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AEC RESEARCH AND DEVELOPMENT REPORT

ORNL-4058 (Rev.)
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TSF-SNAP REACTOR SAFETY ANALYSIS REPORT

Joseph Lewin

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Neutron Physics Division

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Joseph Lewin

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AUGUST 1967

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Oak Ridge, Tennessee

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FOREWORD

A complete description of the TSF-SNAP reactor installation and analysis of its safety is contained in two documents: this ORNL report and a report by Atomics International, NAA-SR-MEMO-11968, "Design Description and Safety Aspects of the SNAP-TSF Reactor." The two documents are therefore submitted together for review of this reactor installation. Although some overlapping of information presented does exist, the two reports supplement each other and together comprise a complete TSF-SNAP Installation Safety Analysis Report.

Contributions to the ORNL report have been made by a number of people. S. J. Ditto of Instrumentation and Controls Division wrote the I and C description. C. E. Clifford and L. B. Holland made many valuable suggestions, as did D. R. Ward, T. J. Burnett, and others.

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ABSTRACT

The Tower Shielding Facility SNAP Reactor (TSF-SNAP) is a small, NaK-cooled reactor of the SNAP-10A and SNAP-2 type that has been modified for use in shielding experiments. It is capable of operation at 10 kw at 716°F NaK outlet temperature, but is not temperature limited up to 800°F. The reactor will be attached to a positioning device constructed specifically for its use and independent of the towers and hoisting equipment used to position the TSR-II.

The design of the reactor and its controls and the siting of the facility are such that normal operation for experiments, as proposed, will not involve radiation levels that are hazardous to the general public. The unique shutdown mechanism inherent in the fuel, hydrogen evolution in a nuclear excursion situation, removes reactivity and disassembles the core before any possibility of fuel melting can arise. The consequences of the maximum credible accident will not involve any unacceptable hazard to the public or to the rest of the Oak Ridge National Laboratory.

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INTRODUCTION

A new reactor will be installed at the Tower Shielding Facility of the Oak Ridge National Laboratory in order to conduct analysis validation and shield optimization experiments in support of the SNAP technology program. This reactor will be a modification of the SNAP-10 and SNAP-2 designs that have been developed and thoroughly tested by Atomics International, A Division of North American Aviation Corporation. Atomics International has designed the necessary modifications, is fabricating the reactor, and is scheduled to deliver the major components to ORNL by February 1, 1967.

The operation of the reactor will be as a research tool, a source of neutrons for experiments designed to provide a test of the accuracy of advanced analytical techniques for calculating accurately the intensity of radiation transmitted through shields for space systems. The advanced analytical techniques include improved Monte Carlo and multigroup (S_n) transport codes, and the coupling of such codes for specific problems. To a large extent, therefore, the operation of the reactor will be tailored to the requirements of the space shielding program and of the analytical program and "routine operation" is a term that, except for safety practices, cannot be defined in advance as with power or test reactors.

The TSF-SNAP reactor assembly will include parts that were originally fabricated for other purposes in the SNAP program at Atomics International. Thus the fuel, the internal and external reflectors, and the control drums were available prior to the start of this project. Design studies were made of their use in this application, and they have been inspected and reworked where necessary.

The pressure vessel, the top head of the vessel, some of the NaK piping, and the fine control drum actuators have been fabricated to existing or modified designs specifically for the TSF-SNAP reactor. New designs were created for the coarse control drum actuators and the NaK-to-air heat exchanger assembly, as well as the reactor-to-shield attachment. The system must be designed to fit within the envelope determined by the shield shadow cone developed by the SNAP-2 lithium-hydride shield. In addition to

the main components, provisions for the NaK expansion, for differential thermal expansion of the structure, and for handling have been designed.

The system will be limited in power by temperatures achieved at graphite bearings that are located in the control drum actuators and the control drum pivot points as well as at parts of the pneumatic actuators for the coarse control drums. This limiting temperature, based on reliability expectations for the parts and not on any catastrophic threshold, is 500°F for in-air operation. It is estimated on the basis of warm-day conditions that the temperature limit will be reached at a power level of 11.5 kw. The system is therefore said to be capable of extended operation at 10 kw. The operating life is nominally rated at 2,000 hrs. This is based on the probable limit of control mechanism wear life at elevated temperature. Thus the probable operational life of the reactor will be greater since much of the operation will be at reduced temperature. Although ref. 1, the attached report by Atomics International, speaks of a design "for continuous operation at a thermal power output of 5 kw, and occasional operation up to a 10-kw level" tests of temperature distributions in the reflector assembly that postdate that report indicate clearly that no temperature limitations will exist up to reactor power of 10 kw and higher. There are no other known limitations on reactor power, either of an operational or safety nature that would precede the temperature limitation.

The nuclear design of the reactor is identical to the SNAP-2 and SNAP-10 systems, and the critical experiments, analyses, and tests conducted in the past by Atomics International are directly applicable to the TSF-SNAP system. Figure 1 is a general view of a SNAP-10A reactor, while Figs. 2 and 3 show top views of an assembled vessel and reflector and a core and reflector individually. Figure 4 is an elevation of the TSF-SNAP reactor, and Fig. 5 is a plan view of the core with some dimensions given.

The nuclear characteristics of these reactors have been extensively explored by Atomics International and a comprehensive program of reactor physics experiments will not be necessary at ORNL. Both endurance runs and two destructive transient experiments have been conducted and the data from these are largely applicable to analysis of normal operation and operational hazards of the TSF-SNAP installation.

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Fig. 1. SNAP-10A Reactor as Built for Space Flight by Atomics International, a Division of North American Aviation, Inc.

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SIDE VIEW

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TOP VIEW

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Fig. 2. SNAP-10A Reactor - Top View and Side View (Pump and NaK Lines Not in Place).

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CORE

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Be REFLECTOR

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Fig. 3. SNAP-10A Reactor - Core and Be Reflector.

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Fig. 4. TSF-SNAP Reactor for Shielding Research.

Fig. 5. TSF-SNAP Reactor - Horizontal Cross Section.

The Tower Shielding Facility will be modified to accept the TSF-SNAP reactor and to carry out the experiments planned for this program. The TSF-II, a water-moderated and water-cooled reactor, will be moved to the north side of the pool and placed in the 12-ft-diam spherical shield. The pool that is presently filled with water and used as a maintenance place for the TSR-II will be drained and rendered incapable of storing water by removal of the drain valve and other measures. The pumps that at present feed water to the pool will be disabled and water piping in the vicinity of the pool will be disconnected from its source and capped. The pool will be used as a detector chamber and will be occupied by a 10-ft-diam detector collimator. This is shown in Figs. 6 and 7.

A three-wall structure will be built of concrete blocks just south of the detector chamber. This will serve both as a shelter for the reactor and as a shield for personnel against gamma rays from the reactor when it is not operating. Manipulators and a shielding window will be installed in the east wall of this structure for observation of and limited remote maintenance on the reactor. Figure 8 shows the reactor in the shelter.

The reactor itself will be suspended at the end of a boom that extends horizontally from a vertical column. This device, called the reactor rotator, will make it possible to locate the reactor over the detector chamber, inside the shelter structure, or at any position in between. It will also make it possible to move the reactor up and down through a stroke of $3\frac{1}{2}$ ft to rotate it about its own axis $\pm 180^\circ$.

All administrative controls for personnel safety, as they existed during operation of the TSR-II, will be maintained for operation of the TSF-SNAP reactor.

I. REACTOR DESIGN SUMMARY

1. Description of Reactor

The TSF-SNAP reactor is basically a modified SNAP-10A/2 reactor. The reactor is a highly enriched uranium-fueled, hydrogen-moderated, beryllium-reflected core. The reactor core is composed of 37 cylindrical fuel-moderator elements arranged in 1.260-in. centers in triangular array,

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Fig. 6. Facilities Arrangement, Showing Core Mapping Collimator in Maintenance Area.

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Fig. 8a. Elevation of Reactor in Shelter.

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Fig. 8b. Plan of Reactor in Shelter.

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resulting in a hexagonal right cylinder approximately 7.7 in. across the flats, 8.8 in. across the corners, and 12.3 in. high.

The fuel can outside diameter is 1.25 in., and the fuel rod diameter is 1.210 in. The overall fuel element length is 12.450 in., while the fuel alloy length is 12.25 in.

The fuel element array is encased in a type 321 stainless steel cylindrical core vessel approximately 16 in. high, 8.875 in. ID, and 0.050 in. thick. The fuel elements are supported and held laterally by grid plates constructed of Hastelloy-C. Fuel element end pins engage holes in the grid plates to position the elements in the vessel. The lower grid plate is a built-up structure and is orificed for proper NaK flow distribution in the SNAP-10A reactor. The top grid plate, 1/8 in. thick, is held in place by the action of six springs compressed between the upper surface of the top grid plate and the vessel top head. The springs are fabricated from René 41 wire and exert a force of 37.5 lbs. each. These springs hold the core in place, but allow relative thermal expansion between the core and reactor vessel.

The space between each grid plate and the top and bottom of the core vessel forms the upper and lower plenum chambers. A baffle plate, part of the lower grid plates structure, divides the lower plenum between the coolant inlet lines and the lower grid plates and acts as a vortex breaker. The flow distribution in the TSF-SNAP reactor will be very nearly flat across the core diameter. Six beryllium reflectors fill the space between the fuel and vessel wall in such a way that the 18 peripheral coolant passages thus formed provide a total flow area of 1.120 in.² Total flow area of the 54 inner coolant passages is 3.989 in.² Total flow area of the 12 unorificed peripheral channels is approximately 0.897 in.², to give a total core flow area of approximately 6.00 in.²

The following listing of design data is a recapitulation of general reactor information:

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Fuel: 93.2% ^{235}U , 6.8% ^{238}U (9.82 wt% U in U-Zr-H alloy)

Total ^{235}U in core: < 4.75 kg

Fuel alloy: U-Zr-H₂ (88.38 wt% Zr); H/U = 45.1.

Fuel used: Existing SNAPTRAN V fuel (see Section 2, Fuel)

Moderator: H in hydrided zirconium

$$N_{\text{H}} = 6.5 \times 10^{22} \text{ atoms/cc (1.8 wt\% H); H/Zr} = 1.85$$

Cladding material: Hastelloy N

Cladding form: Tubes 1.25 in. OD, 0.015 in. thick, 12.82 in. long

Number of active elements: ~ 36

Number of dummy element: ~ 1 (Be in Hastelloy can) (to be established by experiment)

Average beryllium thickness in reflector: 2 in.

Total excess K loaded: ~ \$1.80 with shield in position at 75°F, and NaK in vessel

Worth of each control drum: ~ \$2.00 with one Be shim 1/8-in. thick
~ \$2.15 with three Be shims 3/8 total thickness

Number of drums: 4

Core vessel: Type 321 stainless steel, 8.875 ID x 0.050 w.t. x 16 in. high

Heat exchanger: Type 321, 347, and 304 stainless steel

Nominal heat transfer areas to air:

Heat exchanger: 5018 in.²

Vessel: 542 in.²

Be reflector sides: 530 in.²

Top of Be reflector: 77 in.²

Heat transfer area from fuel rods to NaK: 1655 in.²

Coolant: NaK, 78% K - 22% Na, 2.46 gallons total

Maximum coolant flow rate: 4 gpm

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Nominal maximum coolant temperature: 800°F

ΔT in NaK across core at 10 kw: 96°F

The NaK system is closed and sealed under an atmosphere of argon (see Fig. 9). An expansion chamber at the top of the heat exchanger has a normal fill level at room temperature of 1 in. above the bottom of the chamber. The argon atmosphere over the NaK will be charged at a pressure of $15\frac{1}{2} \pm \frac{1}{2}$ psia. Thus at room temperature the system is under atmospheric pressure. At 750°F NaK temperature, the pressure rises to 60 psia which is the nominal design pressure for the system. The vessel has been tested to 125 psi at room temperature as part of the inspection and acceptance program for all reactor parts.

2. Fuel

The fuel to be used in the TSF-SNAP reactor consists of a set of elements that was fabricated originally for a transient experiment that had been designated SNAPTRAN V. This experiment was cancelled, and the fuel became available for other uses. The principal differences between this fuel and that used in the FS-1, FS-3 and FS-4 systems, that have been operated in space and on the ground, are: (1) the higher hydrogen content of this fuel, 6.50×10^{22} atoms/cc vs 6.28 and 6.35×10^{22} atoms/cc; and (2) the lack of qualification of this fuel for temperatures over 800°F.

The lack of qualification refers only to tests of the ceramic coating that is applied to the interior wall surfaces of the Hastelloy N cans. This coating is intended primarily as a barrier to hydrogen leakage through the walls of the cans at temperatures above 1000°F. In addition, however, the samarium prepoison that is included in SNAP reactors to reduce the cold clean reactivity of the system, is included in this coating in the form of Sm_2O_3 .

In the SNAP systems intended for long runs at rated power the fuel elements are subjected to qualification tests that establish the integrity of the ceramic hydrogen barrier.³⁴ This was not done in the SNAPTRAN V fuel because the hydrogen barrier has little significance in a destructive transient test. In the operation of the TSF-SNAP reactor this will also have no significance since the temperatures of operation will not exceed

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Fig. 9. Vertical Cross Section of NaK-Containing System.

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800°F and hydrogen redistribution and leakage should be insignificant.

The higher hydrogen content of the fuel in the TSF-SNAP reactor should have no effect on the operation of the reactor other than to provide a higher positive reactivity for a given loading and to provide a more rapid shutdown mechanism by hydrogen evolution, in case of a power transient. The SNAPTRAN-1, -2, and -3 tests all used fuel with the higher hydrogen content.

3. Nuclear Characteristics

The excess multiplication normally loaded into a SNAP-10A core is considerably higher than will be necessary for the TSF-SNAP system. The reduction in loaded excess is achieved in the TSF-SNAP core by replacing one or more fuel rods with beryllium rods. It is expected that when the central fuel rod is thus replaced, the total loaded excess in the core will be \$1.80 at room temperature with the SNAP-2 shield in place. This will be allocated as follows:

Fuel burnup and fission product buildup	\$-.01
Samarium burnout	+.01
Xenon buildup	-.04
Power defect	-.02
Temperature defect (to 10 kw)	-1.34
Operational requirement (period of 20 sec)	-.28
Contingency	<u>-.12</u>
Total Reactivity Requirement	\$1.80

The temperature defect plus contingency indicated above would allow an overall temperature rise of 670°F. This is based on an overall temperature coefficient of .22¢/°F as reported in the SLOFS-1 Summary Safeguards Report.² This coefficient is a rather slow feedback effect since it requires the entire system, including reflector, to come to one equilibrium temperature. Another temperature coefficient that is faster acting since it was observed in the destructive transient tests (SNAPTRAN-3) is .19¢/°F.⁵ On the basis of this core coefficient, the temperature defect plus contingency allowed in the loading would allow a core temperature rise of just over 770°F to be achieved, while still retaining the

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control allowance of 28¢. Temperature coefficients, all negative in this reactor, are discussed further in ref. 1.

In order to change the power level of the reactor in a reasonable time interval, an excess multiplication that will produce reactor periods in the range of 20 sec is desirable. A solution of the in-hour equation for the SNAP-10A reactor gives an excess of 27.88¢ required. The operational requirement was therefore set at 28¢.⁶

The time and power dependent reactivity effects were based on 10,400 kw hr of operation.⁶ The xenon buildup value given is one quarter of equilibrium because operation of the reactor will be highly intermittent. The samarium credit taken refers to the samarium that is incorporated, as a burnable poison, in the ceramic hydrogen barrier that is coated on the interior surfaces of the fuel cans.

The power coefficient in this reactor is $-.2¢/\text{kw}$, thus giving 1¢ as the power defect at 5 kw.

The reactivity of a SNAP-10 system is highly dependent on two factors: the hydrogen presence in the fuel for moderation of fission neutrons, and reflection of leakage neutrons back into the core. The hydrogen comes out of the metal hydride as free hydrogen, with increasing rapidity at temperatures above 1,000°F. This provides a shutdown and core disassembly mechanism in case of an excessive power excursion. The possibility of adding reactivity to the system simply by addition of material to the vicinity of the reactor, thus increasing reflection of neutrons, is a hazard that makes it necessary to exercise great care in handling. All nuclear control over the reactor is exercised by varying the reflection of neutrons.

With no beryllium reflector assembly around the core, the system is far subcritical. Even with the reflector on the lower hinge pins and the two halves of the reflector rotated about the pins until the reflector is only 10° away from the vertical surface of the core with all drums in, the system is subcritical about \$7.20,⁷ i.e., this amount of rotation is worth about \$9.00. Bringing the two halves of the reflector up into position with all four control drums fully out, brings the system up to about

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\$6.20 below critical. With each control drum worth about \$2.00 with one 1/8-in.-thick shim, the system goes critical when the last of the four drums is rotated in 44 1/2% of its full travel. If fully shimmed, each drum could reach a worth of about \$2.15.

The maximum positive reactivity effects of various reflectors that have been investigated at Atomics International^{3,7} are given below:

Simulated human body leaning directly on top of core with arms extended around reactor = +\$.84.

Simulated human hands jammed into voids between two withdrawn drums and core = +\$1.11.

Maximum credible worth of human body and hands = +\$2.15.

80 lb. paraffin tangential to reflector = +\$.14.
(18 in. high x 5 in. thick x 28 in. wide)

Maximum worth of 3/8-in. Lucite shims on all shimmable reflector surfaces = +\$.74.

Infinite water reflection (Be reflector in place, all drums out)
No water in core = +\$9.47
Water in core = +\$15.47.

SNAP-2 shield in position below reactor = +\$.35.

NaK in core = +\$.26.

The total worth of all Be shims is \$1.26, or approximately \$.42 per each 1/8-in. layer of shims on all shimmable surfaces.

Other nuclear characteristics of a SNAP-10 core, as taken from ref. 7, follow:

Prompt neutron lifetime (μsec)	6.5
Median fission energy (eV)	0.12
Effective delayed neutron fraction (β_{eff})	0.008

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Thermal flux (< 0.4 eV) at 10 kw:

core average (n/cm².sec) 6.3×10^{10}

peak in core (n/cm².sec) 1.2×10^{11}

Fast neutron leakage (> 0.1 MeV)(n/sec) 1×10^{15}

4. Thermal Design Features

The heat transfer fluid is eutectic sodium-potassium alloy (NaK-78) which is 78 wt% potassium and 22 wt% sodium. This alloy has a melting point of approximately 10°F and a boiling point of about 1440°F at 1 atmosphere pressure and 1505°F at 20 atmospheres. The NaK will be purified to eliminate oxide content to a purity at which a temperature as low as 20°F will not induce precipitation of oxides to inhibit flow. This will be established through the plug loop in the NaK fill cart system (see Chapter VI, Section 3).

The TSF-SNAP system will contain a total of 2.46 gallons at room temperature. At 800°F this volume becomes 2.85 gallons. An electromagnetic pump will help to lift hot NaK up from the core through a vertical riser to the upper end of the NaK-to-air heat exchanger. The NaK will then flow downward through twelve finned tubes of the heat exchanger, to a ring manifold, from which it will return to the bottom plenum of the core vessel. The maximum flow rate that the pump can induce against system resistance is 4 gpm. It should be noted, however, that operation with no pump action and natural convection flow only is entirely feasible at 10 kw (see p. 29 of ref. 1).

Using the Rice-Heilman equation for heat transfer to still air,⁸ Fig. 10 was obtained to indicate the apparent power capabilities of the heat exchanger system for various metal surface temperatures at still air temperatures of 30°F and 65°F. This equation gives

$$q/A = C \Delta t^{1.266} / D^{0.2} T_{avg}^{0.181},$$

where

q/A = rate of heat transfer,

D = characteristic dimension of surface, in.,

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Fig. 10. Heat Transfer to Still Air at Various Heat Exchanger Surface Temperatures.

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T_{avg} = average of surface and ambient temperatures, °F,

Δt = temperature excess of surface over ambient, °F,

C = shape factor

= 1.00.

Since the temperature differential from the NaK to the metal is very low (on the order of 10°F) the metal temperature is essentially the system temperature.

In a SNAP-10 reactor the ratio of peak to average heat flux in a radial direction is 1.4:1; the ratio of average to end flux is 1.2:1; and the ratio of peak to end flux is 1.7:1. The axial flux distribution is such that the ratio of peak to average is 2:1 and peak to end is 5:1.² In the TSF-SNAP system the radial heat flux distribution will be flatter because of the presence of the central beryllium rod.

5. Control Elements and Mechanisms

Control of a SNAP-10 reactor is achieved by varying the effective thickness of the neutron reflector surrounding the core, thus controlling the leakage from the core. The reflector is a hollow octagonal prism of beryllium with a wall thickness of about 2 in. Four semicylindrical beryllium drums spaced at 90° intervals comprise a portion of the beryllium wall. The drums are 2 1/4-in.-deep segments of 7-in.-diam cylinders. Rotation of these drums has the effect of varying the effective thickness of the reflector, thus permitting control of the reactor from subcritical to startup and operation.

Two drums, number 1 and 3, will be used for coarse control and two drums, numbers 2 and 4, for fine control. One of the fine control drums will be used in the automatic power control system. Figures 11 and 12 indicate the general schemes for actuation of the control drums.

The two coarse drums will be in one of two positions at all times. "Drums inserted" is obviously the more reactive position, while "drums withdrawn" is the least reactive position. To insert the coarse control drums, air pressure will be required at the actuator cylinder. The air

Fig. 11. Coarse Control Drum Pneumatically Actuated System.

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Fig. 12. Block Diagram of Electrically Actuated Fine Control Drum Drive System.

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will be supplied to two control valves, one valve for each coarse drum actuator, and will be routed by the valves to the actuators only when electrical power is supplied to the valve solenoids. When the solenoid is not energized, air is bypassed to the atmosphere and a spring drives the piston of the actuator to the "drum withdrawn" position. This is the principal fast-acting shutdown mechanism of the system. It is anticipated that the time required to remove 95% of the reactivity in the coarse drums will not exceed 500 msec from command.

The control valves are commercially available three-way valves and are located at the vertical column of the reactor rotator. Thus they will be accessible for maintenance, even when the shutdown reactor is radio-active but is moved into the maintenance area so that the vertical column is shielded by the north wall of the maintenance area.

The air for the coarse drum actuation system will be drawn from the instrument air supply presently installed in the pump house at the Tower Shielding Facility. The source of this air is a Nash "Hytor" Compressor that puts out oil free but wet air at a maximum pressure of about 80 psig and a rate of 50 SCFM. The air is dried in a Trinity Industrial Dryer to a dew point of -40°F and is filtered both ahead of and following the dryer.

The insertion rate of the coarse drums will be limited by orificing in the air lines. The worth of each drum is expected to be \$2.00, without shims. If shims were added to the flat back face of the drum, the worth can be increased to about \$2.15.

The fine control drums are identical in configuration to the coarse drums. They are actuated, however, by direct current stepping motors which rotate the drums in precise discrete steps. The actuators consist basically of a stator, a rotor, and a brake and are the product of a long development program conducted at Atomics International.¹⁰ Specifically, the electrical stepper motors are the SNAP-8DS actuators with some modifications incorporated for in-air operation. These are described on p. 11 of attached ref. 1. The motors drive the fine control drums through a gear segment and pinion.

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For the TSF-SNAP installation a fine control drum will rotate $1/4^\circ$ per step at a maximum rate of one step per second. At the expected drum worth curve, this will mean a reactivity rate of 1.95% per degree of rotation, at the reactor critical position of the last drum to be inserted. This means $.49\%$ per step, so fine control of the reactor is achievable. The fine control actuators are fed electrical pulses from a controller that sequences the pulses to the proper phase of the stator windings. The fine and coarse control drums will be interlocked so that drums can be inserted only in proper sequence. The control system is more fully described in Section IV of this report.

6. Structural Features

The structures of SNAP-10A flight reactors are designed for two major stress environments: the launch operation and operation in orbit. During launch the reactor is subjected to simultaneous accelerations in both longitudinal and lateral directions. These accelerations are created by both steady and vibratory forces. For design purposes, the following maximum levels were considered in the following combinations:¹³

- (a) 7.5 g aft $\pm 1.0g$ lateral
- (b) 2.5g forward $\pm 1.0g$ lateral
- (c) $\pm 2.0g$ longitudinal $\pm 5.0g$ lateral

All parts of the TSF-SNAP system that were originally designed for operation in space were designed and built to these conditions which are far beyond any normal loads anticipated in operation at the Tower.

The most severe internal pressure conditions in the SNAP-10A system are 33 psig during system ground testing and 0 psia during NaK filling. The pressure conditions in the TSF-SNAP system are expected to be somewhat more severe: at room temperature the reactor will be filled with NaK to a level of 1 in. $\pm 1/2$ in. in the expansion chamber at the top of the heat exchanger. The expansion volume will be filled with argon gas to a pressure of $15-1/2$ psia $\pm 1/2$ psia. As the NaK expands due to temperature rise, the fill gas pressure will rise to 60 psia at a NaK temperature of 750°F . When the hydrostatic pressure of 2.1 lb./in.^2 is added, this makes a total of 62 psia or 47 psig, acting at the bottom of the reactor vessel.

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At this condition the simple hoop stress in the reactor vessel wall is 4000 psi. This does not allow for the redundant assumption of load by the ring structure, at the bottom of the vessel, to which the shield lifting flange attaches. (See drawing of vessel in the Appendix.)

A complete analysis of all structural aspects of the TSF-SNAP reactor assembly has been performed by J. Facha of Atomics International with the following design load criteria:^{14,15}

Maximum shield weight	1,500 lb.
Maximum total weight	2,000 lb.
Shock and acceleration	1/4g max, 10 to 20,000 cps
Attitude	within 5° of vertical

The reactor will be suspended from a horizontal boom, through a bolted flange connection at the top of the support tube. The entire load of the reactor and shield is carried down the support tube into a yoke, then by two support rods into the top head of the reactor vessel. The weight of the shield, the core, the reflector assembly, and the heat exchanger, at temperature, is transmitted upward, through the vessel shell to the weld joint between vessel and head, and finally to the head itself.

The core subsystem, including the fuel rods and internal reflector pieces, weighs 125 lbs. and is supported by a grid plate support ring spot welded to the inside surface of the vessel. The six core hold-down springs, compressed between the top head of the vessel and the top grid plate, exert a 225 lb. downward force on the core assembly. The spring compression load is transmitted from the top grid plate into the six internal beryllium reflectors, down to the lower grid plate and then into the grid plate support ring. The grid plate support ring attachment and the grid plate structure were designed and qualified for the SNAP-10A loading conditions. The fuel is not held rigidly in the vertical direction but is free, within limits, to expand and contract through guide holes in the upper grid.

The bottom 2 1/4-in. portion of the vessel shell, a lower support ring assembly identical to that of the SNAP-10A Flight System, and a bottom ring, which includes the top half of the V-band coupling flange, make up

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a very rigid structural box ring at the base of the vessel. The weight of the shield is carried by the base ring through a V-band coupling into a mating flange bolted to the top of the shield. The 140-lb. weight of the reflector-control assembly is carried on four hinge brackets which are part of the lower support ring assembly. A drawing of the vessel is included in the appendix.

The 120-lb. weight of the heat exchanger, which includes the surge tank, the 12 finned tubes, and the lower ring manifold, is, when hot, supported primarily by the vessel base area through the two NaK coolant return lines and also by the central NaK riser pipe. Part of this weight, the amount depending on system coolant ΔT and the built-in preload, is carried by the expansion coils in the reactor outlet piping between the pump and the surge tank. The basic function of the expansion coils is to compensate the difference in thermal expansion between the hot and cold leg piping. The return lines are supported in the lateral direction near the top end of the reactor vessel by guides which limit horizontal deflection of the return lines relative to the vessel, but do not limit relative vertical movement.

The coolant return lines and the expansion coils were examined as a structural subsystem. This analysis considered the combined effects of weights, piping thermal expansions, flexibility of the expansion coils, piping, and vessel shell at the piping attachment, and thermal expansion movements of the pipe guides at the top end of the reactor vessel. To reduce the possible peak stresses in the vessel shell, at the point where the NaK return line is joined to the vessel wall, it was decided to reduce the lateral restraint on the NaK return pipes, as compared to a SNAP-10A system. The expansion coil with a nominal spring constant of 1,165 lb./in. at room temperature, and 838 lb./in. at 800°F will be welded into the heat exchanger and reactor assembly with no intentional preload. The load introduced into the system due to differences in expansion between the cold and hot legs is less than 30 lb. induced by a compression of the coil of about 0.028 in. Analysis of the effect of complete flow stoppage in a single heat exchanger tube indicated stresses of about 40,000 psi. Plugging and cooling of a single tube would be the most severe case from the

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standpoint of stress in the cold tube. Plugging of a tube in this system is considered extremely unlikely. However, the ASME code, Section III, allows a stress level (computed elastically) of 75,000 psi for the 4,000 cycle limit specified for this system.

The effect of cross winds, causing a temperature gradient from one side of the heat exchanger to the other, is not considered significant because the gradient would be relatively linear, thereby creating very little stress. If one NaK return line, of the two that support the heat exchanger, is colder than the other, the top of the heat exchanger shifts toward the cold side. A 1/4-in. radial clearance, between the support tube and the center tube through the surge tank, allows this to occur so no interference loads are induced. If there is absolutely no cooling on one side of the heat exchanger, twice the nominal system ΔT of $\sim 100^\circ\text{F}$ will occur on the other side, thereby causing the top of the surge tank to shift slightly over 1/8 in. The additional clearance provided allows for assembly misalignment.

The NaK pump throat is subject to a compressive load from the expansion coils, bending moment due to any forced alignment during assembly, and internal or external hydrostatic pressure. All stresses are well within acceptable limits. Forced alignment should be near zero because all assembly is done in a jig that positions and holds the parts during welding.

The reactor vessel top head and the vessel shell were evaluated for the effects of support loading, hydrostatic pressure, and discontinuity at the head to shell junction. Although the analysis of the top head, a circular plate with variable thickness, is not completely rigorous, it is considered to be conservative. A peak stress of 12,230 psi, occurring at the head to vessel weld, is well within acceptable limits.

The detail parts of supporting tie rods and central tube were evaluated, as were the V-band coupling and the shield support ring. All stress levels are well below the maximum allowable limits.

In establishing the vertical loading, a 2g handling factor was applied to all areas, except for support of the heat exchanger through the

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NaK return lines and expansion coils. The handling factor was not applied here because handling is limited to the nonoperating mode. When the system is cold, the weight of the heat exchanger is taken by the manifold support rods (Part No. 7670-12022), located between the vessel top head and the heat exchanger lower manifold. As the system temperature rises, the manifold support rods, made of type 416 stainless steel, unload due to their lower expansion coefficient. This feature prevents overstressing of the return lines if they cool faster than the manifold support rods after a reactor scram. Another function of the manifold support rods is to prevent lateral rocking of the heat exchanger about an axis passing through the two return lines.

All computed stresses were compared to allowable stress intensities established by the ASME Boiler and Pressure Vessel Code for Nuclear Vessels (Section III). A complete summary of calculated stress levels is given in the Atomics International report on this reactor, ref. 1, as well as in refs. 14 and 15.

Although the reactor was not originally designed as a nuclear vessel that complies in all details with the ASME Code Section III, it has been reviewed by ORNL engineering personnel with the ASME Code as a reference. As a result several changes were made in design to facilitate final assembly at ORNL and inspection of the welds joining the head to the vessel and the parts of the heat exchanger to each other. Thus all welds in the NaK containing system will be inspected and the pressure containing system will be tested to code requirements, although the vessel was not designed as an ASME code vessel as such.

II. FACILITY DESIGN SUMMARY

1. General Description

The installation of the SNAP reactor at the Tower Shielding Facility constitutes to a certain extent the installation of a new facility in parallel with the existing TSR-II installation, because every effort has been made not to disable or disturb the TSR-II. Thus the Tower Shielding Facility should be able to revert to TSR-II operation with a minimum of effort.

The control console for the SNAP reactor will be located adjacent to the TSR-II control console in the underground control building. The wiring for controls and instrumentation between the SNAP reactor and its console will be completely independent of that for the TSR-II. All laboratories, shops, offices, as well as all utilities, will be used during the SNAP experiments without major alteration to the structures. However, filtering systems for the ventilating air intakes of Buildings 7702 and 7704 which contain the control room, offices, and shops will be added.

The principal new additions to the facility will be the reactor rotator assembly, the reactor maintenance area and shield, and the concrete slabs that will cover the shield mapping detector chamber. In addition, a gravity drain will be added to the underground silo and additional lighting will be provided for both the experiment and maintenance areas.

Figure 13 shows a plot plan of the facility with locations of both old and new equipment indicated. A description of the facility as it existed before addition of the SNAP experiment is contained in Appendix I.

2. Description of Reactor Rotator

The reactor will be suspended from a boom that extends out from a vertical column mounted on the southernmost concrete slab over the detector chamber. The structural strength of the assembly is such that a nominal 1,500 lbs. can be suspended at the end of the boom with a factor of safety of 4 to 5 over yield stress included in the design. The entire assembly is called the reactor rotator, shown in Fig. 14.

Three motions are possible with the mechanisms incorporated into the reactor rotator: column rotation, reactor rotation, and vertical motion of the reactor. Motion interlocks are included that prevent column rotation and vertical motion while the reactor is operating. Rotation of the reactor about its own axis will be required for the program of experiments.

The column rotator makes it possible to move the reactor along an arc of 11 ft 10 in. radius through a total rotation of the column of about

Fig. 13. Plot Plan of Tower Shielding Facility with Location of SNAP Reactor Installation.

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Fig. 14. Reactor Support Structure.

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180°. The two extremes of position of the reactor are over the detector chamber and inside the storage shelter. The motion can be stopped at any intermediate position and is rate variable from 0 to 1/10 rpm. The maximum linear speed of motion of the reactor will thus be 1/8 ft per second. The normal mode of operation will include an acceleration period at the initiation of motion, and a deceleration period of about 17-1/2 seconds when a stop is called for, before power is completely cut off. In addition, it will be possible to move the boom around at the lowest speed only for fine positioning. This lowest speed will be selected through a variac setting when the rotator is assembled and will probably be in the range of 1/10 of the maximum speed.

One set of controls for this motion will be located in the control room at the main control console and another duplicate set will be located at the storage shelter shield wall, next to the viewing window. A schematic diagram and a listing of controls are given in the Appendix.

The mechanical drive system for rotation of the reactor and vertical motion of the reactor is shown schematically in Fig. 15.

The reactor rotation motion makes it possible to turn the reactor about its own vertical axis. The motion possible is nearly 180° in both directions of rotation. The speed of rotation will be variable between the limits of 1/45 rpm and 1/5 rpm. A set of controls for this motion is located at the main control console, with a duplicate set at the window of the storage shelter shield wall.

The vertical motion of the reactor will make it possible to position the bottom of the reactor at any elevation between 9 ft 3-1/2 in. and 12 ft 9-1/2 in. above the plane of the concrete pad in the maintenance area. The vertical motion is accomplished through a nut and screw arrangement with the nut rotating and the lead screw translating only. The rate of vertical motion of the reactor is constant at 1.7 in. per minute.

The structure of the boom and column is designed primarily to limit deflections and the stress levels achieved in the various structural members are therefore quite low. In an effort to minimize the amount of material used in the boom, a guy is used to take some of the bending load

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Fig. 15. Schematic Diagram of Mechanical Drive Systems for Reactor Rotation and Elevation Motions.

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out of the 6-in. aluminum alloy channels that make up the boom. The guy runs from the top of the column to the end of the boom. Should the guy break, however, the stresses in the boom, with a 2,000-lb. load suspended, would still reach only 33,000 psi, well below the ultimate strength of 6061-T6 aluminum.

The part from which the reactor is suspended directly is a hollow shaft with an external thread. The internal diameter of the shaft is 2 in. and the root diameter of the external thread is 2.664 in., thus making a structural tube with an effective wall thickness of some $5/16$ in. This tube wall is penetrated by a key slot to a depth of 0.15 in. which reduces the tensile area by some 0.037 in.^2 , a negligible amount. The direct tensile stresses then, in this shaft, for any loads currently anticipated are substantially below $1,000 \text{ lb./in.}^2$. The net tensile area of the shaft is 2.31 in.^2 and even if 2,000 lb. were suspended here, the simple tensile stress does not exceed 850 lb./in.^2 . The other stresses that may be induced in the shaft are correspondingly low to give ample factors of safety over yield stresses in every conceivable situation. A drawing of the rotation and elevation mechanism is given in the Appendix.

3. Maintenance Area and Shield Wall

The maintenance area for the SNAP reactor installation is a 14-ft square portion of the concrete pad immediately to the south of the detector chamber. It is enclosed on three sides by solid concrete block walls as shown in Fig. 16. There will be a roof over the area so that at least partial protection from the weather will be provided, and during assembly of the reactor additional protection can be provided by hanging canvas across the open fourth side.

The maintenance area will serve several purposes. It will provide a place where the reactor parts can be brought, gathered, and assembled. It will be the location for the core-mapping collimator during the initial, low-power stage of reactor operation. It will be the shielded area into which the reactor may be rotated to protect it from weather and to shield the rest of the TSF area from gamma-ray radiation.

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Fig. 16. Maintenance Area and Shield Wall.

After the reactor becomes sufficiently radioactive so that direct inspection is no longer possible, the viewing window in the east wall of the maintenance area shelter will provide a visual access. The thickness of the east wall, 32 in., was chosen so as to provide shielding against the maximum anticipated fission product concentration in the reactor core and maintain a dose rate of 2-1/2 mr/hr at the outer wall face when the reactor is 3 ft from the inner wall face.

The viewing window is of a standard hot cell design with laminated inner and outer panes. The inner pane (the hot side) is made of two 1-in.-thick plates of 2.7 density annealed (non-browning) glass with a laminating interface of polyvinyl butyral. The outer pane (the cold side) is made of two 3/4-in.-thick plates of 2.5 density glass. The window cavity will contain a 2.50 to 2.54 density zinc bromide aqueous solution having the standard luminous transmittance. Around the window frame, wells made of pipe are provided for electrical heaters to provide protection against freezing of the solution in cold weather. A total of ten wells are provided for 100 to 150 watt cartridge-type resistance heaters.

4. Shield Mapping Detector Chamber

This chamber will be created by converting the TSR-II handling pool into a dry chamber. This will be accomplished primarily by removing the drain valve from the pool, thus leaving an unobstructed gravity drain that prevents accumulation of water.

During most of the experiments, the detector chamber will have to be covered over by 3 reinforced concrete slabs, 2 ft thick as shown in Fig. 16. The southernmost of these slabs incorporates the base plate on which the reactor rotator mechanism is mounted. This slab will remain in position, therefore, indefinitely. The center and northernmost slabs, however, will be removable, thus providing an access opening to the top of the chamber, 15 ft wide by 20 ft long. This opening will be required for installation of the large spherical collimator to be used for mapping of the dose transmitted through the lithium hydride shields. These shields will be positioned directly over the circular opening in the center slab and will be supported on a structure of piled concrete blocks, about 5 1/2 ft high and

12 ft square in plan view. The structural strength of the slabs is adequate to support their own weight plus the weight of the shield support blocks and shield collar.

The neck of the present pool, which is a 4-ft wide by 12-ft long auxiliary pool to the north of the main section, will become the access passage into the detector chamber. A stairs and ladder combination will allow personnel to descend into the detector chamber. This passage will remain open at all times.

5. Maintenance Table and Manipulators

A table that will be capable of vertical motion of the working surface over a total travel of 2 ft 8 in. will be located inside the maintenance area. The lift mechanism will be a standard purchased item. The table is rated for a lift capacity of 2,000 lbs.

The table top will be made of flat 6061-T6 aluminum alloy plate so that various jigs and fixtures can be mounted on the table through holes tapped into the aluminum. Probably the most important of these fixtures will be the one for positioning and holding the weather shield (see Topic 8).

The elevation of the maintenance table is such that with the reactor in its lowest elevation position and the table in its uppermost, the bottom flange of the reactor is just even with the table top. Thus the possibility of running the reactor into the table is minimized.

The principal uses for the table will be: (1) to position and hold the reactor vessel during assembly of the core and during the welding of the vessel head to the heat exchanger; (2) to position and hold the reflector installation carriage around the reactor for the final removal of the poison sleeve and installation of the beryllium reflector assembly; (3) to position and hold the weather shield when it is to be placed around or removed from the reactor; (4) to hold the SNAP shield or shields that may be assembled to the reactor for structural scattering experiments; and (5) to hold the vessel in which the reactor will be placed for storage when it is finally removed from the reactor rotator.

Two Model-8 manipulators will be installed on the east shield wall of the maintenance area. They will be available throughout the program to perform various remote operations, only two of which can be predicted at present. These two are: clamping and unclamping the Marman clamp that joins the reactor to the SNAP-2 shield; and disengaging the reactor from the rotator after final shutdown. The process of disengaging the reactor will itself involve two major steps: loosening the bolts at the mechanical interface between reactor and rotator; and cutting control wires and the pneumatic tube for the coarse control actuators.

6. Storage Pit

As shown in Fig. 17, a deep concrete-lined pit, sometimes referred to as an "underground silo," is available just to the south of the maintenance area for storage of the reactor after the program of experiments is completed. A gravity drain will be added to this facility, as an addition to the sump pump that is already installed. This should prevent any accumulation of water in the pit.

7. Lighting

Lights will be added to the area to illuminate the reactor and the maintenance area at night. Lights will also be added to the detector chamber and will be necessary both at night and in the daytime since the chamber will usually be covered by the concrete slabs.

8. Auxiliary Equipment

A special piece of auxiliary equipment that is designed specifically for this installation is the so-called reactor weather shield. This is shown in Fig. 18 and is a thin cylindrical shell of stainless steel that has 10 kw of electrical heating elements mounted on the inside surface.

This cylinder can be brought up around the reactor and heat exchanger and hooked into a circular top plate that is attached to the flange at the bottom of the reactor rotator lead screw. At its lower end, the weather shield will be restrained from lateral motion by a circular flanged ring that is attached to the SNAP shield lifting ring.

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Fig. 17. Underground Silo at the TSF.

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Fig. 18. Weather Shield, Table and Weather Shield Guide.

The purpose of the weather shield is to protect the reactor from moisture and from extremes of weather, although no part of the reactor is catastrophically vulnerable to rain or snow as such. It can also serve to reduce rates of temperature drop of the system after shutdown on cold or windy days. Subject to requirements of the experimental program, the weather shield will be installed around the reactor whenever necessary and particularly during periods of shutdown of the facility.

III. DESCRIPTION OF THE SNAP-10A REACTOR CONTROL AND SAFETY SYSTEMS

1. Nuclear Instrumentation

Startup Channel. The startup channel includes a fission counter, preamplifier, linear amplifier, and log count rate meter with level and period outputs. The level and period outputs are displayed on two strip chart recorders. The range of the counter channel is five decades.

The fission counter is suspended from a boom which can be rotated in a horizontal plane so that the counter can be shielded from the reactor by the wall of the reactor shed during high power operation. The positions of the reactor boom and the fission counter support boom are displayed on a dual synchro receiver located on the operating console. Interlocks are provided to prevent physical interference between the two booms.

Log N Channel. The log N channel consists of a compensated ionization chamber (one half of a dual section chamber), a log N amplifier, period amplifier, and composite amplifier. The range of this channel extends about six decades down from full power. The instrument chassis used in this channel are the units normally used for the Tower Shielding Reactor. The interconnecting wiring for the channel is an integral part of the SNAP-10A installation to facilitate transfer of the instruments with minimum delay after shutdown of the TSR. The ionization chamber is mounted on the reactor boom at a location about 7 ft from the reactor with some lateral adjustment available for calibration.

The outputs of the log N, both level and period, are displayed on strip chart recorders. Interlocks on level and period provide extension of the startup control through the intermediate power range. The composite

safety amplifier consisting of a period amplifier and a sigma amplifier is used to provide a fast period scram.

Level Safety Channels. Two level safety channels are used to monitor neutron flux level over the upper two decades of power and provide level scram signals to the magnet amplifiers controlling the air supply to the coarse control (scram) reflector drums. The safety channel sensors are non-compensated ionization chambers. One safety chamber shares the gas envelope with the log N chamber described above; the other shares an envelope with the compensated chamber of the linear flux channel described later. In addition to the ionization chamber, each channel consists of a preamplifier and safety amplifier. The preamplifiers and sigma amplifiers are those normally used in the Tower Shielding Reactor.

The outputs of these two channels are displayed on meters on the safety trouble monitor that is also a part of the normal instrumentation complement of the TSF.

Linear Flux Channel. The linear flux channel consists of a compensated ionization chamber (one half of a dual chamber), a compensated chamber high voltage power supply and a picoammeter with range change capability. The output of the channel is displayed on a strip chart recorder and is also used as the input to an automatic flux controller. The controller is used to position one of the two fine control drums in the automatic, or servo, mode of operation. The range of automatic control is expected to extend over about six decades up to full power.

Figure 19 is an instrument block diagram showing the functional arrangement of the components of the various channels.

2. Reactor Non-Nuclear Instrumentation

The reactor non-nuclear instrumentation includes six thermocouples to measure the coolant temperature, three monitoring the inlet to the reactor and three monitoring the outlet. In addition, capability is included for attaching thermocouples to points on the reactor assembly as required.

Fig. 19. Instrument Block Diagram.

Coolant flow information is derived from the electromagnetic pump by accurate measurement of the voltage and current supplied to the pump. The pump will be calibrated against current and voltage, by means of a venturi meter and the pumping and heating facilities of the NaK cart supplied by Atomics International, over a range of pumping heat that will cover the resistance of the system at all flows up to $4\frac{1}{2}$ G.P.M.

3. Control Drums

Fine Control Drums. The fine control drums (reflectors) are positioned by stepping motors. Figure 12 is a schematic diagram of the basic features of the fine control drum system that is subject to automatic control. Each drum has a variable reluctance position transducer to monitor the angular position of the drum. A demodulator is used to convert the ac phase sensitive signal to a 0-5 volt dc ramp for display and control. The voltages for both of the drums are displayed on a dc voltmeter calibrated in degrees of rotation. In addition, the voltage from the demodulator associated with fine drum number 2 (which is the automatically controlled drum) is used to drive a strip chart recorder pen movement. The purpose of the recorder movement is to provide switch actuation representing discrete drum positions to limit the span of automatic control to about $0.5\% \Delta K/K$.

The stepping motors are actuated by successively energizing four stator phase windings, with each step causing $1/4^\circ$ of rotation of the fine drum. The switching is accomplished by stepping motor controllers which receive their input from switches on the console or from the automatic flux controller (in the case of fine drum number 2 when the reactor is in the automatic control mode). The control logic for the reactor, including control drum actuation, is shown on Fig. 20, the block diagram RC 16-1-1B.

Each fine drum drive actuates three limit switches, two at the withdraw limit and one at the insert limit. In addition to providing the normal readout function and rotation limiting function, the two withdraw limit switches are used to prevent insertion of either of the coarse control drums unless both fine drums are fully withdrawn. This interlock

is used to prevent achieving criticality by insertion of the coarse drums, since the rate of insertion of reactivity by the pneumatic drives of the coarse drums may be greatly increased by certain malfunctions.

Coarse Control Drums. The two coarse control drums are pneumatically inserted to add reactivity. The actuating piston is spring loaded to drive the coarse drums out when air pressure is relieved by a scram or by a manual withdraw request. Figure 11 is a schematic diagram of the basic features of the pneumatic system. The normal insertion time for one coarse drum is 3 seconds. The reactivity change as a function of time for a scram is shown in Fig. 21.

A scram solenoid valve is energized to insert a drum, by a high current magnet amplifier which is controlled by the sigma bus voltage in the conventional ORNL safety system. Off-normal sigma bus voltage cuts off the current to the valve to produce a fast scram. A slow scram or manual withdrawal of a coarse drum is accomplished by interrupting the solenoid supply by a relay within the magnet amplifier.

Four position switches are installed on each coarse drum drive. Two are actuated at the insert limit and two at the withdraw limit. These switches are also used to enforce the proper sequence of control drum insertion. Two switches at each limit are used to provide redundancy for improved reliability. Both coarse drums must be completely inserted before either fine drum can be inserted. Furthermore, neither coarse drum can be inserted unless the other is either fully inserted or fully withdrawn. The latter limitation prevents simultaneous insertion of both coarse drums. The status of each switch is indicated on the control console by pilot lights.

4. Reactor Control Logic

The reactor control system (Fig. 20) is designed to facilitate routine startup, steady-state operation, and shutdown of the reactor. Automatic flux level control is available over about six decades of power and is accomplished by automatic positioning of the No. 2 fine control drum. The span of drum motion over which the automatic system has control

Fig. 21. Reactivity as a Function of Time for a Scram and for Insertion of Coarse Drum.

is limited to about one-fourth the total motion of the drum. The flux set point of the controller is adjusted by the operator by the picoammeter range selector switch and a vernier control.

The reactor operating controls available at the console include a key switch to permit reactor operation, individual actuator switches for the four control drums, a switch to permit simultaneous insertion or withdrawal of both fine drums, a scram switch, picoammeter range selector, servo set point vernier control, annunciator acknowledge and reset buttons, slow scram relay test push buttons, servo on and off pushbuttons, fission chamber motion pushbuttons, reactor vertical and rotational motion push-buttons, and reactor support boom rotation pushbuttons. A side panel on the console will contain actuating switches for various collimator and shield motions necessary for the experimental program.

The positions of the reactor boom, the fission chamber, the azimuthal position of the reactor, and the positions of the two fine control drums are displayed as analog readouts on the console. Pilot lights on the console indicate the actuation of various limits switches associated with the positions of the reactor, fission chamber, and control drums. Pilot lights also are used to indicate the functional status of certain parts of the control systems. A console arrangement is shown in Fig. 22 (RC16-2-1) along with a list of switches and lights. Figure 23 is an elementary diagram of the pilot lights and annunciator circuits (RC16-1-2C).

The control logic is enforced by control system interlocks. Applicable interlocks presently in use at the Tower Shielding Facility for personnel control are retained and are included as part of the reactor control system. Startup is supervised by period signals from either the counting channel or the log N channel. This supervisory control consists of control drum insertion "inhibits" and automatic withdrawal of the fine drums only. Reverses are also used to withdraw the fine drums when reactor outlet temperature, differential temperature, or flux exceeds preset levels. The reverse action is used in this way to correct abnormalities and to avert, where possible, the necessity for initiation of a scram. Elementary diagrams, Figs. 24, 25, 26, and 27, show the electrical circuits used to accomplish the various control functions. Interlocks with the reactor rotator motion circuits are provided to prevent motion of the

ITEM NO	EOP	FUNCTION	REMARKS	ITEM NO	EOP	FUNCTION	REMARKS
1	I 519	"Area Secure" Indicator	Master Specialties Co Cat #IDEAC2J3L(W)NIRIV3 AREA SEC 10P	36A	I 551	"Fiss Ch Rotation Inhibit" Indicator	Master Specialties Co Cat #IDEAC2J3L(W)NIRIV7 F, C, INHIBIT F, C, CW LIMIT
2	I 520	"Operate Permit" Indicator	MS Co Cat #IDEAC2J3L(W)NIRIV3 OPERATE PERM T	36B	I 550	"Fission Chamber CW Limit" Indicator	
3	S 331	Operate Switch	G.E Type SBM See RC16 1 2D for development	37A	I 534	"Reactor Boom CW Decelerate Limit" Indicator	Master Specialties Co Cat #IDEAC2J3L(W)NIRIV7 REACTOR BOOM, CCW DEC LIM REACTOR BOOM, CCW LIMIT
4A	I 521	"Scram No.1" Indicator		37B	I 535	"Reactor Boom CCW Limit" Indicator	
4B	I 522	"Scram No.2" Indicator	MS Co Cat #IDEAC2J3L(W)NIRIV25 SCRAM, NO.1 SCRAM, NO.2	38A	I 536	"Reactor Boom CW Decelerate Limit" Indicator	Master Specialties Co Cat #IDEAC2J3L(W)NIRIV7 REACTOR BOOM, CW DEC LIM REACTOR BOOM, CW LIMIT
4C	S 332	Scram Reset Pushbutton		38B	I 537	"Reactor Boom CW Limit" Indicator	
5	S 332	Scram Switch	G.E Type SBM See RC16 1 2D for development	39A	I 549	"Fission Chamber CW Motion Indicator"	MS Co Cat #IDEAC1FJ3L(W)NIRIV4 ROTATE, F, C, CCW
6	NPI	Nameplate	Engraved "SCRAM NO.1 RELAY TEST" See RC16 for details	39B	I 503	"Rotate Fission Chamber CCW" Pushbutton	
7	S 41	Scram No.1 Test Pushbutton	Micro Switch No. 2 PB11	40	I 501	"Stop Fission Chamber Rotation" P.B	MS Co Cat #IDEAC1FJ3L(W)NIRIV4 STOP, F, C, ROTATION
8	NPI2	Nameplate	Engraved "SCRAM NO.2 RELAY TEST" See RC16 for details	41A	I 548	"Fission Chamber CW Motion Indicator"	
9	S 42	Scram No.2 Test Pushbutton	Micro Switch No. 2 PB11	41B	I 509	"Rotate Fission Chamber CW Pushbutton"	MS Co Cat #IDEAC1FJ3L(W)NIRIV4 ROTATE, F, C, CW
10A	I 525	Ann Ack Indicator (Flashing)		42A	I 532	"Reactor Boom CW Motion Indicator"	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 ROTATE, REACTOR BOOM, CCW
10B	S 43	Annunciator Acknowledge Pushbutton	MS Co Cat #IDEAC1FJ3L(W)NIRIV3 ANNUNCIATOR, ACKNOWLEDGE	42B	I 555	"Rotate Rea for Boom CCW" Pushbutton	
11	S 44	Annunciator Reset Pushbutton	MS Co Cat #IDEAC1FJ3L(W)NIRIV3 ANNUNCIATOR, RESET	43	I 560	"Stop Reactor Boom Rotation" Pushbutton	MS Co Cat #IDEAC1FJ3L(W)NIRIV4 STOP, REACTOR BOOM, ROTATION
12A	I 500	C.D.No.1 "Wdr Limit" Indicator No.1	MS Co Cat #IDEAC2J3L(W)NIRIV25 WDR, LIMIT WDR, LIMIT	44A	I 531	"Reactor Boom CW Motion Indicator"	
12B	I 501	C.D.No.1 "Wdr Limit" Indicator No.2		44B	I 559	"Rotate Reactor Boom CW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 ROTATE, REACTOR BOOM, CW
13A	I 502	C.D.No.1 "Insert Limit" Indicator No.1	MS Co Cat #IDEAC2J3L(W)NIRIV25 INS, LIMIT INS, LIMIT	45	S 54	"Jog Reactor Boom CCW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 JOG, REACTOR BOOM, CCW
13B	I 503	C.D.No.1 "Insert Limit" Indicator No.2		46	I 551	"Emergency Stop R Boom Rotation Pushbutton"	MS Co Cat #IDEAC1FJ3L(W)NIRIV4 EMERGENCY STOP, REACTOR BOOM
14	S 33	Coarse Drum No.1 Actuate Switch	G.E Type SBM See RC16 1 2D for development	47	S 57	"Jog Reactor Boom CW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 JOG, REACTOR BOOM, CW
15	I 514	"Solenoid Power" Indicator	MS Co Cat #IDEAC1J3L(W)NIRIV3 SOLENOID POWER	48A	TR 638	Reactor Rotation Position Synchro Receiver	Scientific Atlanta Model No. 4406 Dual Synchro Receiver
16A	I 507	C.D.No.3 "Wdr Limit" Indicator No.1	MS Co Cat #IDEAC2J3L(W)NIRIV25 WDR, LIMIT WDR, LIMIT	48B		"Synchro Receiver (Not used)"	
16B	I 508	C.D.No.3 "Wdr Limit" Indicator No.2		49	I 547	"Reactor Lower Limit" Indicator	MS Co Cat #IDEAC1J3L(W)NIRIV4 REACTOR, LOWER, LIMIT
17A	I 509	C.D.No.3 "Insert Limit" Indicator No.1	MS Co Cat #IDEAC2J3L(W)NIRIV25 INS, LIMIT INS, LIMIT	50	I 546	"Reactor Upper Limit" Indicator	MS Co Cat #IDEAC1J3L(W)NIRIV4 REACTOR, UPPER, LIMIT
17B	I 510	C.D.No.3 "Insert Limit" Indicator No.2		51	I 543	"Reactor CCW Limit" Indicator	MS Co Cat #IDEAC1J3L(W)NIRIV4 REACTOR, CCW, LIMIT
18	S 35	Coarse Drum No.3 Actuate Switch	G.E Type SBM See RC16 1 2D for development	52	I 542	"Reactor CW Limit" Indicator	MS Co Cat #IDEAC1J3L(W)NIRIV4 REACTOR, CW, LIMIT
19	I 518	Servo "Insert Error" Indicator	MS Co Cat #IDEAC1J3L(W)NIRIV3 INSERT ERROR	53A	I 544	"Reactor Up Motion" Indicator	
20	I 517	Servo "Withdraw Error" Indicator	MS Co Cat #IDEAC1J3L(W)NIRIV3 WITHDRAW, ERROR	53B	I 593	"Raise Reactor" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV3 RAISE, REACTOR
21A	I 515	FD No.2 "Out of Auto Range" Indicator	MS Co Cat #IDEAC2J3L(W)NIRIV30 OUT OF RANGE AUTO, WDR, LIMIT	54A	I 545	Reactor "Down Motion" Indicator	
21B	I 516	FD No.2 "Auto Wdr Limit" Indicator		54B	I 591	"Lower Reactor" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV3 LOWER, REACTOR
22	I 448	Fine Drum No.2 Position Meter	Westinghouse Type KK 241, 0.5V J-C, Style #291B201A11	55A	I 539	Reactor "CCW Motion Indicator"	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 ROTATE, REACTOR, CCW
23A	I 523	"Servo On" Indicator	MS Co Cat #IDEAC1FJ3L(W)NIRIV3 SERVO, ON	55B	I 577	"Rotate Reactor CCW" Pushbutton	
23B	S 40	Servo On Pushbutton		56	I 583	"Stop Reactor Rotation" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 STOP, REACTOR, ROTATION
24A	I 524	"Servo Off" Indicator	MS Co Cat #IDEAC1FJ3L(W)NIRIV3 SERVO, OFF	57A	I 538	Reactor "CW Motion" Indicator	
24B	S 40	Servo Off Pushbutton		57B	I 575	"Rotate Reactor CW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 ROTATE, REACTOR, CW
25A	I 504	FD No.2 "Wdr Limit" Indicator No.1	MS Co Cat #IDEAC2J3L(W)NIRIV25 WDR, LIMIT WDR, LIMIT	58A	I 540	"Increase Rotation Speed" Indicator	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 INCREASE, ROTATION, SPEED
25B	I 505	FD No.2 "Wdr Limit" Indicator No.2		58B	I 579	"Increase Rotation Speed" Pushbutton	
26	I 506	FD No.2 "Insert Limit" Indicator	MS Co Cat #IDEAC2J3L(W)NIRIV3 INSERT, LIMIT	59A	I 541	"Decrease Rotation Speed" Indicator	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 DECREASE, ROTATION, SPEED
27	S 34	Fine Drum No.2 Actuate Switch	G.E Type SBM See RC16 1 2D for development	59B	I 581	"Decrease Rotation Speed" Pushbutton	
28	S 37	F Drums Nos 2 & 4 Group Actuate Sw	G.E Type SBM See RC16 1 2D for development	60	I 505	"Jog Fission Chamber CCW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 JOG, F, C, CCW
29	I 449	Fine Drum No.4 Position Meter	Westinghouse Type KK 241, 0.5V J-C, Style #291B201A11	61	I 507	"Jog Fission Chamber CW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 JOG, F, C, CW
30A	I 511	FD No.4 "Wdr Limit" Indicator No.1	MS Co Cat #IDEAC2J3L(W)NIRIV25 WDR, LIMIT WDR, LIMIT	62	I 571	Jog Reactor "CCW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 JOG, REACTOR, CCW
30B	I 512	FD No.4 "Wdr Limit" Indicator No.2		63	I 573	Jog Reactor "CW" Pushbutton	MS Co Cat #IDEAC1F2J3L(W)NIRIV4 JOG, REACTOR, CW
31	I 513	FD No.4 "Insert Limit" Indicator	MS Co Cat #IDEAC2J3L(W)NIRIV3 INSERT, LIMIT	64A	I 526	"EM Pump On" Indicator	
32	S 36	Fine Drum No.4 Actuate Switch	G.E Type SBM See RC16 1 2D for development	64B	I 545	"Start EM Pump" Pushbutton	
33	I 533	Boom Speed Up + 1 Lower Limit Ind	MS Co Cat #IDEAC1J3L(W)NIRIV4 BOOM SPEED, CONTROL LOWER LIMIT	65A	I 527	"EM Pump Off" Indicator	
34A	TR 638	Reactor Boom Position Synchro Receiver	Scientific Atlanta Model No. 4406 Dual Synchro Receiver	65B	I 546	"Stop EM Pump" Pushbutton	MS Co Cat #IDEAC1FJ3L(W)NIRIV3 STOP, EM PUMP
34B	TR 638	Fiss Ch Position Synchro Receiver					
35A	I 553	"Boom Rotation Inhibit" Indicator	Master Specialties Co Cat #IDEAC2J3L(W)NIRIV7 BOOM, INHIBIT F, C, CCW LIMIT				
35B	I 552	"Fission Chamber CCW Limit" Indicator					

Fig. 22. Reactor Control Console Arrangement.

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Fig. 25. Elementary Diagram Drum Controls - Sheet 2.

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Fig. 26. Elementary Diagram, Safety Trouble Monitor.

S31- Operate Switch (Key-lock in handle removable only in "Off" position)

CONTACTS (HANDLE END)	POSITIONS (REAR VIEW)	GRID No.
(1) A	OPERATE	14
(2) B	OFF	17
(3) C	RESET	6
(4) D		410
(5) E		402
(6) F		306
(7) G		11
(8) H		15
(9) I		Sp.
(10) J		Sp.
(11) K		560
(12) L		136
(13) M		Rec.
(14) N		Rec.
(15) O		Rec.
(16) P		Rec.
(17) Q		Rec.
(18) R		Rec.
(19) S		Rec.
(20) T		Rec.
(21) U		Rec.
(22) V		Sp.
(23) W		Sp.
(24) X		Sp.

S32- Manual Scram Switch

CONTACTS (HANDLE END)	POSITIONS (REAR VIEW)	GRID No.
(1) A	SCRAM	402
(2) B	NORMAL	405
(3) C	SCRAM	419
(4) D		421
(5) E		
(6) F		
(7) G		
(8) H		

S33- Drum #1 Actuate Switch
S35- Drum #3 Actuate Switch

CONTACTS (HANDLE END)	POSITIONS (REAR VIEW)	GRID No.
(1) A	INSERT	313
(2) B	HOLD	319
(3) C	WDR	317
(4) D		317
(5) E		
(6) F		
(7) G		
(8) H		

Spring Return

S34- Drum #2 Actuate Switch
S36- Drum #4 Actuate Switch

CONTACTS (HANDLE END)	POSITIONS (REAR VIEW)	GRID No.
(1) A	INSERT	313
(2) B	HOLD	316
(3) C	WDR	325
(4) D		328
(5) E		
(6) F		
(7) G		
(8) H		

Spring Return

S37- Drums #2 & 4 Group Actuate Switch

CONTACTS (HANDLE END)	POSITIONS (REAR VIEW)	GRID No.
(1) A	INSERT	309
(2) B	HOLD	
(3) C	WDR	
(4) D		314
(5) E		315
(6) F		324
(7) G		329
(8) H		

Spring Return

S47- TSRI-SNAP 10A Selector Switch (Key-lock in handle removable in either switch position)

CONTACTS (HANDLE END)	POSITIONS (R.V.)	GRID No.
(1) A	TSRI	5
(2) B	SNAP 10A	6
(3) C		11
(4) D		13
(5) E		15
(6) F		15
(7) G		136
(8) H		136
(9) I		66
(10) J		Sp.
(11) K		184
(12) L		184
(13) M		185
(14) N		575
(15) O		184
(16) P		574
(17) Q		185
(18) R		573
(19) S		Sp.
(20) T		306
(21) U		Sp.
(22) V		Sp.
(23) W		Sp.
(24) X		Sp.

I.C. No.	LOCATION	OPERATION AND SETPOINT	GRID No.
1	Count Rate Period Recorder	Closes Count Rate $T > 20$ sec.	313
2	Log N Period Recorder	Closes Log N $T > 20$ sec.	313
3	Reactor ΔT Recorder	Closes Reactor $\Delta T > 1.0$ °F (Reverse)	349
4	Safety Trouble Monitor	Opens on 1 Safety Trouble	572
5	Safety Trouble Monitor	Closes $\phi > 120\%$	333
6	Reactor Outlet Temp. Recorder	Closes $T_o > 1.0$ °F (Reverse)	350
7	Count Rate Meter	Closes CRM in "Use"	341
8	Count Rate Recorder	Closes < 8000 cps.	341
9	Count Rate Recorder	Closes > 2 cps	341
10	Log N Amplifier	Closes Log N Amplifier in "Operate"	342
11	Log N Recorder	Closes $N > .001 N_0$	342
12	Drum #2 Position Recorder	Closes Drum #2 within "Auto Range"	343
13	Drum #2 Position Recorder	Opens Drum #2 when beyond "Auto Swing Wdr Limit"	344
14	Count Rate Period Recorder	Closes Count Rate $T < 5$ sec.	345
15	Log N Period Recorder	Closes Log N $T < 5$ sec.	346
16	*	Opens Reactor Coolant Flow Low	569
17	*	Opens Pump Current Low	569
18	Reactor Outlet Temp. Recorder	Opens $T_o > 1.0$ °F (Scram)	351
19	Reactor ΔT Recorder	Opens $\Delta T > 1.0$ °F (Scram)	352
20	Safety Trouble Monitor	Opens Fast Scram	570
21	Safety Trouble Monitor	Opens $\phi > 120\%$	563
22	Safety Trouble Monitor	Opens on 1 Safety Trouble	412
23	Safety Trouble Monitor	Closes Period Normal	342
24	Log N Recorder	Closes Log N > 1.0	353
25	Monitron Monitor	Opens 2 of 3 Monitrons High Level or Trouble	402
26	Monitron Monitor	Closes 2 of 3 Monitrons High Level	409
27	Serve Amplifier	Closes Serve Insert Error	347
28	Serve Amplifier	Closes Serve Withdraw Error	348

*To be determined later

NO.	REVISIONS	DATE	APPD	APPD
1	DATE	DATE	DATE	DATE
2	DATE	DATE	DATE	DATE
3	DATE	DATE	DATE	DATE
4	DATE	DATE	DATE	DATE
5	DATE	DATE	DATE	DATE

REFERENCE DRAWINGS		DWG. NO.
DESIGN RESPONSIBILITY		
SNAP 10A-TSF		
ELEMENTARY DIAGRAM CONTACT TABULATIONS		
INSTRUMENTATION AND CONTROLS DIVISION OAK RIDGE NATIONAL LABORATORY		
OPERATED BY UNION CARBIDE CORP.		
SUBMITTED	APPROVED	DATE
R. C. Odom	J. H. Smith	11/10/66
RC16-1-2D Ro		

LIMITS ON DIMENSIONS UNLESS OTHERWISE SPECIFIED
FRACTIONS: $\pm$ _____
DECIMALS: $\pm$ _____
ANGLES: $\pm$ _____
SCALE: _____

Fig. 27. Elementary Diagram, Contact Tabulations.

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reactor when the pneumatic control valves for the two coarse drums have power available. Only the column rotation and vertical motion of the reactor are inhibited in this way. Rotation of the reactor about its own axis is permitted.

5. Nuclear Safety System

Protection of the reactor against an uncontrolled nuclear excursion is provided by the nuclear safety system. This protection is afforded by two level safety channels which initiate a reactor scram when the neutron flux exceeds 150% of design point. The response of the scram is required to be adequate to protect against any credible failure of the control system, and in particular must be able to protect against either the maximum rate or maximum magnitude of reactivity that can be inserted by a malfunctioning control system. A discussion of conceivable malfunction accidents is given in Section VII of the attached ref. 1. Because the coarse control drums are pneumatically inserted, there is concern that certain failures could cause the drums to move toward the core at a rate much greater than normal. Thus, we have provided circuitry to prevent insertion of the coarse drums unless both fine drums are fully withdrawn. Since the reactor loading will be such as to require insertion of at least three drums to achieve criticality, the insertion of both coarse drums will still leave the reactor subcritical by about \$2. The sequence enforcing circuits also prevent insertion of the fine control drums unless both coarse drums are inserted. Inasmuch as the intent of the enforced sequence is to assure adequate time response of the safety system by limiting the response required, the system used for programming is treated as a vital part of the safety system. Redundant limit switches and associated circuits are used and the circuits are made as independent as is feasible, considering the limitations imposed by the reactor configuration.

Although not strictly considered as a safety channel (on the basis that a failure is tolerable) the period channel also is connected to the fast scram circuits. The primary function of the period scram is to terminate a short period low level transient before a significant power level is reached.

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The outputs of the two level safety amplifiers and the period safety amplifier are connected to a sigma bus so that the bus voltage rises as any one safety amplifier output approaches the scram point. The sigma bus is connected to the inputs of the two high current magnet amplifiers which supply the current to the scram solenoids. The magnet amplifiers are designed to cut off the solenoid current when the sigma bus voltage reaches the scram point of 43 volts. They also have bus protected circuits which cut off the current when the sigma bus voltage falls below ~ 35 volts. A relay is included in each magnet amplifier so that scrams can be initiated by non-nuclear instrument channels driving two relay scram channels. A relay scram is identified as a "slow" scram in contrast to the fast, or electronic, scram that derives its input from the sigma bus. Although not required for protection of the reactor, slow scrams are initiated when reactor outlet temperature or reactor coolant differential temperature exceeds preset limits, or the radiation level is sensed by monitrons in the control room area exceeds 7-1/2 mr/hr. The monitrons are connected in a two of three arrangement, with a separate relay matrix driving each of the two slow scram relays. Since the same signal trips both scram relays, a test circuit has been included so that the capability of either relay to initiate a scram independently of the other can be demonstrated periodically.

6. Annunciators and Conditions for Operation and for Scram

Certain abnormal conditions cause actuation of audible alarms with visual indication. These conditions, together with the source of information that causes the annunciation, are listed in Fig. 23. Other annunciators will be included to alert the operator to conditions which may not be abnormal but may require his attention.

In summary, the following are the positive conditions that must exist for operation.

1. A selector switch for directing control power either to the TSR-II or to the SNAP must be turned manually to the SNAP circuit. This is a key switch (S47) located in the control room. The key must be withdrawn after making the selection and used

in the control power key switch at the SNAP control console.

2. The key from step 1 must be used in the key switch (S31B) at the control console. The switch must be reset and turned to the "on" position.
3. The ramp chain switch (XSS-22B) must be energized, i.e., the chain barring the exit from the control house to the reactor area must be in place.
4. Door to ramp must be closed (XS-22C) or ramp monitron must be free of trouble and reading below 7-1/2 mr/hr (XS-21). (This latter condition allows access to the ramp hoist station but not to the reactor area. The hoist station is a shielded area located between the door to the ramp and the chain in step 3.)
5. West gate (ZSW) and north gate (ZSN) must be closed.
6. Back door (on north end of control house) must be closed (XS-22A). An exception to this is explained in step 9.

Steps 3, 4, 5, and 6, when accomplished, constitute the "exclusion area secure" permission circuit as shown on Fig. 24 and energize relay K302.

7. Time delay of 3 min from actuation of key switch in step 2 must elapse and time delay switch T305 must close. (A horn blows in the reactor area during these 3 min, thus warning of impending reactor operation.)

When step 7 is accomplished and "exclusion area secure" circuit is still energized, the "operate permit" circuit becomes energized.

8. Scram reset switch (S38) must be actuated to reset the slow scram relays to normal mode.

After these positive steps the circuit of relay 308 is energized and power for the pneumatic control valves is available so that on signal compressed air may be directed to the coarse drum actuators to drive the coarse drums into the reflector.

9. A low power mode is available that makes it possible for people to leave through the north door of the control building and through the west gate without shutting down the reactor. Bypass request switch HS22B must be actuated and the power level must be down where radiation at the road leading to the west gate is at a safe level. The bypass contacts are K5F and K7A in the exclusion area secure circuit. This bypass mode will not be utilized until the Health Physics Division establishes that radiation levels are permissible and the power level that must not be exceeded for this.

To operate the reactor a number of passive conditions must exist as well. In general, these are the reverse of the conditions that initiate a scram or a reverse or a shutdown of the reactor. An example of the latter is the lack of compressed, dry air, which would not initiate an electrical scram nor a reverse but would make it impossible to drive in or hold in the coarse drums.

Conditions that will initiate a reactor scram once the reactor is operating are as follows:

1. Switch (S31B) is turned off.
2. One of the remote scram switches, HS28A, 28B, 28C, 28D, is actuated. These are emergency switches located in the reactor area and at the west gate that may be used by a person caught outside of the control building to prevent or stop operation of the reactor.
3. The west gate or north gate is opened.
4. The ramp door or north door to the control building is opened.
5. The chain barring the ramp to the reactor area is removed.
6. Scram test 1 or scram test 2 switch is energized.
7. Manual scram switch is energized. This is a switch (S32), and is the normal scram switch used when rapid shutdown by withdrawing both coarse drums is required.

8. ΔT across the core exceeds allowable limit.
9. Outlet temperature exceeds allowable limit.
10. Two of the three radiation monitrons exceed a reading of 7-1/2 mr/hr or an indication of two monitrons malfunctioning is received, or a combination of one malfunction and one high dose rate exists.
11. Power level exceeds 150% of nominal upper level.
12. Log N reactor period shorter than 1 second.

IV. AUXILIARY FACILITIES

1. Television Monitors

There are at present three television cameras in the TSF area. Transmissions from them can be received on screens in the control room, both at the reactor control console and at the data collection area, and in the hoist control station at the bottom of the ramp leading out of Building 7702. There are two screens at the reactor control console, each of which can receive from any one of the three cameras. Two cameras are remotely controllable both in rotation and in focus so that all locations of the experimental area may be observed. For the TSF-SNAP installation one of the cameras that have the "zoom" lens will be relocated so that the reactor storage-maintenance area is completely accessible to television observation.

2. Communication Systems

Communication between the TSF and the main Laboratory area is maintained through three commercial telephone lines and two radio sets. In addition, the general Laboratory public address system feeds into the public address system at the Tower.

These telephone lines are an extension of the general Laboratory telephone network with direct dialing access to any extension in the Laboratory as well as to lines on Oak Ridge, Knoxville, and the FTS nationwide network.

A fourth commercial telephone line provides closed-circuit communication with the guard post at the entrance to the TSF road only. All telephone lines remain active on emergency power, in case of main power failure at the Laboratory.

Two-way radio communication can be maintained with the plant security communication center by means of a transmitter and receiver installed in a rack in the reactor control room, and by a portable receiver-transmitter unit. The fixed radio installation can operate either on main or on TSF emergency power.

Within the Tower Shielding Facility an intercommunication system links the underground control room and offices with the above-ground area. Microphones are available in the main office, at the reactor control console, and outside at the two TSR-II hoist control stations. Jacks for microphones are available so that voice communication can be maintained from any place in the above-ground area. One microphone on an extension cord will be placed at the maintenance area shield wall and another will be placed at the detector chamber.

3. Radiation Monitors and Interlocks

Radiation monitoring at the TSF is effected by a system of stationary, remote reading radiation monitors as well as by portable radiation survey meters of various types that are carried by personnel.

Three of the stationary monitors, both neutron and gamma sensitive, are interlocked with the reactor control system so that a dose rate reading of over $7\frac{1}{2}$ mr/hr on any two of the three monitors scrams the reactor. The three stationary monitors that are capable of scrambling the reactor are located as follows: one just inside the north door of Building 7702, one just inside the south door (the ramp door leading to the reactor area), and one just inside the emergency escape hatch of Building 7704.

Some monitors installed around the area have remote read-outs in the reactor control room, but cannot shut down the reactor automatically. Three such monitors, gamma sensitive only, are located along the TSF area public exclusion fence; one gamma plus neutron monitor is at the 600-ft

exclusion fence at the gate leading into the immediate TSF area; one other is outside the south door of Building 7702 at the ramp hoist station; four gamma electrometers are at present located at the TSF-II pool. These latter four will be relocated for the TSF-SNAP installation but will remain in the area of the SNAP detector chamber and maintenance area shield wall so as to continue effective monitoring for both SNAP and TSR-II operation.

Another gamma radiation monitor is located at the main (west) gate (about 4,000 ft distance) of the TSF exclusion area but can be read locally only. Since the guard at this post is always aware of the operation of a reactor at the TSF, this monitor is observed intermittently during operation. However, a recorder is also present and in continuous operation.

4. Control Building Ventilation System

Air flow in the underground buildings, 7702 and 7704, is maintained by three large air conditioners. Two of the air conditioners draw in fresh air from above ground and the third recirculates air already in the buildings. Heat for the building is supplied through thermostatically controlled electric heating coils located in the air ducts serving the various rooms.

In anticipation of the TSF-SNAP installation, an intake air-filtering system will be installed. This installation will change the ventilation system only to the extent that all air drawn into the buildings will pass through filtration. Two filter assemblies, one at each air intake, will each consist of: one prefilter of fiberglass matt, disposable type; one HEPA (high efficiency particulate) filter; one activated carbon filter purchased to ORNL Spec. XSP-224, for removal of iodine from air. These are standard filter arrays used on other ORNL reactors but usually used to protect the environment from discharge out of the reactor containment.

V. ADMINISTRATION, REVIEWS, AND PROCEDURES

1. Administration

For routine operation of a program of experiments at the TSF, the functional organization shown in Fig. 23 is used.

Direct responsibility for management of the facility rests with the TSF Supervisor. Within the limits of established guidelines approved by the review organizations, he makes assignments of individuals to specific duties and is the only person who can authorize changes in established procedures and job duties.

Actual operation of the reactor, including startup, power level setting, shutdown, and all motion of the reactor, is restricted to those qualified individuals who have been certified by the TSF Supervisor, as outlined in the TSF manual. The only exception to this is the authorization to scram the reactor or open the reactor motion circuits in an emergency situation. The TSF technicians are made familiar with the reactor system and experimental setup so as to provide the necessary assistance during operation and maintenance as well as in case of an emergency.

The experimental supervisor prepares the data plan for conducting shielding experiments and schedules the day-to-day activities connected with the program.

As in the past, the TSF will continue to operate on a restricted access basis so that the number of people in the area is known at all times and each person in the area can be identified through the badge number, issued to both visitors and staff upon entering the TSF area.

2. Reactor and Experiment Reviews

The Tower Shielding Facility is operated by the Neutron Physics Division. The Tower Shielding Facility Staff reviews proposed experiments and may routinely approve them if the maximum change in reactivity that could be caused by the experiment is conservatively less than the total reactivity controlled by the servo system. If the change in reactivity that could occur is more than that controlled by the servo system or if

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Fig. 28. Facility Organization Chart.

other hazards are involved, the staff will orally present a description of the experiment to the Neutron Physics Division Reactor Safety Review Committee. Written approval from the Committee will be required before the experiment can be initiated.

Aside from the intradepartmental safety reviews, the Laboratory maintains a number of standing review committees which report to the Laboratory Director and whose functions are to provide internal safety surveillance independent of the various operating and research divisions. These committees are composed of senior members of the ORNL staff selected for their competence in the particular field, but in general not directly associated with the projects they review. Of these committees, two will be concerned with the TSF-SNAP operations. These are as follows:

The Reactor Operations Review Committee reviews for all ORNL reactors not only safety analysis reports and operating safety limits but also summaries submitted by the reactor supervisors, of their yearly operations including such data as power levels, shutdown experience, and in particular an analysis of unusual occurrences. Consideration is given by the committee to the condition of the operating procedures, maintenance program, personnel changes, and reactor mechanical details which could affect the reactor shutdown margin.

In connection with these reviews, the committee conducts inspections of the reactor. During these inspections, a special point is made to observe reactor startup and shutdown procedures and to scrutinize the log book and other procedural material. At the time of the formal review, the committee may question the operating group concerning any of the items observed during the inspection, contained in the report, or otherwise brought to their attention.

As a result of this review, specific recommendations are made to the Laboratory management by the committee concerning continued operation of the reactor.

In addition to this review function, concurrence of the Reactor Operations Review Committee is required before any changes are made in reactor operation which may have a significant effect on safety. This

committee will review the initial installation and operating plans for the TSF-SNAP reactor.

The Criticality Review Committee has jurisdiction over operations which involve the handling, storage, and transportation of significant quantities of fissionable material. Reactor fuel within a reactor core is specifically exempted from this; however, procedures for handling of fuel before insertion and after removal must be approved by this committee. The committee acts in many respects as a consulting group and gives assistance in problems involving criticality. It also conducts an annual review of each facility to ensure that approved procedures are being followed.

Finally, the Laboratory has established, as a staff function of the Director's office, a Safety and Radiation Control Department. This organization establishes, on behalf of Laboratory management, policy with respect to radiation protection and ascertains that this policy is met at all times. It promulgates criteria, for example, for facility containment, and serves a liaison function between the various Laboratory divisions. Staff members of the Safety and Radiation Control Department are assigned responsibilities for following closely the activities of those Laboratory divisions which handle significant quantities of radioactive materials. Specialists in key elements of the radiation safety program, such as containment, waste disposal, criticality, reactor safety, etc., are on the staff of the Director of Safety and Radiation Control.

One of these specialists was assigned to follow the progress of the TSF-SNAP procurement program and of the design of the installation, and has been informed of all design and safety questions that have arisen on the project.

3. Operating Procedures

The reactor will be operated in conformance with operating procedures approved by the TSF Supervisor.

During normal operation a shift supervisor and one other responsible person must be present when the reactor is operated; either a supervisor,

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an engineer, or a technician must be present in the control room during constant power operation; and a supervisor or an engineer will observe the operation condition of the reactor and record instrument readings at least once an hour during operation.

SNAP reactors are designed to operate for the expected core life in space environment without maintenance. The TSF-SNAP reactor has been designed to duplicate this performance for ground level operation. During the first two phases of the SNAP experiment it will be possible to do some maintenance on the reactor. However, it will be required that two shift supervisors review what maintenance is required and to prepare a detailed list of the steps of performing the maintenance. To perform the maintenance the TSF supervisor must approve the procedure and at least three persons, including two shift supervisors, must be present during the maintenance.

At the TSF an overall data plan is written for each experimental program or each major portion of the program. This plan outlines the objectives and specifies the shield configurations, the number of reactor operations, and the data that are to be taken. To specify which items of the overall plan will be done on a given day, a daily data plan is written. The daily data plan will deviate from the overall plan as the information acquired in earlier experiments dictates. All the above information is filed in a binder.

A somewhat unusual situation will exist during assembly and first criticality of the TSF-SNAP. As described in Section VI of this report, a welding operation and inspection will be required after the first cold, dry critical experiment. Every day's activities during this period will be reviewed by the TSF staff and the list of operations planned for the following day must be approved by the TSF Supervisor or a designated alternate. Adequate emergency procedures will be formulated for conceivable accident situations, and all personnel will be acquainted with them before each day's program is undertaken.

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VI. OPERATION OF THE REACTOR AND FACILITY

1. Deactivation of the TSR-II

During the period of assembly of the SNAP reactor and during periods of its operation, the TSR-II will be placed on the concrete pad to the north of the handling pool (the detector chamber of the TSF-SNAP installation). The reactor will be in the 12-ft diam beam shield which will attenuate gamma rays from the core.

The water hoses and electrical cables as well as the hoisting cables will remain in place and will be suspended from the towers. Since the TSF-SNAP experiments do not require the towers, no mechanical interference will occur between the SNAP reactor and the TSR-II. Therefore, the TSR-II can remain ready for operation. It is anticipated, however, that a transfer of some control components, such as magnet amplifiers, will be required between the two systems, thus effectively disabling the system from which the component is removed.

2. Reactor Shipment

The reactor will be disassembled into components and subassemblies at Atomics International and will be shipped in parts from Canoga Park to Oak Ridge. Shipment will be by commercial carrier.

The major parts will be:

1. The vessel assembly
2. The vessel head assembly
3. The structural support tube and control wiring
4. The internal reflectors
5. The grid plates (including hold-down springs)
6. The fuel
7. The external reflector assembly
8. The NaK pump

Upon arrival at the Tower Shielding Facility the parts will be unpacked and inspected. The degree of inspection to be performed on the parts will be determined by ORNL Inspection Engineering requirements and through an examination of the final inspection reports of Atomics

International. Ultrasonic examination of major metal parts, die penetrant and x-ray photography of welds, and helium leak checks of all NaK containing subassemblies will be used to establish their soundness and adequate wall thicknesses.

The heat exchanger parts will be welded together into an assembly at ORNL and all fabrication is under the continuous surveillance of the ORNL Inspection Department.

3. Reactor Assembly and Installation

All assembly of the reactor core and of the core-reflector-heat exchanger system will be done at the Tower Shielding Facility, within the shielded maintenance area. There will be two distinct major stages of assembly: (1) assembly of the core and reflector for a cold, dry critical experiment; and (2) assembly and sealing of the NaK containing components.

Detailed procedures for the loading of the fuel rods and for progressively checking multiplication as the system approaches criticality will be formulated, based largely on the experience accumulated at Atomic International with similar systems.

Several steps in the assembly procedure will require welding operations and mechanical work to be performed on and around the vessel with all the fuel in place. Each step has been reviewed in detail and adequate safeguards against nuclear or mechanical hazards are assured. A "rehearsal" of the assembly operation will be performed on site before fuel is handled to assure adequate working space and personnel safeguards for the people who will do the work.

The principal safety measures that will be implemented will include the use of jigs and fixtures that will hold the vessel rigidly on a stable work platform and will make possible precise mechanical assembly of the reflector halves to the vessel, and of a "poison" sleeve that fits around the vessel in place of the reflector and holds the system subcritical in any circumstance.

The first stage of the assembly operation will be mechanical fit-up of the reflector to the vessel utilizing the reflector handling machine

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that has been developed by Atomics International. Dimensional checks of reflector to vessel clearances will be made as well as checks of the alignment of the reactor rotator to the vessel and the routing of control wiring and piping to the reflector. Once mechanical compatibility of all parts is established and control system operation is verified, loading of fuel can proceed.

The dry critical experiment will include determination of the fuel arrangement that will produce the desired loaded excess multiplication in the system. All control of the reflector will be effected from the underground control room and all personnel will retire to the control room before any control drums are moved.

Following the dry critical experiments the vessel head will be welded to the vessel and the heat exchanger assembly, including the NaK pump, will be welded to the vessel and head. During this welding operation a specially designed poison sleeve will be in place around the vessel.

After this welding is completed, the system may be charged with NaK. This will be done with a NaK charging "cart" built by Atomics International. This is a complex of equipment that includes a NaK reservoir, pumps, heater, inert gas blanket system, cold traps and an oxide content test unit, as well as associated controls and recorders. Following completion of all work with the NaK cart the two nipples that joint the system to the NaK cart will be pinched off and welded shut. The NaK system will then be sealed and no longer accessible.

The final step in the assembly of the reactor will be the installation of the reflector on the vessel. This will be done, as before, by means of the reflector handling machine and with void filler blocks in place behind the two coarse drums. Void filler blocks are poison assemblies that fit behind withdrawn control drums and physically prevent rotation of the drums. The final manual operation required, will be removal of the void filler blocks as was done in the FS-4 system prior to its launching into orbit around the earth.

The reactor will now be in its fully operational configuration and all fixtures and platforms used in assembly can be removed.

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4. Operating Plans

The operation of the reactor at the Tower Shielding Facility will proceed in roughly four stages. The first stage will include assembly of the reactor and achievement of criticality at elevated temperature, as described in topic 3 of this section. The second state will include low power operation, not exceeding 10 watts, for core leakage radiation mapping purposes. This mapping will be carried out entirely within the shielded maintenance area as shown in Fig. 29. The third stage, though not yet firmly defined, will include operation at powers of up to 1 kw and may be directed toward study of secondary gamma-ray production in various material combinations. In this stage the reactor will be operated outside the sheltered area.

The first three stages of reactor operation, then, will not be at elevated temperature and will not produce a large inventory of fission products. The philosophy of operation will be to operate at the lowest power levels consistent with production of useful experimental data. This plan of operation will minimize the hazards incurred while experience in operating the reactor is accumulated. It is probable that the first three stages of operation will consume approximately one calendar year.

The fourth stage of operation will include shield transmission experiments, as pictured in Fig. 7, at power levels in the range of the reactor's capacity. Once this stage is entered the reactor will become an increasingly dangerous source of shutdown gamma rays and will begin to build up a significant fission product inventory. The total energy evolution that is foreseen for the entire operating life of the reactor at this time is 12,000 kw-hrs.

5. Power Calibration

A power calibration of the reactor will be required some time toward the close of the third stage of operation. Until that time the reactor will be controlled through the leakage flux of neutrons from the core that will be related to actual fission rate through a calculated conversion factor.

Fig. 29. Mapping of Leakage Radiation from Core.

Two methods for measuring energy generation rate versus neutron leakage flux are proposed. One method would utilize the weather shield with its heaters and would require some insulation of the weather shield from the surrounding air. By first feeding electrical energy to the heaters at the same rate, W kw, until the temperatures at various points of the system have stabilized, then replacing the electrical energy input or some part of it, ΔW , by reactor heat generation so that the same temperatures are reached in the system, the reactor power can be equated to ΔW . By doing this at several levels a curve of reactor thermal power versus neutron flux reading on the compensated ionization chamber channel can be generated.

One possible source of error in this method is the difference between temperature distributions in the system for the two cases of energy input. This difference can doubtless be minimized by insulation of the exterior of the weather shield. The effect of insulation on the reactivity of the system and on the leakage flux will be checked by critical experiment.

An indicator of thermal power is, of course, the ΔT across the core for a known rate of flow of the coolant. The accuracy of flow indication derivable from the electromagnetic NaK pump is at present not thought to be better than 20 percent. Should calibration of the pump and subsequent experience with it indicate a higher degree of accuracy, the mass flow rate multiplied by the temperature rise and the heat capacity would yield an additional check on the power calibration.

6. Final Shutdown

After all required experiments have been performed the reactor will be shut down and stored for a period of about two years in the concrete-lined pit that is in existence, 53 ft to the south of the detector chamber. Figure 30 shows the estimated gamma dose rate at a distance of 10 ft from the reactor, as a function of time after shutdown. This estimate was made on the basis of the Phoebe code calculations of fission product buildup and on the assumption that gamma-ray leakage out of the core is 25% of the total gamma energy. The energy-to-dose conversion factor used was 6×10^5 MeV/sec $\text{cm}^2/\text{r/hr}$.

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Fig. 30. Dose Rate at 10-ft Distance from Shutdown Reactor.

The reactor will be placed into a perforated stainless steel drum and the drum will be lowered into the silo where it will rest on the grating, about 20 ft below the surface of the concrete pad. A cover plug will be placed over the mouth of the silo so that direct precipitation will be excluded.

It is anticipated that at the end of two years it will be possible to move the reactor out of the silo and remove it to a hot cell where complete disassembly will be effected. During the two-year storage period the drain pipe from the silo will be monitored for effluent radioactivity.

Figure 17 shows the location and a cross section of the silo to be used for storage of the reactor.

VII. HAZARDS AND SAFEGUARDS

The operation of the TSF-SNAP reactor will be divided into four stages. The first three stages will be at low power, for core mapping, secondary gamma and other experiments whose maximum thermal power requirements on the reactor will not exceed 1 kw for any extensive periods of time. The last stage will require the higher power levels in the capability range of this reactor, with its attendant higher temperatures of 700-800°.

Although the SLOFS-3 reactor has been operated continuously for some 10,000 hrs in a vacuum chamber, at a NaK outlet temperature of over 900°F^{16,17} and one reactor has operated in space,¹⁷ there has been no experience of operation in air, with repeated startups, as will be the case of the TSF. However, before the TSF-SNAP reactor is taken to powers at which appreciable fission product buildups are achieved the reactor will have been operated for approximately 1 year at low powers but including some operation at elevated temperatures.

The maximum release of fission products from a SNAP-2/10A type core is at present somewhat in doubt. Two destructive tests on essentially identical cores but with different test setups have been conducted by Atomics International and Phillips Petroleum Company at NRTS, Idaho.^{18,24}

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These tests, the SNAPTRAN-2 and -3 experiments, represent extremes of consequences of the most severe type of accident in connection with the TSF-SNAP system. Both tests confirmed that hydrogen evolution from the fuel is the controlling destructive mechanism and that melting of the fuel does not occur. The SNAPTRAN-3 test was conducted on April 1, 1964, while SNAPTRAN-2 was conducted on January 12, 1966. The SNAPTRAN-3 experiment evolved about 45 Mw-sec while the SNAPTRAN-2 evolved approximately 54 Mw-sec. In this regard the two tests were much more severe than any credible destructive accident to the TSF-SNAP reactor with its far lower available reactivity and its calculated maximum evolved energy of 28 Mw-sec (see ref. 1) in the maximum credible accident.

However, in evaluating the potential hazard to the public from a maximum credible accident a reasonably conservative approach is to assume the worst possible release of fission products based on experiment, and a maximum conceivable fission product inventory. This has been done and is described below and the results, based on the best current information, indicate no unacceptable hazard to the public.

The maximum release of fission products that is credible is the release apparently achieved in the SNAPTRAN-2 tests in which a step insertion of \$4.63 at an initial level of about 1.5 mw produced destruction of the core through rapid hydrogen evolution before fuel melting temperatures were reached.¹⁸ The maximum power level apparently achieved before reversal of the transient was 71×10^3 Mw.

The SNAPTRAN-3 test was an earlier destructive test in which \$3.75 of reactivity, above critical condition, was stepped into a system, but with the total step beginning with the reactor in a subcritical condition. The core was in this case also disassembled through the rapid evolution of hydrogen from the U-Zr-hydride fuel.⁵

The SNAPTRAN-2 core was dry; that is, without NaK. The vessel was open at both ends so that the fuel cans were in air. The core was surrounded by a beryllium reflector, very similar to the reflector that will be used in the TSF-SNAP reactor, and the step insertion of reactivity was introduced by rapid inward rotation of two beryllium control drums. This type of event is postulated as the MCA-producing action in the TSF-SNAP.

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1. Accidents and Their Consequences

Calculations of the consequences of various accidents involving nuclear excursions have been performed at Atomics International and are reported in ref. 1. The maximum credible nuclear accident is judged to be a case in which procedures are disregarded and all interlocks fail, allowing the most unfavorable mode of insertion of all control drums. No power level scram is allowed to occur, and the fuel cans as well as the reactor vessel are ruptured by the hydrogen liberated from the zirconium hydride. The accident postulated as the MCA for this system is akin to the SNAPTRAN-2 destructive test, although the total energy evolved is about half of that evolved in the destructive test. This is because of the much larger amount of positive reactivity available in SNAPTRAN-2, about \$5. as compared to \$1.80 in the TSF; and because of the very rapid insertion of this excess multiplication into the system, about \$370/sec as compared to about \$2/sec in the TSF credible MCA.¹

When a scram is allowed to occur at 20 kw, the maximum fuel temperature remains far below the fuel can burst temperature (800°F as compared to 1450°F) and the system is not damaged. It should be noted that level scram in the actual installation will be set at 15 kw.

Another class of accident whose consequences should be considered is mechanical damage to the reactor and the reactor suspension system. The most extreme consequences of such an accident is dropping the reactor onto the concrete pad or on the ground. The most severe results that can be envisioned would not exceed the severity of the MCA. If the reactor becomes detached from the support structure the air and electrical supply to the coarse drums would be interrupted, immediately scrambling the two drums. The fine-drum actuators, upon interruption of electrical supply, will be locked in place by the brakes in the actuators. It is impossible for both coarse drums to be driven all the way in simply by action of the reactor falling over on one side. This is evident from an examination of the structural details of the reflector assembly. However, it is conceivable that collapse of a structure, such as the shielding wall, around the reactor might produce a set of circumstances through which the drums

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were all forced into the reflector. The results, in the nuclear sense, would be no more than the MCA.

It should be noted that in both the SNAP-10A and the TSF-SNAP reactors, the reflector halves are held around the vessel by a spring loaded metal band (visible in the views of Fig. 2). If this band parted, the reflector halves in the SNAP-10A reactor were free to rotate, about hinge pins, away from the core. In the SNAP-10A this rotation was intended to be a final shutdown mechanism and an explosive device was provided to sever the band on electrical command signal. In the TSF-SNAP reactor a clevis pin is used to link the reflector retaining band since there is no intention of using reflector rotation as a shutdown mechanism.

Another set of circumstances that is unlikely but conceivable would create a condition in which the reactor would continue to operate without any normal means for shutting it down. Such a situation was considered by Atomics International³ in considering the consequences of re-entry and landing of an intact SNAP-10A core after a space flight. Such a situation was found to be possible if the reactor core landed in a crater in the earth, which then began to fill up with water. It was found that a steady-state operation could be achieved with water boiling off from reactor heat as rapidly as new water was added. It was found that earth alone around the core would not produce criticality and that when the ratio of core radius to the crater radius exceeded 0.7 not even complete filling of the crater with water would produce criticality. Thus a large pool of water is required for critical water reflector. This also means that reflection from a thin film of water over the reactor is not critical. Thus rain is not a reactivity hazard. Total immersion of the reactor in water is not considered credible at the TSF.

The following sections examine the most extreme consequences of fission product release, NaK release and NaK fire, to the public. Obviously the staff at the TSF will be much more vulnerable to the consequences of accidents, particularly if an accident should occur when the reactor is shut down and the staff is in the open in the reactor area. However, the combination of interlocks and strictly enforced procedures should reduce this possibility to a minimum.

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In case of an MCA there would be some dispersion of pieces of fuel and of reactor structure due to the force of the explosion. This should not exceed the dispersion observed in the SNAPTRAN-2 test. In this test the vast majority of pieces of the reactor were found within a radius of 100 meters from the reactor while the absolute outer radius of identification of particles was 200 meters. This is well within the TSF inner fence, which is an 8-ft high, security type fence topped with three strands of barbed wire.

2. Fission Product Inventory and NaK Activity

"Phoebe" code calculations were performed to evaluate the fission products built up in the core after various periods of operation. The maximum gross fission products buildup was achieved in the case equivalent to operation of 12,000 kw/hr with a final operating period of 8 hr to 30 kw.

The gamma activity immediately following this operation is 1.392×10^5 curies and the gross beta activity is $1.2.2 \times 10^5$ curies, and the total weight of fission products is 0.5284 grams. The average gamma energy is 1.074 MeV/photon and the average beta energy is 1.475 MeV/particle.

The inventory of tellurium and iodine in the reactor at the end of the last operating period of a sequence of 300 cycles of 8 hr at 5 kw followed by 16-hr shutdown was calculated by the method outlined in the Operation BREN Hazard's Report,²³ in which the buildup of each nuclide is obtained as the sum of a finite geometric series:

$$\sum ax^n = \frac{a(1 - x^{n+1})}{1 - x},$$

where

$$\begin{aligned} a &= \text{activity at end of 8-hr operation} \\ &= A_s(1 - e^{-\lambda 8}), \end{aligned}$$

A_s = saturated activity at 5 kw

$$= \left[\frac{3.2 \times 10^{13} \times 5 \text{ fissions}}{3.7 \times 10^{10} \text{ second}} \times Y \right] \text{ curies,}$$

Y = fission yield of a nuclide,

λ = decay constant of nuclide,

$$x = e^{-\lambda t},$$

n = 300 cycles.

The buildups thus obtained are the following:

<u>Nuclide</u>	<u>Half Life</u>	<u>% Yield</u>	<u>Saturation Curies</u>	<u>Accumulated Curies</u>	<u>Equivalent ¹³¹I, Curies</u>
¹³¹ I	8.05 days	2.9	125	40	40
¹³² Te	77 hr	4.4	190	67	
¹³² I	2.4 hr	4.4	190	63	2.3
¹³³ I	20.8 hr	6.5	281	121	32.5
¹³⁴ Te	44 min	6.7	290	290	
¹³⁴ I	52.5 min	7.6	328	328	5.6
¹³⁵ I	6.68 hr	5.9	256	157	13.0
TOTAL ¹³¹ I EQUIVALENT					93.4
RELEASED ¹³¹ I					65.4

The buildups of the significant noble gases are the following:

<u>Nuclide</u>	<u>Half Life</u>	<u>% Yield</u>	<u>Saturation Curies</u>	<u>Accumulated Curies</u>
Kr ^{85m}	4.36 hr	1.5	65.	47.7
Kr ⁸⁷	78 min	2.7	117.	115.
Kr ⁸⁸	2.77 hr	3.7	160.	134.
Xe ^{133m}	2.3 days	.16	7.	2.6
Xe ¹³³	5.27 days	6.5	281.	97.5
Xe ^{135m}	15.6 m	1.8	78.	<u>78.</u>
TOTAL				622

The coolant in the TSR-SNAP reactor will circulate through the core at a rate of 4 gpm and will become activated principally through the reactions

The activation was estimated using the equation

$$\Delta_k(t') = NF\sigma(t'/T) [1 - e^{-(k+1)\lambda T}]$$

as given in Rockwell¹² where

F = average core thermal flux
= 1.12×10^{11} n/cm² sec at 10 kw,

σ = neutron capture cross section
= 0.505×10^{-24} cm² for ²⁴Na_a
= 1.30×10^{-24} cm² for ⁴²K,

t'/T = ratio of coolant cycle time in core
= 0.1036,

N = number of atoms per cc
 = $4.8 \times 10^{21} \text{ cc}^{-1}$ for ^{23}Na
 = $6.1 \times 10^{20} \text{ cc}^{-1}$ for ^{41}K ,

K = number of cycles of coolant, completed,

λ = decay constant, $1.28 \times 10^{-5} \text{ sec}^{-1}$ for ^{42}K ,

T = coolant cycle period, 42.3 sec.

Figure 31 shows the activity present in the entire system after various times of continuous operation at 20 kw. The total of core plus plenums was estimated on the basis of data obtained by Atomics International and represents activation of NaK outside the core region, in the upper and lower plenums of the reactor vessel. This activity represents 7.6% of the total activity in the system.¹³ This activity is a very conservative estimate of the amount that might be present at any given time in the reactor. A check of the buildup that would result from 300 cycles of 8-hr per day operation at 5 kw yielded a value of only about 1 curie of ^{24}Na .

3. Hazards from Dispersal of Radioactivity

Release of Fission Products. The maximum release of fission products that is credible is the release achieved in the SNAPTRAN-2 test, which was the following:¹⁸

Iodine released	70%
Noble gases released:	75%
Tellurium released:	45%
Solids released:	5%
Total fission products released:	20%

In the SNAPTRAN-3 experiment,²⁴ less than 1% of the total fission products and under 5% of the noble gas products were released.

In laboratory tests, releases due to high temperatures were considerably lower than releases due to a destructive explosion of the core. Irradiated fuel samples were held at high temperatures for long periods of time.²⁵ For our case the most valid comparison is with the 2000°F data, the temperature that was held for six and eight hours in the tests of

Fig. 31. NaK Activity as a Function of Time at 20 kw.

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ref. 25. In the MCA this temperature would be approached for only fractions of a second.

For iodine-131 a release fraction of 0.002 was found; for Xe-133, 0.02; for Sr-89, 0.00002; for Cs-137, Ce-144, 0.00004, Ru-103, Ru-106, 0.000001 and so on.

The release numbers adopted on the basis of SNAPTRAN-2 data, therefore, appear to be the highest that may be encountered in an accident.

Dose to Thyroids. The dose that may be delivered to the thyroid glands through inhalation of iodine from a cloud that would be released in an accident was estimated by the method used by Burnett in ref. 23:

$$\text{Dose (rem)} = D_{\text{I-131}} I Q \left[\frac{\text{TID}}{Q} \right] \times 10^6$$

where

$D_{\text{I-131}}$ = specific iodine dose
= 1.484 rem/μcurie inhaled

I = inhalation rate
= 5×10^{-4} m³/sec (30 liter/min),

Q = I-131 equivalent released
= 65.4 curies,

$\left[\frac{\text{TID}}{Q} \right]$ = cloud density-duration at 1,000 meter distance
= 4.8×10^{-4} (curies-sec/m³ curie released).

The cloud density was the same as in ref. 26, and represents a highest value for the TSF site 1,000 meter exclusion fence, from a ground source assumed to be an instantaneous point source.

The dose thus delivered to the thyroids is 23.3 rem. This is far below the 300 rem acceptable as a public hazard.

Gross External Exposure. An estimate of the gross external exposure from total fission products released that may be inflicted at the 1,000-meter exclusion fence was made on the basis of the expression given in ref. 23:

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$$D \text{ (rem)} = 0.26 E_s \left[\frac{TID}{Q} \right] \times Q ,$$

$$\left[\frac{TID}{Q} \right] = \text{cloud density-duration at 1,000-meter distance,} \\ = 4.8 \times 10^{-4} \text{ (curie-sec/m}^3 \text{ released),}$$

$$.26 = \text{conversion constant (m}^3 \text{ rem/curie-sec),}$$

$$E_s = \text{effective energy for submersion, 1.5 MeV/disintegration,}$$

$$Q = \text{curies released}$$

$$= 20\% \text{ of beta curies plus } 20\% \text{ of gamma curies}$$

$$= 5.2 \times 10^4 \text{ curies,}$$

$$D = 9.7 \text{ rem.}$$

Such a release of fission products in a volatile cloud is considered to be impossible. The work of ref. 25 demonstrated that due to temperature effects alone, the SNAP fuel retains the fission products.²⁵

External Exposure from Noble Gases. Many previous studies of fission product release and the resulting hazards have shown that the inhalation of iodine isotopes results in the governing internal dose, to the thyroids, while the external dose is largely the result of exposure to a cloud of the noble gases, krypton and xenon.

Using the previous expression for dose at 1,000 meters resulting from instantaneous release of a cloud of fission products and $E_s = 1.61 \text{ MeV}$ for the effective decay energy from the total of 622 curies of noble gases²⁷ we obtain

$$D = 0.26 E_s \left[\frac{TID}{Q} \right] Q ,$$

where

$$D = 0.26 (1.61)(4.8 \times 10^{-4}) [622 \times 0.75] \\ = 152 \text{ mrem.}$$

It is evident from this result that external exposure to radiation is a small hazard to the public even if 100% of the volatiles were to be released.

External Dose from Fission Products and NaK Cloud. To estimate the whole-body radiation exposure that might be inflicted on a person at the 1,000-meter exclusion boundary of the TSF in the event of a nuclear accident, $D = .26 \times E_s \bar{Q}t$ rem was used, where

D = whole body dose in rem,

E_s = effective energy of decay radiation

= $(E_\gamma + 1/3 E_\beta)$, MeV, 4.59 MeV for ^{24}Na , 3.37 MeV for ^{42}K ,

$\bar{Q}t$ = cloud density times period of submersion

= $\left[\frac{\text{TID}}{Q} \right] \times Q$ (curies/ m^3) x sec,

Q = curies released instantaneously, 4 curies ^{24}Na , 1.2 curies ^{42}K ,

$$\left[\frac{\text{TID}}{Q} \right] = 4.8 \times 10^{-4} \text{ sec}/\text{m}^3 \text{ (ref. 26)}$$

The value of $\left[\frac{\text{TID}}{Q} \right]$ for TSF conditions was calculated independently

of ref. 26, using data from AECU-3066, "Meteorology and Atomic Energy" and a value of 1.48×10^{-4} was obtained assuming stable metrological conditions. The more conservative value of ref. 26 was used in estimating the dose.

For ^{24}Na the dose is 2.3 mrem.

For ^{42}K the dose is 0.5 mrem.

For fission product release, the dose is 152 mrem.

Total whole body dose - 155 mrem.

4. Chemical Hazards

Any rupture of the NaK containment system will result in loss of part or all of the coolant and would probably mark the end of operation of the reactor. The hazard to operating personnel will be identical to that experienced with sodium and NaK-cooled systems, but this hazard is limited to chemical effects. No damage to the fuel is credible as a consequence of NaK spillage.

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Exposure of NaK to air leads to burning of the sodium and potassium according to the following reactions.^{28,29}

Other reactions with oxygen are possible, but in our case with ample air available consideration of only the complete oxidation reactions is thought to be pertinent.

The possible reactions with water are:

Since the reaction with water goes at room temperature, it is clear that oxidation will occur more quickly in moist air and that the presence of water constitutes a hazard in case of a rupture of the system.

The pool temperature of a small volume of burning sodium is 1,630°F, the temperature of volatilization of the metal. For NaK the temperature is 1,444°F. This is the maximum temperature achievable in a system open to ambient air³⁰ and is far below the 2,550°F melting point of a eutectic of uranium-zirconium alloy and Hastelloy N.²⁵ Thus no melting of the fuel can take place as a result of exposure of NaK to air.

There are many other reactions with various elements, possible for NaK, some of which proceed at an explosive rate. As a result, care will be taken to eliminate from the vicinity any such materials. In particular, the only fire-fighting materials to be allowed in the area are calcium carbonate, graphite powder, and a commercial fire extinguisher known as MET-L-X. In general, however, no attempt will be made to fight a fire that develops during operation of the reactor and operating personnel will be instructed to remain in the shielded control room area if a fire should develop.

Inhalation of sodium and potassium vapors or of the fumes from a NaK fire can cause damage to the eyes, nose, throat, and lungs. Operating personnel should therefore take cover if not wearing protective clothing and breathing apparatus.

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The hazard of burning NaK to people at the 4,085-ft exclusion fence is somewhat difficult to define because only one of the possible constituents of fumes from a fire has a well-defined recommended allowable limit in air. This is the NaOH, for which a limit of 2 mg/m^3 has been established by the American Conference of Government Industrial Hygienists.³¹ All other oxides and hydroxides of sodium and potassium are referred to the NaOH data.

It is not credible that all the sodium and potassium can react with air and moisture so as to create an instantaneous point source of all the NaK present in the TSF-SNAP system. Even a violent rupture of the containment would spill the NaK over a finite time and reaction with air would then proceed at some finite rate, even though at elevated temperature this rate may be quite high.

In tests conducted at CANEL by Pratt and Whitney Aircraft,³² NaK at a temperature of $1,310^\circ\text{F}$ was discharged to the air and temperatures of the metal floor on which the NaK was spilled were measured. The temperature did not rise above $1,300^\circ\text{F}$ and it is estimated that 45 to 60 percent of the NaK burned. The rest ran into a sump and did not burn because of the self-extinguishing characteristics of a NaK fire when the NaK gathers into some stable volume. The formation of oxides on the surface of such a volume excludes further air from the surface of unreacted NaK.

In these tests, then, not more than 60 percent of 5.5 gallons burned in about 14 min. The total duration of burning was actually 18 min and the duration of "vigorous" burning was 14 min. To be conservative, we shall take the shorter duration. The rate of burning then under these conditions is about 0.24 gallons/min and at a NaK density of 0.74 g/cm^3 , the mass rate is 672 gallons/min . This will be the source rate for fumes, and will be equal to creation of about $1,012 \text{ gallons/min}$ of NaOH and KOH.

This rate of cloud formation will then be diluted as it moves toward the exclusion fence. The degree of dilution may be estimated by the equation given in AECU-3066³³ for a continuous point source:

$$\bar{Q} = \frac{2Q}{\pi C^2 \bar{\mu} x^{2-n}} e^{-(y^2+h^2/C^2 x^{2-n})}$$

where

$\bar{Q}$ = downwind concentration, grams per cubic meter,

Q = source strength, grams/sec,

C = generalized diffusion coefficient, meters^{n/2},

n = nondimensional parameter associated with stability,

$\bar{\mu}$ = mean wind speed, meters per second,

x = downwind coordinate measured from a ground point beneath a continuous source.

Assuming conservative values for the parameters, $n = 0.50$, a condition for a violent inversion $C = 0.3$, a condition for a stable atmosphere $\bar{\mu} = 1.5$ meters/sec, a representative value of wind velocity and assuming that the source and receiver are both at ground level ($h = 0$, $y = 0$), the dilution factor is 1.8×10^{-5} and the density of cloud delivered at the fence, 1,247 meters away, is $18.2 \text{ mg/m}^3 \text{ sec}$. Although this is a factor of nine above the ACGIH recommended limits for occupational tolerance, it is still a very low concentration from which no permanent effects would accrue. When the factor for the height of the tower above the exclusion fence is included, i.e., $h = 100$ meters, the concentration at the fence drops to $1.42 \text{ mg/m}^3 \text{ sec}$.

At a rate of burning of 672 grams/min, it would take 8,371 grams in the system 12.5 min to burn completely. Assuming that a person at the exclusion fence during this period breathes at the rapid rate of 30 liters/min, a total of 375 liters of NaOH-containing air will be inhaled. This represents an inhalation of 6.8 mg of the products of the burning NaK, assuming the higher concentration of 18.2 mg/m^3 is maintained steadily for 12.5 min in the cloud. Since the recommended limit of 2 mg/m^3 would result in a total inhalation of about 30 mg in an 8-hr working day, it must be assumed that 6.8 mg do not constitute a hazard.

It is obvious that for operating personnel the hazards of a NaK fire are quite severe and protective equipment as well as administrative controls will be required to obviate them.

5. Leakage Radiation from TSF-SNAP Reactor

The Tower Shielding Facility is situated within a general exclusion area which is enclosed by a 6-ft-high chain-link fence topped with three strands of barbed wire (see Fig. 32). The closest approach to the reactor along this fence is at a radius of 3,800 ft from the reactor along an arc which is marked M2, M3, and M4 in Fig. 32, but the terrain is so steep and heavily wooded that this area is not considered accessible to the general public. The nearest accessible approach to the reactor (see M1 in Fig. 32) is at a distance of 4085 ft where TVA lands of Melton Hill Lake are adjacent to the exclusion fence and the fence is reasonably accessible for pedestrians from a road that runs along the lake. All other points along the exclusion fence are at a greater distance from the reactor location and the topography surrounding the TSF (see Fig. 33) is such that at no point along the exclusion boundary is a line of sight from the fence to the SNAP reactor possible.

Four gamma-ray sensitive detectors are located at points M1, M2, M3, and M4 along the exclusion fence. Calculations³⁶ have shown that monitors located at the indicated points will give adequate information to insure that at no place along the exclusion fence will a higher dose be received than that which is indicated at the monitor stations even for a highly collimated beam of radiation from a reactor such as the TSR-II. The gamma-ray dose rate at the four monitor stations is continuously recorded on a strip chart in the control room. The neutron to gamma-ray ratio will be measured for the TSF-SNAP reactor as it was for the TSR-II,³⁵ and this ratio will be used with the monitor readings to calculate the daily dose accumulated at the exclusion fence. The daily dose is recorded in the operations log book and total dose at the fence is published in operating reports that are now issued semiannually.³⁷ The allowable dose in an area which is accessible to the general public is 500 mr/yr. Operation of the TSF-SNAP installation will be governed by the dose

Fig. 32. Area Topography, Tower Shielding Facility, ORNL.

Fig. 33. Topographic Map of Tower Shielding Facility

calculated from the measurements at the point of closest access to the reactor, the 4085 ft point.

Some estimates of the dose rates to be expected at the fence are possible both on the basis of calculations and of measurements made for the Health Physics Research Reactor (HPRR) and the TSR-II. It is estimated that about 40% of the fission neutrons leak out of a SNAP 10A core⁸ and this includes both fast and thermal neutrons. At 10 kw this means a source strength of 3.2×10^{14} n/sec. Assuming isotropic leakage, this produces a current of 2.74×10^{10} n/cm² sec at a radius of 1 foot from the reactor. Calculations by Straker for neutron transport in an infinite air medium¹⁹ indicate a neutron dose at a distance of 4085 ft of 0.18 mrad/hr for a reactor power of 10 kw. Later calculations that included the effects of the ground, showed that the dose per source neutron at the 4085-ft distance is reduced by a factor of 1.95 over the infinite air case when the source is 91.4 m above the ground and the detector is 1 m above ground. In absolute terms, this calculation indicated a dose rate of $5.7 \times 10^{-8} (4\pi R^2) \text{ rad cm}^2 (\text{n/sec})^{-1} \text{ hr}^{-1}$, where n/sec refers to source leakage neutrons. This yields for our case a dose rate of .093 mrad/hr at 10 kw. If an RBE of 10 is assumed for all neutrons, the biological neutron dose rate is .93 mrem/hr.

For comparison with measured doses, the experience of the HPRR is probably the best indication of what is to be expected from the TSF-SNAP, since at ORNL, both reactors are located near the ground, and in similar terrain within the same exclusion area. The HPRR is also a high leakage (about 50% of fission neutrons), compact reactor. Measurements of dose rates made in the HPRR are as follows:²¹

At 3,300 ft from the reactor to the north:

Gamma ray dose	0.02 mrem/hr kw
Thermal neutron dose	0.01 mrem/hr kw
Fast neutron dose	<u>0.05</u> mrem/hr kw
	0.08 mrem/hr kw

Using Straker's relaxation length of ~ 180 m, the dose rate at a distance of 4085 ft would be about .02 mrem/hr kw.

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To obtain an indication of the effects of terrain, it is possible to compare these numbers with those obtained during a test of the HPRR in flat terrain in Nevada.²² At a distance of 4100 ft, with the reactor at an elevation of 27 ft and the detector at ground level, the specific dose rates were:

neutrons	$3.1 \times 10^5 \text{ R}^2 \text{ mrad ft}^2 \text{ hr}^{-1} \text{ kw}^{-1}$
gamma rays	$4.1 \times 10^5 \text{ R}^2 \text{ mrad ft}^2 \text{ hr}^{-1} \text{ kw}^{-1}$

At 10 kw and a distance of 4085 ft this yields dose rates of:

0.187 mrad/hr neutron dose
0.250 mrad/hr gamma ray dose

If RBE is still assumed as 10, the total dose rate is 2.12 mrem/hr, as compared to 0.8 mrem/hr at 3300 ft measured at ORNL. The measurement at Nevada was made with a direct line of sight from the detector to the reactor. On an equivalent distance basis, this indicates that the terrain and slightly higher air density at ORNL reduces the dose rate by a factor of about 10.

Thus, assuming 250 operating days per year, the limitation of 500 mr/yr delivered at the exclusion fence would limit the daily dose to 2 mr/day which would limit operation at 10 kw to 2 1/2 hr/day if the dose rates measured at 3300 ft from the HPRR are applied directly, and to some 10 hrs per day if the dose rate is reduced to the 4085 ft distance.

Monitors are located near each exit inside the control rooms of the TSF as well as at the exclusion fence. The reactor is automatically interrupted if the radiation level exceeds tolerance levels at any two entrances. Mazes of concrete blocks have been erected to keep the radiation level entering the passageway below maximum acceptable levels. These mazes may be increased if necessary but it is not anticipated that they will be, since the TSR-II has been operated bare at 100 kw suspended closer to the control building than the location of the SNAP reactor. Furthermore, radiation could not be measured through the concrete and earth shield above the control rooms during any previous reactor operation at the TSF.

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6. Integral Dose to Structure and Components

Radiation damage to the TSF-SNAP reactor should not be a factor during the anticipated operational life of the system. Estimates of the integral fluxes to be delivered to the vessel walls during the anticipated operating life of the reactor, conservatively taken at 12,000 kw hrs indicates 2.8×10^{17} nvt (th) and 5.27×10^{18} nvt (over 0.4 eV). Comparison with known radiation damage thresholds for metals indicates no difficulty is to be expected from this direction.

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APPENDIX A. SITE AND FACILITY DESCRIPTION

The Tower Shielding Facility (TSF) (see Fig. 33) at the Oak Ridge National Laboratory was built in 1954 to enable studies to be made of asymmetric shield configurations for the Aircraft Nuclear Propulsion Project. This research required that the reactor radiation source be located in a region free from ground or structure scattering. The facility that was developed consists of four 315-ft-high towers erected on the corners of a 100- by 200-ft rectangle and the reactor is suspended between two of the towers on the 200-ft side of the rectangle. The other two towers are used to support other equipment such as secondary shields and detectors; the bridges which connect each pair of towers are also used for this purpose. Underground buildings near the tower array house the control equipment and the operating crew, and a pool provides shielding needed during shield changing and reactor servicing. The original reactor, known as the Tower Shielding Reactor I, was a box-shaped 500-kw MTR-type reactor. It was replaced in 1960 by a spherically symmetric reactor, the TSR-II which is capable of operating at 3 MW and more closely simulates the idealized spherical reactor used as a basis for machine calculations.

The TSF has turned out to be remarkably versatile, and useful for many studies other than the mobile reactor shielding studies for which it was originally intended. Among these varied uses are irradiation of animals for biological studies and investigation of transport and shielding of radiation from nuclear weapons.

A.1. Site Location

The Tower Shielding Facility is located on a knoll with an elevation of 1.069 ft 2.35 miles south-southwest of ORNL, 6 to 13 miles from the city of Oak Ridge, and 17 to 25 miles from the city of Knoxville (see Fig. A.1). The Melton Hill Dam of the Tennessee Valley Authority is located 0.8 mile south of the TSF on the Clinch River, which forms a natural boundary of the restricted area. The nearest ORNL facilities are the Health Physics Research Reactor (HPRR) and the High-Flux Isotope Reactor, both of which are over 6,000 ft from the TSF and separated from

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Fig. A1. Map of Cities and Counties Surrounding Oak Ridge Area.

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it by an offshoot of the highest point of Copper Ridge (see Fig. A.1). The immediate terrain on all sides of the tower structure slopes downward at the base of the towers; approximately 400 ft to the north of the towers the grade gradually rises to the top of Copper Ridge.

A.2. Fencing

Since both the TSF-II and the HPRR are classified as unshielded, they are situated within the exclusion area shown in Fig. 33 which is enclosed by a 6-ft-high chain-link type of fence topped with three strands of barbed wire. Within the exclusion areas the TSF and HPRR are separated by a 5-ft high field wire fence. In addition, the TSF is enclosed by a standard AEC-type security fence with a 600-ft radius and the HPRR reactor building is enclosed by a 5-ft-high chain-link fence. The reactor is operated so that radiation levels at the perimeter fence do not exceed civilian tolerance, and the levels are continuously monitored as an additional precaution.

A.3. Underground Buildings

Two reinforced-concrete underground buildings, with dimensions of 59 x 30 x 10 ft and 115 x 30 x 10 ft, adjacent to and north of the towers (see Fig. A.2) provide a shielded working area for personnel during reactor operation. The smaller building is used primarily as a service and shop area and is connected to the larger building by an 8-ft-wide walkway. The larger building contains the reactor controls, data-collecting facility, counting room, and offices (see Fig. A.3). The buildings are shielded against radiation by an 18-in.-thick concrete roof covered with 3 1/2 ft of earth.

A.4. Tower Description

The tower structure is braced and guyed steel frame forming a 100-by 200-ft rectangle, with a column placed at each of the four corners (Figs. A.2 and A.4). Each column is 9-ft square and 315-ft high and terminates at the lower end in an inverted truncated pyramid. Since the objective of the design was to minimize the scattering of radiation by

Fig. A2. Tower Shielding Facility Site Plan.

Fig. A3. Control House, Tower Shielding Facility, ORNL.

No representation or warranty, expressed or implied, is made that the use or disclosure of any information, apparatus, method or process disclosed in these drawings may not infringe private rights of others. No liability is assumed with respect to the use of or for damages resulting from the use of, any information, apparatus, method or process disclosed in these drawings. The drawings are being made available for information to bidder and are not to be used for other purposes, and are to be returned upon request of the forwarding contractor.

Fig. A4. Tower Shielding Facility, Oak Ridge National Laboratory.

structural material, the unit weight of the steel in the tower structure was kept below 400 lb/ft. Each pair of towers is joined at the top by a horizontal truss-type bridge running east and west as shown in the three-dimensional view of Fig. A.4. The bridge between the two legs of the north tower at the 100-ft level provides access to the terminals for instrument and reactor control cables.

A.5. Guy Cables

A pair of galvanized plow-steel cables (or sister guys) 2 in. in diameter stretches between leg I of the north tower and leg II of the south tower, and another pair stretches between leg IV of the north tower and leg III of the south tower. Each cable of the pair is held in 40,000-lb tension by two similarly constructed pairs of guys which extend from the top of the towers to the ground anchors in the north-south direction. The towers are guyed in the east-west direction by eight similar cables. Each inclined guy cable was originally set at 76,000-lb tension and makes an angle of 32° with the towers.

A.6. Wind Loading

Although the towers were designed to deflect no more than 11 in. with a 105-mph wind and with the maximum loads at the maximum altitude, the reactor is not elevated for operation when winds exceed 40 mph. To ensure that operations are not conducted in dangerous wind conditions, contact is maintained with the weather bureau and local observations are made with an anemometer and wind-direction indicator. In addition, the TSF is equipped with weather radar to track local thunderstorms so that the loads, especially the reactor, can be lowered before a thunderstorm arrives at the TSF area.

A.7. Lightning Protection System

The tower structure is completely shielded by means of a wire grounding net. As shown in Fig. A.4, a copper-clad steel-strand shielding wire is mounted on porcelain insulators to form a rectangle at a 5-ft minimum above the entire structure. Shielding wire also extends from the top of

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each tower to the ground to protect the inclined guys. The steel towers, the inclined guy wires, and the grounding system are connected to a buried counterpoise. The resistance to the ground of the above-ground grid system is between 1 and 3 ohms.

A. 8. Hoisting Equipment

The equipment for hoisting the reactor and for hoisting and positioning other loads consists of six major hoists, designed to be operated in pairs or singly, with the hoist drums and motors located in a hoist house to the west of the towers. The hoisting lines and the necessary ground sheaves, idlers, etc., extend along the ground from the hoist house to the base of the towers and then to fixed multipart sheave blocks near the top of the columns or on the horizontal bridges atop the towers and then down to multipart traveling blocks. The reactor traveling blocks are connected to the loads by pairs of 2-in.-diam cables; the other blocks are connected to their loads by single 2-in.-diam cables. Two of the hoists, which can lift 55 tons to an altitude of 200 ft, are used to move the reactor and shield in the plane of the west tower legs (I and II). Another pair of hoists can raise a load of 40 tons to the same altitude in a plane 35 feet east of the plane of legs I and II. The third pair of hoists can lift 30 tons in the plane of the east tower legs (IV and III), or together with the second pair of hoists can be connected to one 30-ton load so that the load can be positioned at various distances from the reactor. During this type of operation an auxiliary constant-tension hoist can be used to limit the relative oscillatory motion of the auxiliary load and the reactor due to winds.

There are two separate and independent hoist control stations: one, used for local control, is located in a partially shielded ramp at the south entrance to the underground building and contains a closed-circuit television for observing the moving loads, and the other used for remote control, is located adjacent to the reactor-handling pool. The hoist controls are designed to provide smooth acceleration and retardation through five speeds and to brake automatically in case of power failure. In the event of an emergency an auxiliary battery power source can be operated from either hoist control station to release the brakes and thus

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lower the loads. The loads are protected from operator error by limit switches and slack-line switches. Moving loads can also be observed by television from the reactor control console. Procedures for operating the hoists are in Section 6.0 of the TSF manual.

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APPENDIX B. DRAWINGS

The drawings presented in this appendix have been selected as basic for indicating some mechanical and electrical features of the reactor installation. Two of these, Figs. B7 and B8 are drawings produced by Atomics International specifically for the TSF-SNAP reactor. The remaining drawings were produced at ORNL for the mechanical and electrical systems for positioning the reactor.

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REMARKS

- REMARKS:
1. PART 1 MAY BE PURCHASED FROM THE TORRINGTON COMPANY, SOUTH BEND 21, IND.
PART 11 IS A SELF-ALIGNING CYLINDRICAL ROLLER THRUST BEARING - 7.000 BORE
TO 50.00 D.D. NO. 70 TPS 129.
2. THE FOLLOWING PARTS MAY BE PURCHASED FROM OHIO GEAR COMPANY, 1335 EAST 17TH ST., CLEVELAND, OHIO 44115. PART 19 - RA-1 SPEED REDUCER RATIO 1:1 ASSEMBLY D.
PART 30 - RA-1 SPEED REDUCER RATIO 1:1 ASSEMBLY E.
3. THE FOLLOWING PARTS MAY BE PURCHASED FROM BOSTON GEAR WORKS, QUINCY 71, MASS.
PART 20 - COUPLING CAT. NO. FC8B0 HOLE DIA. 1 1/2 ONE END 1" ONE END. PART 2
FLANGE CAT. NO. 60R3A HOLE DIA. 1 1/2 ONE END 1" ONE END. JOINT CAT. NO. J100.
PART 48 - BEARING CAT. NO. FR-2024-10.

NOTES

- NOTES:
1. SHIM AS REQUIRED BETWEEN PARTS 35 AND 36 TO CLEAR PART 39 WITH PART 35 .001 TO .002.

POSITIONING MECHANISM-DET.SH.6	EJ-213
REACTOR SUPPORT STRUCTURE-ASSY.	EJ-300
REACTOR POSITIONING DRIVE-ASSY.	EJ-200
POSITIONING MECHANISM-DET.SH.6	EJ-212
POSITIONING MECHANISM-DET.SH.4	EJ-211
POSITIONING MECHANISM-DET.SH.9	EJ-210
POSITIONING MECHANISM-DET.SH.2	EJ-209
POSITIONING MECHANISM-DET.SH.3	EJ-208

REFERENCE DRAWINGS	NO.
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OAK RIDGE NATIONAL LABORATORY

OPERATED BY
Union Carriage Corporation

UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

10A EXPERIMENT-TSF BLDG NO 770

FOR POSITIONING MECHANISM

ASSEMBLY

ASSEMBLY

PREPARED <i>F. J. ...</i>	ACCEPTED <i>Joseph ...</i>	APPROVED <i>H. S. ...</i>
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✓	M20859	EJ	207	E	REV 1
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Fig. B1. Reactor Positioning Mechanism Assembly.

SUPPORT COLUMN ROTATOR MOTOR	ED-002-D
REFERENCE DRAWINGS	NO.
OAK RIDGE NATIONAL LABORATORY OPERATED BY UNION CARBIDE CORPORATION OAK RIDGE, TENNESSEE	
SNAP-TSF INSTALLATION REACTOR EQUIPMENT NO. 7701	
LIMIT SWITCH ELECTRICAL DETAILS	
SUBMITTED 11/1/66 ACCEPTED 11/1/66 APPROVED 11/1/66	SUBMITTED 11/1/66 ACCEPTED 11/1/66 APPROVED 11/1/66
E 20862	ED 007 D REV

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TOLERANCES UNLESS OTHERWISE SPECIFIED:
 FRACTIONS =
 DECIMALS =
 ANGLES =
 SCALE: 1/4" = 1'-0"

NO.	REVISIONS		DATE	APPD	APPD
	DATE	REVISIONS			
1	5-24-66	MEADE	5-24-66		
2	5-23-66	BROWN	5-23-66		
3	7-10-66	CHECKED	7-10-66		

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Fig B2. Limit Switch - Electrical Details.

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Fig. B3. Fission Chamber Motor - Electrical Schematic.

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Fig. B4. Reactor Elevating Motor - Electrical Schematic.

Fig. B5. Reactor Rotator Motor - Electrical Schematic.

Fig. B6. Support Column Rotator Motor - Electrical Schematic.

REVISIONS			
REV	DATE	DESCRIPTION	APPROVED
1	10/10/66	1. DRAWING PART NO. CHANGED 7670-11006 (WAS) EX 7670-11006	

FOR INFORMATION
WILL NOT BE REPLACED AND MAY
BE CHANGED WITHOUT NOTICE

12. AFTER SPINNING AND BEFORE MACHINING, FULL ANNEAL PER SPEC LA 0111-024, EXCEPT TEMPERATURE SHALL BE 1950°-2050°F.
13. SIZE TO BE DETERMINED BY FABRICATOR.
14. ULTRASONIC TEST RAW MATERIAL PER SPEC NA 0115-003.
15. CARBON CONTENT RANGE .06% TO .08%.
16. FINISHED PART TO BE PENETRANT INSPECTED PER SPEC NA 0115-001.
17. PLANE "X" SHALL BE DEFINED AS A PLANE PERPENDICULAR TO THE $\hat{C}$ OF THE SHELL AND PASSING THROUGH THE LOWEST POINT OR POINTS OF THE SHELL. ALL OTHER LOW POINTS OF SHELL BOTTOM TO BE WITHIN .03 OF PLANE "X".
18. INSIDE DIA 8.875 ± .010, SIZE DETERMINED BY CIRCUMFERENTIAL TAPE $\pm .001$ MINUS 2X AVE. WALL THICKNESS.
19. MAX OUT OF ROUNDNESS .020 T.I.R.
20. MAX OUT OF ROUNDNESS .010 T.I.R.
21. INSIDE DIA 8.875 ± .020, SIZE DETERMINED BY CIRCUMFERENTIAL TAPE $\pm .001$ MINUS 2X AVE. WALL THICKNESS.
22. IDENTIFY PER SPEC NA 0104-001 DO NOT IMPRESSION STAMP.
1. P&P DATE PER SPEC NA 0103-003.
- NOTES: UNLESS OTHERWISE SPECIFIED

HALF SIZE PRINT

QTY	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	DATE	MATERIAL	DATA SPECIFICATIONS SIZES, NOTES, VENDORS	ZONE	LINE NO.
1	7670-12006	SHELL					010
							009
							008
							007
							006
							005
							004
							003
							002
							001

ITEM	QTY	UNIT	REMARKS	DATE	BY	CHK BY	APPROVED BY	QTY IDENT NO	SIZE	DATE	BY	CHK BY	APPROVED BY
1	1	TSF	7670-12006					09974	E	7670-12006			

Fig. B8. Vessel Assembly.

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