation considered was that from air.

Line source

$$\phi = B \frac{S_L}{2\pi a} F(\theta, b)$$

where

$$S_L = \frac{S_0}{L} = \gamma's/cm$$

S_0 = gamma source strength of energy E ($\gamma's/sec$)

L = length of test pad (1.83×10^3 cm)

B = dose buildup factor for water

b = μt for air

$$\theta = \tan^{-1} \left[\frac{1/2L}{a} \right]$$

a = distance from test pad to dose point.

In each case the source strength term was calculated using the gamma energy release rate (MeV/sec) for seven gamma energy groups as determined by a computer program [21]. The computer code accumulates the fission product activity for successive periods of reactor operation and shutdown.

The conversion from gamma flux to dose rate was obtained from TID-7004 [20].

3.62 Direct Radiation During a Transient. The total transient dose includes the dose from fission neutrons, prompt fission gammas, and short-lived fission product gammas. A point source geometry was used to calculate the neutron and prompt gamma contributions. To account for the short-lived fission product gammas, the prompt gamma dose was multiplied by a factor of two [22].

In calculating the neutron dose, no self-attenuation due to core materials was assumed. Only the attenuation from beryllium and oxygen in air was considered. The equation used is similar to the point source equation for gamma calculations except for the S_0 and b_1 terms which are defined as follows:

S_0 = total neutron leakage from the core (neutrons/sec)

$$b_1 = \sum_{i=1}^n \mu_i t_i$$

where:

μ_i = removal cross section (0.1248 cm^{-1} for beryllium [23] and $9.1 \times 10^{-25} \text{ cm}^{-1}$ for oxygen [24]).

The conversion from neutron flux to dose rate was obtained from TID-7004[20].

3.63 Cloud Radiation Doses. Radiological doses, resulting from exposure to the cloud containing radioactive gases and particulates, were calculated for controlled (lapse) meteorological conditions since these conditions will prevail during all reactor tests for which a release of fission products is credible. The meteorological conditions and the analytical techniques used in all cloud radiological calculations are as follows:

(1) Meteorological Conditions: Before presenting the specific numerical values for the meteorological parameters most frequently used throughout the analysis, a definition of each parameter is in order.

n = Sutton's stability parameter corresponding to temperature condition considered

C_y = Sutton's horizontal diffusion coefficient (meters ^{$n/2$})

C_z = Sutton's vertical diffusion coefficient (meters ^{$n/2$})

$\bar{U}$ = mean wind velocity (meters/second).

The analysis of the radiological hazards accompanying partial damage and planned destructive tests has been carried out for controlled (lapse) meteorological conditions in which the following parameters apply [19]:

(a) Sutton's stability parameter, $n = 0.2$

(b) Sutton's diffusion coefficients, $C_y, C_z = 0.35$

(c) mean wind velocity, $\bar{U} = 7$ meters/sec (≈ 15 mph).

Accidental release of fission products could occur only during transient tests, all of which will be conducted under controlled (lapse) meteorological conditions. Assuming weak lapse conditions, calculations of the radiological hazards resulting from an unplanned destructive excursion have been based on the following criteria:

(a) Sutton's stability parameter, $n = 0.2$

(b) Sutton's diffusion coefficients $C_y, C_z = 0.23$ [19]

(c) mean wind velocity, $\bar{U} = 1$ meter/sec (≈ 2 mph).

(2) Mathematical Techniques -- Equations Used. External body exposure arises from the cloud of radioactive material as it passes over the receptor and from radioactive materials deposited on the surfaces over which the cloud travels. The cloud dose is acquired only during passage and thus is of short duration while the deposition dose is acquired over an extended period of time until all deposited substance has decayed to insignificant levels or until the recipient exits from the contaminated area.

Ingestion and inhalation doses constitute the internal dosage. Inhalation doses are acquired through inhalation of radioactive gases, and ingestion doses result from the intake of radioactive matter.

(a) Total Integrated External Cloud Dosage -- Instantaneous Point Source. The total integrated dosage, expressed in the following equation, is the maximum amount of radiation to which a recipient may be exposed as the result of the passage of a cloud of radioactive material [25].

$$TID = \frac{2 Q f(t) C_f}{\pi C_y C_z \bar{U} X^{2-n}} \exp \left[\frac{-h^2}{C_z^2 X^{2-n}} \right]$$

where

Q = initial source activity (curies)

$f(t)$ = correction for radioactive decay

C_f = conversion factor ($2.46 \text{ by } 10^{-1} \text{ rem-meter}^3/\text{curie-sec-ond}$) [26]

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

C_y, C_z = Sutton's diffusion parameters where values are those obtained from Reference 19.

n = Sutton's stability parameter where values are those obtained from Reference 19

X = distance downwind (meters).

For all SNAPTRAN 2/10A-1 cloud dose calculations presented in previous sections, the cloud height was assumed to be zero to give the most conservative dosage for ground level; $Q f(t)$ can be replaced by $Q(t)$ where $Q(t)$ is obtained directly by means of the "Curie" computer program. Therefore, the cloud gamma expression reduces to:

$$TID = \frac{2 Q(t) C_f}{\pi C_y C_z \bar{U} X^{2-n}}$$

(b) Total Integrated Deposition Gamma Dosage. The cloud deposition dose was calculated for a one-year exposure by means of the following expression:

$$D_d = \frac{2 C_f}{\pi C_y C_z \bar{U} X^{2-n}} \int_0^{1 \text{ year}} Q(t)_s V_{g_s} dt + \int_0^{1 \text{ year}} Q(t)_h V_{g_h} dt$$

where

$C_y, C_z, \bar{U}$, and X = as previously defined

$Q(t)_s, Q(t)_h$ = fission product activities for solids and halogens, respectively. Values are obtained from the "Curie" computer program

$(V_{g_s}), V_{g_h}$ = deposition velocity for solids and halogens; V_{g_s} is 0.002 meters/second and V_{g_h} is 0.010 meters/second[19]

C_f = conversion factor $(2.78 \text{ by } 10^{-3} \text{ rem-meter}^2/\text{curie-second})^{[27]}$.

(c) Inhalation Dosages. Inhalation dosages to the thyroid and other body organs were calculated by means of:

$$D_{inh} = \frac{2 B_r f_a K}{\pi C_y C_z \bar{U} X^{2-n} M} \sum_i Q(t)_i \bar{E}_i T_{e_i} + Q(t)_2 \bar{E}_2 T_{e_2} + \dots$$

$$+ Q(t)_i \bar{E}_i T_{e_i}$$

where

$C_y, C_z, \bar{U}$, and X = as previously defined

B_r = breathing rate = $\frac{(3.47 \times 10^{-4} \text{ m}^3)}{\text{sec}}$ [28]

f_a = fraction of activity reaching critical organ [28]

K = conversion factor

= $\frac{(1.6 \times 10^{-6} \text{ ergs/MeV}) (3.7 \times 10^{10} \text{ disintegrations/curie-second})}{(100 \text{ ergs/gram-rad})}$

$Q(t)_i$ = activity of isotope i (curies)

$\bar{E}_i$ = effective energy of isotope i (MeV) [28]

T_{e_i} = effective half-life of isotope i (seconds) [28].

(d) Ingestion Dose. Iodine 131 and 133 isotopes are considered as primary contributors for the ingestion dose. The expressions used for obtaining the ingestion doses are:

$$D_{ing} = (D_{inh_{131}} \times 1200 + D_{inh_{133}} \times 100)$$

where

D_{ing} is obtained by the expression for the inhalation dosage to the thyroid. The numerical values 1200 and 100 in the above equation are the ratios of the ingestion to inhalation dose for I-131 and I-133, respectively. These values were obtained from the results of the AED-IDCERT test [18].

The equation presented applies to a child; the adult dose is a factor 10 less than that for a child because of the smaller critical organ of the child.

(3) Fission Product Activities. The values of $Q(t)$, fission activity at time t , used in the various dose calculations were obtained from the "Curie" computer program. The activities, based on an assumed 100 percent release, are presented in Tables VII-14, -15, and -16.

TABLE VII-14

FISSION PRODUCT INVENTORY (CURIES) PARTIAL DAMAGE TESTS

Time After Release (sec)	Iodine 131	Iodine 133	Noble Gases	Halogens	Solids
1.4×10^0			2.1×10^5	4.2×10^5	8.2×10^5
7.1×10^0			9.9×10^4	1.3×10^5	2.9×10^5
1.4×10^1			6.5×10^4	6.1×10^4	2.0×10^5
7.1×10^1			1.6×10^4	1.0×10^4	7.2×10^4
1.4×10^2			7.1×10^3	3.7×10^3	4.2×10^4
7.1×10^2			1.2×10^3	2.3×10^2	9.5×10^3
1.4×10^3	7.8×10^{-1}	5.1×10^0	4.3×10^2	1.8×10^2	4.9×10^3
7.1×10^3	1.0×10^0	9.1×10^0	7.7×10^1	1.4×10^2	6.5×10^2
1.4×10^4	1.1×10^0	9.8×10^0	4.7×10^1	7.0×10^1	3.0×10^2
7.1×10^4	1.1×10^0	6.8×10^0	1.2×10^1	1.3×10^1	7.1×10^1
1.4×10^5	1.0×10^0	3.1×10^0	5.4×10^0	6.5×10^0	4.0×10^1
7.1×10^5				1.1×10^0	1.7×10^1
1.4×10^6				3.7×10^{-1}	1.3×10^1
7.1×10^6				9.7×10^{-4}	5.0×10^0
1.4×10^7				8.0×10^{-7}	2.5×10^0
7.1×10^7				8.7×10^{-9}	3.3×10^{-1}

TABLE VII-15

FISSION PRODUCT INVENTORY (CURIES) PLANNED DESTRUCTIVE TEST

Time After Release (sec)	Iodine 131	Iodine 133	Noble Gases	Halogens	Solids
1.4×10^0			2.1×10^6	4.2×10^6	8.2×10^6
7.1×10^0			1.0×10^6	1.3×10^6	2.9×10^6
1.4×10^1			6.5×10^5	6.1×10^5	2.0×10^6
7.1×10^1			1.6×10^5	1.0×10^5	7.2×10^5
1.4×10^2			7.1×10^4	3.7×10^4	4.2×10^5
7.1×10^2			1.2×10^4	2.3×10^3	9.5×10^4
1.4×10^3	1.1×10^0	5.0×10^1	4.3×10^3	1.8×10^3	4.9×10^4
7.1×10^3	4.0×10^0	9.1×10^1	7.6×10^2	1.4×10^3	6.3×10^3
1.4×10^4	4.5×10^0	9.8×10^1	4.5×10^2	6.8×10^2	2.8×10^3
7.1×10^4	4.5×10^0	6.0×10^1	1.1×10^2	1.2×10^2	5.4×10^2
1.4×10^5	4.3×10^0	3.1×10^1	4.2×10^1	5.5×10^1	2.3×10^2
7.1×10^5				6.4×10^0	3.8×10^1
1.4×10^6				1.9×10^0	2.0×10^1
7.1×10^6				4.4×10^{-3}	4.1×10^0
1.4×10^7				3.6×10^{-6}	1.7×10^0
7.1×10^7				3.1×10^{-9}	1.4×10^{-1}

TABLE VII-16

FISSION PRODUCT INVENTORY (CURIES)
UNPLANNED DESTRUCTIVE EXCURSION AT IET

Time After Release (sec)	Iodine 131	Iodine 133	Noble Gases	Halogens	Solids
5.0×10^0			1.3×10^6	2.0×10^6	3.7×10^6
2.5×10^1			4.8×10^5	3.6×10^5	1.5×10^6
5.0×10^1			2.6×10^5	1.7×10^5	9.9×10^5
2.5×10^2			4.2×10^4	1.5×10^4	2.8×10^5
5.0×10^2			2.0×10^4	4.2×10^3	1.5×10^5
2.5×10^3	4.3×10^0	9.0×10^1	2.5×10^3	2.0×10^3	2.9×10^4
5.0×10^3	5.6×10^0	1.0×10^2	1.2×10^3	1.8×10^3	1.1×10^3
2.5×10^4	5.6×10^0	8.0×10^1	3.2×10^2	3.5×10^2	1.8×10^3
5.0×10^4	5.0×10^0	1.3×10^1	1.7×10^2	1.9×10^2	9.1×10^2
2.5×10^5	4.0×10^0	1.2×10^0	2.0×10^1	3.1×10^1	1.4×10^2
5.0×10^5				1.2×10^1	7.4×10^1
2.5×10^6				6.0×10^{-1}	2.3×10^1
5.0×10^6				4.6×10^{-2}	1.2×10^1
2.5×10^7				1.2×10^{-8}	1.8×10^0
5.0×10^7				1.1×10^{-8}	7.3×10^{-1}

4. BERYLLIUM-TOXICITY HAZARDS

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.

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: (a) acute berylliosis, (b) chronic berylliosis, and (c) dermatitis.

4.1 During Nondestructive Tests

During elevated temperature 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 ensure that the beryllium concentration in the stack effluent does not exceed the tolerance limit of $0.01 \mu\text{g}/\text{m}^3$ 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.

(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, nor 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 ensure the following AEC conditions are met:

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

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

(c) Beryllium concentration should not exceed $25 \mu\text{g}/\text{m}^3$ 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 safely 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.

4.2 During Destructive Test

From the data of Lee and Brick [29], which give the compressive or rupture strength of beryllium crystals as $\approx 200,000 \text{ lb/in.}^2$, it is estimated that relatively little shattering of the beryllium will occur during the destructive test since the pressures generated in the core will be considerably below the rupture strength. However, it was assumed that one percent (40.40 g) of the beryllium would fragment to grain size and, therefore, be available for release to the atmosphere.

During the planned destructive test, the on-site beryllium concentration will not present a hazard to personnel due to the strict meteorological control that will be exercised. Assuming a one percent release during lapse conditions with a wind velocity of 7.5 mph, the beryllium concentration is calculated to be $0.45 \mu\text{g}/\text{m}^3$ at a distance 7000 feet from the release point. The concentration decreases rapidly with distance and is essentially zero at the nearest site boundary.

The beryllium concentration values presented above were calculated by using the following equation [25]:

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

where

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

h = height of release, 0 meters

V_d = velocity of deposition, 10 cm/sec

Q = beryllium initially released, 40.40 grams

$$c^3 = c_x c_y c_z$$

$$c^2 = c_y c_z$$

All other values are defined under Radiological Analytical Techniques.

5. CONCLUSIONS

The radiological analysis for the SNAPTRAN 2/10A-1 and -2 safety testing programs is, in general, based upon conservative assumptions resulting in upper limit values for the potential radiation hazards to NRTS personnel and to the public.

During all nuclear operations 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 approximately 1-1/2 miles. During nuclear operation all personnel in this area will be located in the underground shielded control and equipment building. Following all tests, reentry to the test area will be subject to Health Physics control.

For those tests in which there is no fission product release to the atmosphere, the maximum total integrated dose to on-site personnel is calculated to be less than 5.0×10^{-3} rem. This dose is received in the IET shielded control room. The post-test dose rate in the control room is negligible. The calculations are based on a 20 MW-sec energy release and on the conservative assumption that the radiation is attenuated only by materials external to the reactor cylinder.

All tests in which fission product release can reasonably be expected will be conducted under strict meteorological control as defined in Section VII-3.3. The maximum scheduled event involving the release of fission products to the atmosphere will be the planned destructive test (200 MW-sec). For this case the maximum total integrated dose to on-site personnel is calculated to be 4.8×10^{-2} rem which is received in the control room at the point nearest the reactor. This is a conservative value because the effect of self-attenuation by the core materials has been neglected in the calculations. The dose-rate in the control room following the destructive test is negligible.

The downwind doses at the nearest site boundary ($\approx 10^4$ meters) are as follows:

	Whole Body Dose (rem)	Thyroid Dose	
		Inhalation (rem)	Ingestion (rem)
Dose to a child	8.4×10^{-4}	2.0×10^{-3}	4.4×10^{-2}
Dose to an adult	8.4×10^{-4}	1.3×10^{-3}	4.4×10^{-3}

The whole body dose includes the gamma dose from the passing cloud and the deposition gamma dose for a one-year exposure. These values are well below the recommended limits presented in the AEC "Standards for Radiation Protection", Chapter 0524.

Due to the protective features of the reactor systems and administrative controls as described in Section VII-2, the occurrence of an accidental destructive excursion at any time other than during transient testing is not considered credible. Since all transient tests in which fission products could be released to the atmosphere will be conducted under strict meteorological

control, an unplanned destructive excursion could occur only under lapse meteorological conditions. Assuming an energy release of 220 MW-sec during weak lapse conditions, the calculated maximum doses at the nearest site boundary are as follows:

	Whole Body Dose (rem)	Thyroid Inhalation Dose (rem)
Dose to a child	5.2×10^{-3}	6.3×10^{-2}
Dose to an adult	5.2×10^{-3}	4.2×10^{-2}

The ingestion dose to the thyroid is not included in the results since this dose may be eliminated by breaking the milk-cycle chain.

The above doses are well below the reference safety values in Chapter 0524 of the Standards for Radiation Protection.

On the basis of the preceeding analysis, it is concluded that the SNAPTRAN 2/10A-1 safety test program and the planned destructive test, SNAPTRAN 2/10A-2, can be conducted without undue risk to on-site personnel or to the general public.

VIII. EMERGENCY ACTION PLANS

1. PHILLIPS PETROLEUM COMPANY BASIC PLAN

1.1 Emergency Action Planning Committee

This committee is responsible for formulating and at least reviewing 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.

1.2 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.

1.3 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 coordinate PPCo activities under the direction of the ID Control Group. The senior Operations supervisor in the GPP and TRA areas, 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, reentry, etc, and for immediate notification to the PPCo Control Group and the ID Control Group (through the CFA Security Dispatcher, Ext. 2345).

1.4 All PPCo Supervisors

All supervisors are responsible for: (a) being familiar with the provisions outlined in these plans, (b) instructing employees in these provisions, (c) accounting for employees upon evacuation, (d) rendering all assistance possible as directed by the PPCo Control Group, (e) maintaining controls to prevent

sabotage and/or damage of vital equipment, and (f) providing emergency assistance to non-Phillips Petroleum Company areas under conditions listed below:

- (1) Immediate emergency assistance to save lives or attempt to save lives of personnel may be made without prior approval providing the Project Manager and the respective Assistant Manager are notified as promptly as possible.
- (2) Other assistance will require prior approval of the Project Manager or respective Assistant Manager (excluding Transportation Section which will dispatch buses during regular working hours for plant evacuation, then notify one of the above individuals).

1.5 Branch and Special Group Assignments

1.51 Wardens. Wardens are designated by supervision to participate in 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 areas. If possible, without jeopardizing personal safety, wardens 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.

1.52 Emergency and Special Reentry 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.

1.53 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.

1.54 Security Branch. The 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 reentry, 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.

1.55 CFA Maintenance. The Transportation Section is responsible for providing emergency transportation for specific NRTS transportation as directed. This section will maintain a current record of buses needed for plant areas during regular working 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.

1.56 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 (J.P. Lyon) will be called. (If deemed necessary, the PPCo Control Group will be contacted.)

1.57 Relocation Centers. Relocation centers for Control Group activities in event of a site-wide or national emergency or local TAN emergencies will be 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 602, or
- (2) turnaround room at IET.

1.58 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 of 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:

- (a) strategic warning may not be received and tactical range warning lead-time may be short or nonexistent,
- (b) enemy attack may involve biological or chemical agents, nuclear weapons in the multimeton range, or any combination of the three,
- (c) sabotage may occur, and
- (d) 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.

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.

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.

1.59 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, or 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.

1.510 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 must 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. STEP EMERGENCY ACTION PLAN -- IET FACILITY

2.1 Declaration of Emergency

Authority to declare an emergency is delegated to:

Assistant Manager, Nuclear Safety Technology

Engineering and Test Branch (STEP) Manager

Engineering Section Chief

Nuclear Test Section Chief

Responsible supervisor in control room during reactor operation

Responsible Health Physicist assigned to the reactor area.

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.

2.2 Responsibilities

The senior supervisor present at the IET control room will be responsible for:

(1) reporting to the person who observed the event or to the person who declared the emergency and then taking charge of all activities,

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

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

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

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

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

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

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

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

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

(8) notifying the first member of the PPCo 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 PPCo Control Group, the 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,

(9) assembly of STEP Senior Staff and secretary,

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

(11) recognition of senior person in charge of the reactor area,

(12) determining the location and identity of all persons in the IET area,

(13) 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,

(14) ensuring surveillance of meteorological and radiation conditions in the IET control room area, and

(15) documentation of events and ensuing activities through photographic, sound recording, or other means.

The responsible supervisor in the control room has the responsibility for:

(1) safe shutdown of the reactor as rapidly as possible,

(2) 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,

(3) providing at least one person in the control room to maintain communication with the reactor area and with the senior supervisor,

(4) use of all monitoring instrumentation and recorders available in the control room.

The senior person in the IET reactor area will be responsible for:

(1) directing the evacuation of the reactor area if the incident results in an evacuation signal or order,

(2) 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.

(3) 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,

(4) monitoring radiation levels at the reactor building and ordering evacuation of the area if the levels warrant it,

(5) removal of portable radiation monitoring equipment from the reactor area when evacuation takes place,

(6) placing in service any radiation detectors and recorders in the reactor area with sufficient supply of paper and ink,

(7) providing sufficient transportation to evacuate reactor area, and

(8) specifying evacuation route and terminal point when the evacuation order is initiated at the reactor area.

The Security Guard on duty at Post 710 or IET Guard Post has the responsibility for:

(1) allowing no one to enter or leave the STEP area unless on official business related to the emergency,

(2) maintaining a record of all entries and departures to or from reactor areas and the control center following notification of an emergency, and

(3) 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

3.1 Operational Information

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.

All evacuation switches are of the nonreturn 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 reentry following evacuation with the evacuation horn possibly still sounding.

It should be emphasized that an "alert" is not necessarily followed by an "evacuation"; neither is an "evacuation" necessarily preceded by an "alert".

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.

3.2 Alert

The alert signal is an approximate three-minute continuous sounding of the horn with nonvarying intensity.

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.

All operating equipment is to be shut down and personnel will safeguard classified documents in their possession in preparation for evacuation.

3.3 Evacuation of IET Area

The evacuation signal is an oscillating horn at five seconds on and five seconds off repeated for five minutes.

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.

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.

3.4 All Clear

The all clear signal is three (3), twenty (20) second continuous horn signals separated by twenty (20) second intervals of silence. The "all clear" will be sounded by manual operation of the siren control on the "alert" position.

Upon receipt of an "all clear" following an alert, employees are to resume normal assignments.

Upon receipt of an "all clear" following evacuation, reentry to the reactor areas will be made in the usual manner.

3.5 System Tests

The STEP-IET evacuation alarm system should be tested at intervals not to exceed two months, in cooperation 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 Emergency Action Planning Group, for further action, such as repair or modification.

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.

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.

The "all clear" will follow every test and will be acknowledged by the reactor area via its control room.

3.6 Evacuation Transportation

A minimum of two 6-passenger vehicles will be stationed in the turnaround room at IET during nuclear operations for emergency evacuation. The conditions stated under Fighting Fires, Section VIII-5.3, shall apply.

The locomotive will be available for emergency evacuation from IET through the underground hatchway during a nuclear test.

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.

3.7 Emergency Transportation Drivers

A minimum of four (4) personnel working in the IET building during tests will be assigned to drive the vehicles from the turnaround room.

3.8 Wardens

Wardens are responsible for thoroughly knowing 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 evacuation alarms sound, 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.

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. STEP-IET EMERGENCY REENTRY PLAN

4.1 Authority and Responsibility

The authority and responsibility for assignment of personnel to the emergency reentry teams shall reside with the STEP Project Manager. Makeup of the reentry team shall include the following as a minimum:

- (1) a senior STEP supervisor who acts as planning director,
- (2) a plant operator of the applicable facility,
- (3) one STEP Health Physicist,
- (4) one surveillant physicist, and
- (5) one reactor technician.

The existing Fire Brigade will supplement the reentry team to handle fire fighting and emergency first aid as necessary.

The planning director of the emergency reentry team will be responsible for providing the team with all available information necessary to make the reentry. This includes plant photographs, drawings, and any other pertinent information.

4.2 Training

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 reentry 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.

4.3 Emergency Equipment Available

4.31 Protective Clothing

- (1) 100 coveralls
- (2) 100 cloth shoe covers
- (3) 50 cotton gloves
- (4) 25 latex gloves
- (5) 24 rubber gloves
- (6) 20 cotton caps

- (7) 10 lab coats
- (8) 20 rubber boots
- (9) 5 plastic suits and head covers
- (10) 12 Comfo respirators
- (11) 3 thirty-minute press demand Scott Air-Packs
- (12) 12 hard hats

4.32 First Aid

- (1) 3 first aid kits
- (2) 3 burn kits
- (3) 1 decontamination kit
- (4) 1 MSA cabinet
- (5) 3 stretchers with blankets

4.33 Fire Protection (NaK)

- (1) 2 Met-L-X #30 Ansul extinguishers
- (2) 1 Met-L-X #300 unit mounted on test dolly
- (3) 2 shovels
- (4) 2 full length leather coats
- (5) 2 leather leggings
- (6) 2 elbow length leather gloves
- (7) 2 full face shields
- (8) 2 -- #50 cans Met-L-X powder

4.34 Radiation Instruments

- (1) Ample supply portable radiation instruments
- (2) 3 nuclear accident dosimeters

4.35 Facility Drawings. The STEP Design Engineering Section Chief is responsible for assuring maintenance of current drawings of each reactor area for emergency reentry use.

5. FIRE SAFETY AND PREVENTION

5.1 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.

5.2 Reporting Fires

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.

Instructions on filling out formal reports of a fire may be found in the Phillips AED Administrative Instruction Bulletin, AIB-6.06.

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.

5.3 Fighting Fires

Every employee is expected to fight a fire which develops in his work area to prevent its spreading and to control it if possible.

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.

The Brigade members will have first priority of available government transportation in the STEP areas in event of a fire.

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.

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.

5.4 Fire Prevention

(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 area.

(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 combustible 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.

5.5 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.

5.6 Fire Brigade

5.61 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.

5.62 Brigade Responsibilities

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.

In the event of a fire during normal working 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.

Fire Brigade members will not attempt to fight fires in any electrical substation or transformer area until the AEC Fire Department arrives.

In the event of a fire, Brigade members will have first priority of available government transportation in the STEP areas

Brigade members will be responsible for providing emergency first aid assistance as required either as a consequence of fire or in conjunction with emergency reentry activities.

5.63 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.

5.7 Fire Alarms

5.71 Response to Fire Alarms

(1) The area supervisor will immediately report to the scene of the fire.

(2) Fire Brigade members will immediately report to the scene of fire.

(3) Persons not engaged in fighting the fire shall remain away from that area.

5.72 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.

5.73 Maintenance and Testing

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.

Whenever feasible, all personnel will be notified of a scheduled fire alarm test prior to the test and at the time of the test.

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