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APPLICATION OF SNAP 2 TO MANNED ORBITAL SPACE STATIONS

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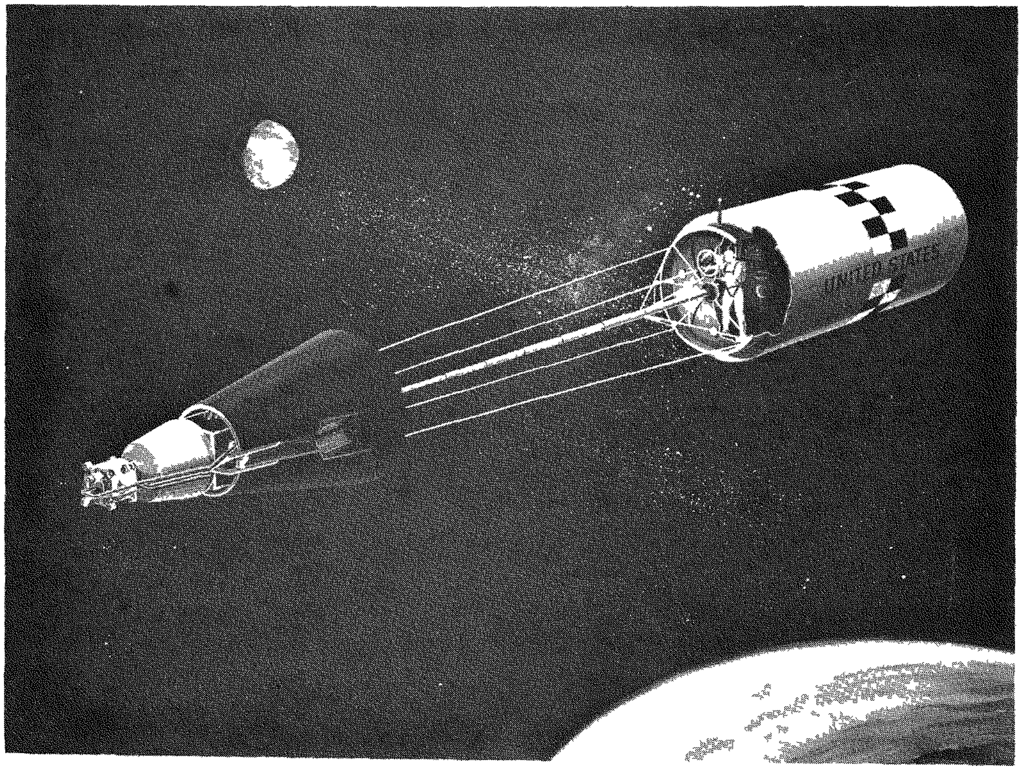

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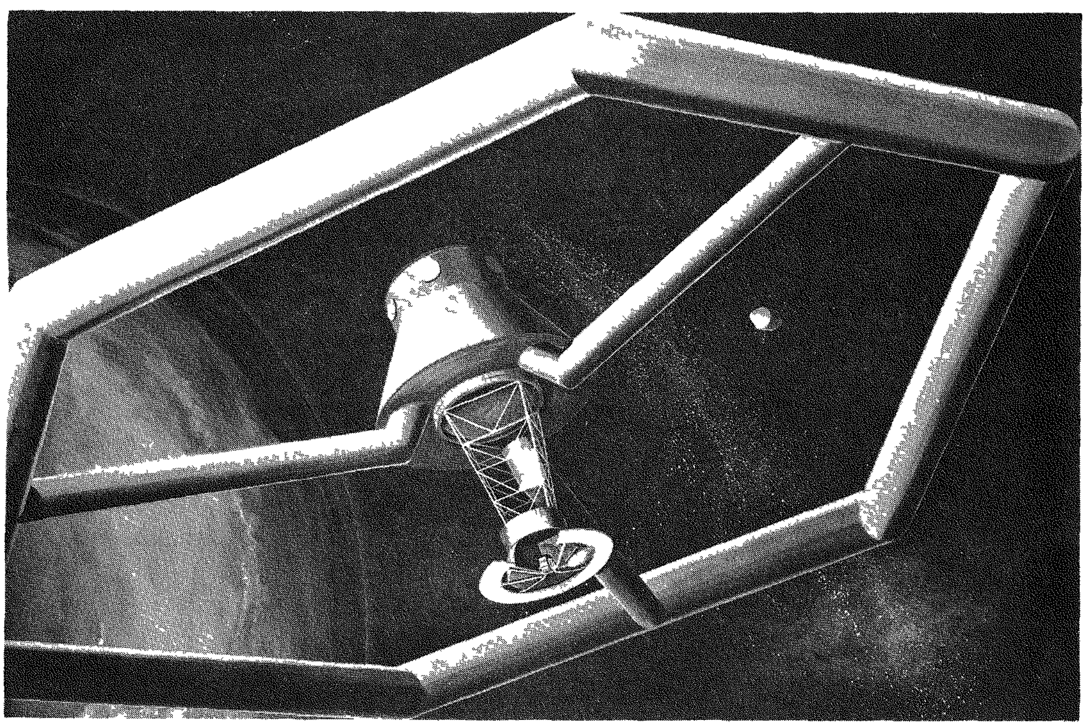
Cylindrical Space Station with SNAP 2 Power Plant



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I. SUMMARY

Extension of man's capability and utilization of space will require the development and operation of large long-lived space stations. These systems will require electrical power generating systems in the 10 to 100 kwe range with lifetimes on the order of years. In this application nuclear-electric space power systems exhibit certain operational and economic advantages over solar systems at relatively modest power levels. At the higher power levels the nuclear systems are probably the only systems that can seriously be considered.

The present series of SNAP reactor systems, i.e., the 0.5 to 2 kwe SNAP 10A, the 3 to 11 kwe SNAP 2, and the 35 kwe SNAP 8, represents a closely related family of nuclear power systems that can be modified to accommodate the power range of interest. The advanced stage of development of these space power systems and the scheduled demonstration flight tests will uniquely qualify them for consideration in the manned orbital space station application in the time period required.

This report presents the results of a study of the installation and operational characteristics of a SNAP reactor system integrated with a manned space station. The reference system selected was an 11 kwe version of a SNAP 2 system employing multiple power conversion units coupled to a single reactor source. Of prime importance is the reactor radiation shield required for the manned system and the use of design features which minimize the shield weight. The weight of the radiation shield is highly dependent upon the geometrical configuration of the space station and the reactor since "shadow" shielding of the manned compartments is required for minimum weight systems.

The installation and shielding requirements of the 11 kwe system were considered for two types of space station configurations illustrated in the frontispiece; one was a 10 ft diam. cylindrical station with a reactor separation distance of 50 ft and the other a 150 ft diam torodial station with the reactor located in the hub. The weight of the power system installed in the cylindrical space station was about 9000 lb of which 6000 lb was required for shielding. The weight of the system for a torodial station was ~25,000 lbs of which 20,000 lbs was required for shielding. However, it is important to note that these weights are relatively insensitive to power level and that doubling or tripling the power output will only increase these weights by a small percentage. In addition, the designs

developed for these two concepts permit the replacement of the reactor and power conversion system with the radiation shields becoming essentially a permanent part of the space station. Hence the large weight penalty associated with the reactor shield only has to be incurred once during the life of the space station.

Because of the dominance of the reactor shield weights for manned systems, parametric curves of shielding weight are presented to enable the space vehicle designer to estimate nuclear power system weights for configurations other than those selected in this report.

II. INTRODUCTION

A. THE MANNED ORBITING SPACE STATION

With the advent of large booster systems and the success of the Mercury manned orbital flights, interest has intensified in the manned space station concept. Of all the manned space missions that might be undertaken in the near future, the utility and benefits derivable from such a system are the most discernible. The reliability of automatic, unattended long-lived complex space systems has proved, to date, somewhat disappointing and man may well have an important role to perform in space if only as a highly versatile serviceman for these exceedingly sophisticated systems. Certainly the manned space station offers a highly flexible concept for a multipurpose centralization of many of the applications envisioned for earth satellites.

The more ambitious missions, such as manned exploration of near space and the adjacent planets, will require the development of a technology of very large and long-lived man-spacecraft systems. Because of this the manned orbiting space station represents not only a goal unto itself, but a logical intermediate development objective towards extended manned missions farther into space.

B. SELECTION OF A POWER SYSTEM

The power system for the manned space station must be capable of delivering 10 to 100 kwe for extended periods of time (a year or longer). The electrical load for support of the men only will be on the order of 1 to 5 kw/man. In addition the number of functions or missions that can be performed will be to a large extent dependent upon the electrical power available in the space station.

For space power systems in this power range and application, solar, nuclear, or chemical systems can be considered; however, because of the immense fuel requirements for even the most efficient chemical systems, nuclear or solar systems will require the least logistics support. The various solar systems all suffer from extremely large area requirements for collectors or panels, coupled with the complexity of maintaining an accurate constantly changing solar attitude. The nuclear system, on the other hand, requires a relatively small area, does not require a preferred orientation and is unaffected by the sun-shade transient. Figure 1 compares the area requirements

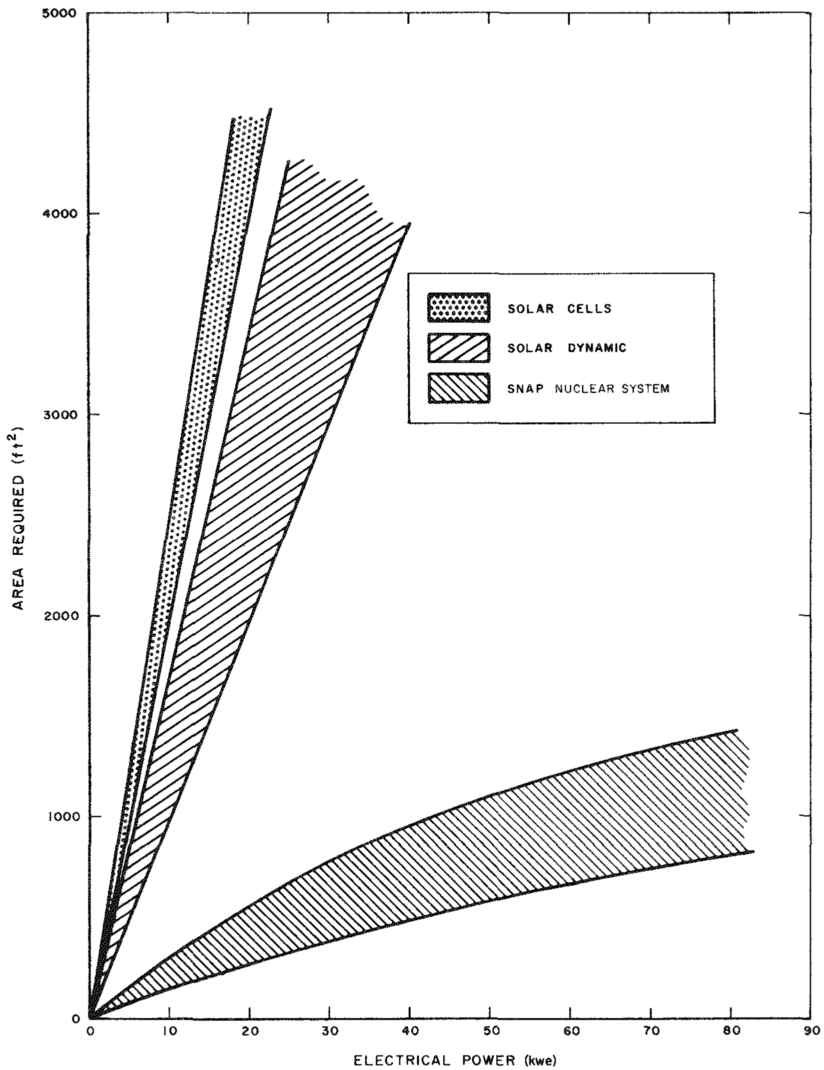

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Figure 1. Comparison of Area Requirements for Power System

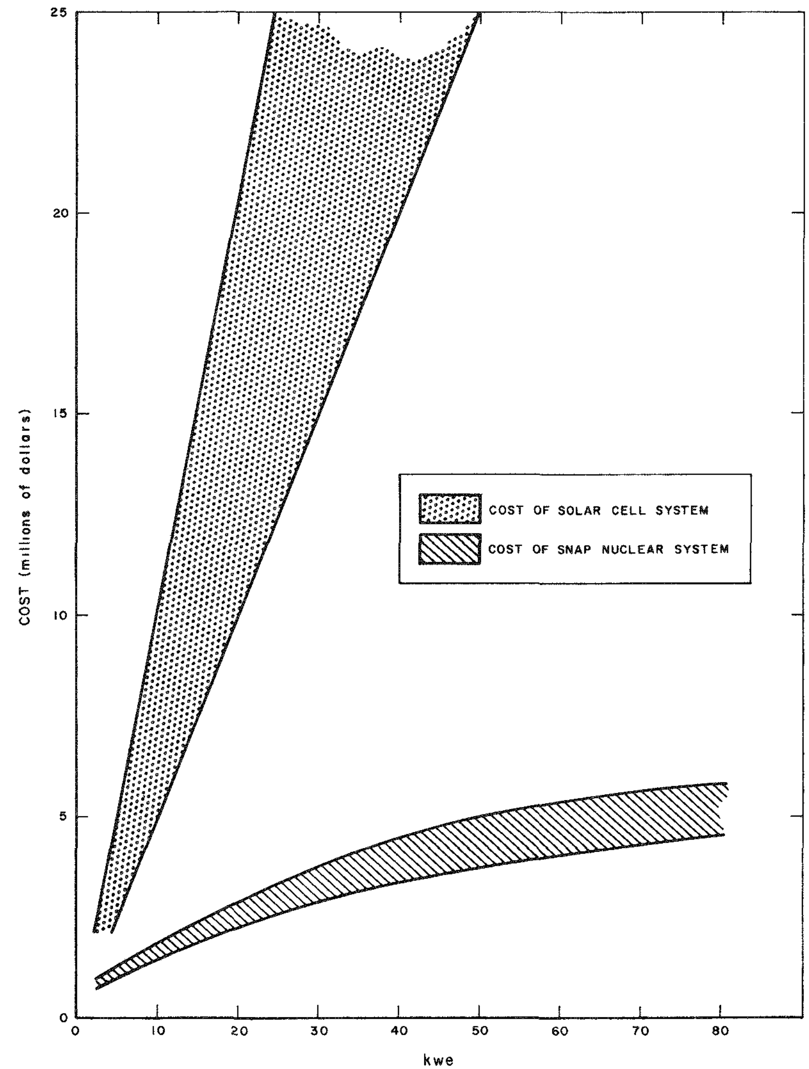

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Figure 2. Comparison of Cost of Power Systems

for the solar cell, solar dynamic, and the SNAP 2 through 8 series of nuclear systems. In the region above 10 kwe, the solar systems begin to be prohibitively large in size.

In the area of costs another distinct advantage can be shown for nuclear systems. Solar cell systems require a very large number of expensive units while the nuclear system contains a relatively small number of components that can be manufactured at reasonable costs. A 3 kwe SNAP 2 produced in small quantitites will cost about 0.75 to 1.0 million dollars. An 11 kwe advanced SNAP 2 incorporating three power conversion systems would cost 1.5 to 2.0 million dollars. The 30 kwe SNAP 8 system might cost on the order of 3 to 4 million compared to a cost of 15 to 30 million dollars for a solar cell system. Figure 2 presents a cost comparison of the solar cell and nuclear system vs power. Cost figures on the solar dynamic systems were not available for comparative purposes. The large surfaces and high accuracies required for the collectors doubtlessly will make these systems expensive.

The weight of a nuclear power plant is extremely dependent upon the shielding geometry which is determined by the configuration of the space station. Figure 3 compares the weight of a nuclear system to a solar cell system as a function of power output and for two types of space station configurations. For a 10 ft diam cylindrical space station with a 50 ft separation distance, the nuclear system exhibits a weight advantage at about 12 kwe while, for a 150 ft diam torodial station with the reactor located at the hub, a weight advantage appears at about 37 kw.

By far the largest part of the weight of the nuclear system for manned missions resides in the radiation shield (~80% for the large torodial space station). For large long-lived space stations it is reasonable to expect that the power plants will, for one reason or another, be replaced during the operational life of the stations. However, it should be unnecessary to replace the large biological shield associated with the reactor if the system is properly designed so that replacement weights for the nuclear system will be those of the unshielded-nuclear system weights shown in Figure 3.

From the preceding discussion it can be seen that, in the 10 to 100 kwe range for manned space stations, nuclear power systems are considerably more compact and less expensive and offer a significant weight advantage for the higher power levels and for those lower power levels of favorable geometry.

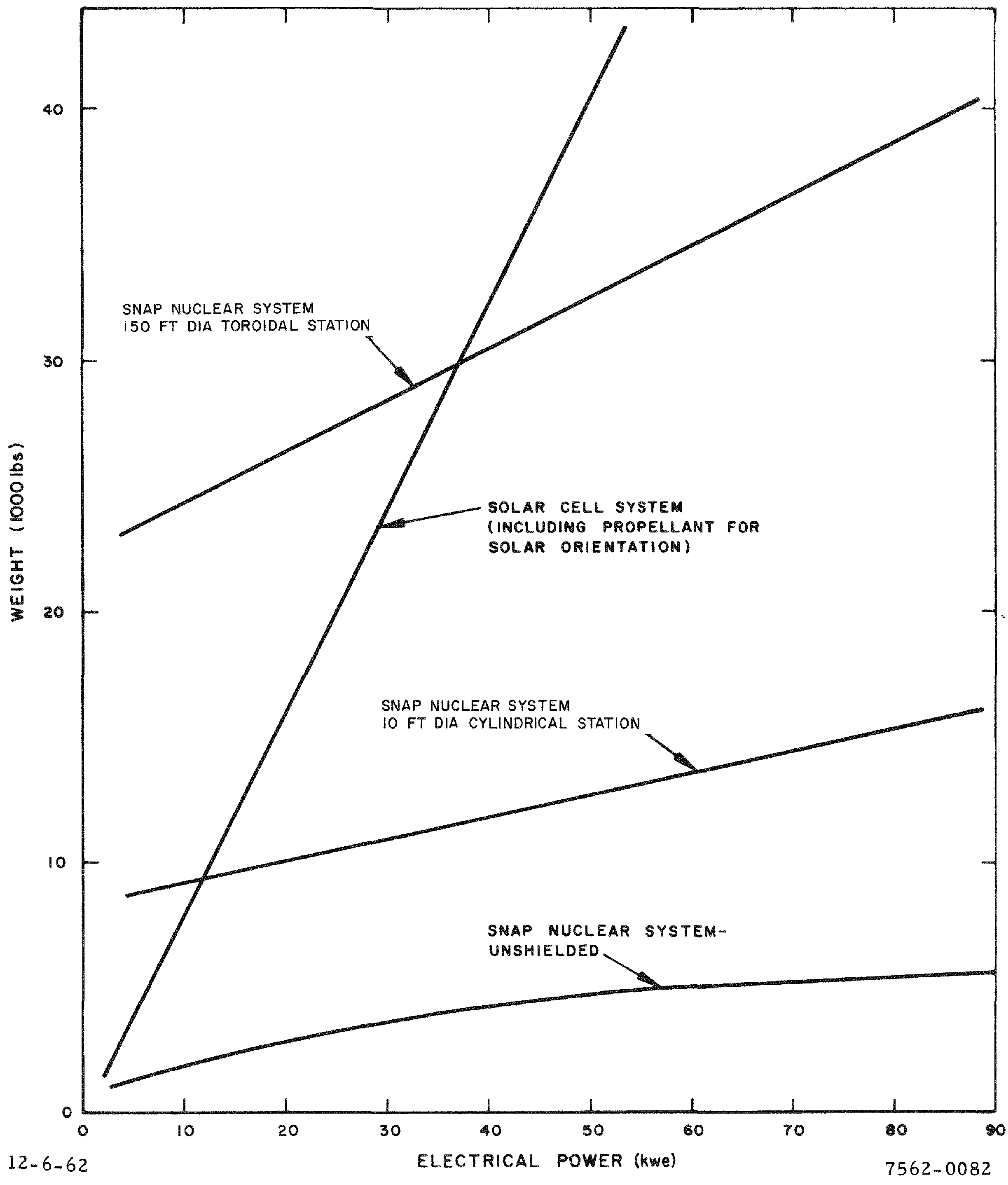

Figure 3. Comparison of Weights of Power Systems

III. DESIGN CONSIDERATIONS

A. TYPES OF SPACE STATIONS

Although there are a large number of general configurations possible for a manned space station, the requirement for artificial gravity has favored either a dumbbell type arrangement, with two or more widely separated masses, or a large rotating torus.

Most of the important factors involved in integrating a nuclear power plant, with the outstanding exception of the shielding factor, are independent of the choice of configuration. Some of the different types of space station configurations present more favorable shielding geometries than do others.

B. ACCEPTABLE LEVELS OF RADIATION

Radiation has always been a part of man's natural environment. Cosmic rays, radioactive elements in the earth, and atmosphere, and radioactive materials, such as K^{40}, which are present in the human body, represent some of the natural radiation background. Obviously, small amounts of radiation can be tolerated; however, setting upper limits on radiation exposures has been a perplexing and difficult problem. The following list presents a summary of the recommendations by the National Committee on Radiation Protection (NCRP) as related to personnel working in the Atomic Energy field.

1) The maximum permissible dose (MPD) shall not exceed 0.1 rem in any one week.
2) The MPD shall not exceed 5 times the age minus 18, i.e., 5 (age-18) rem, during lifetime.
3) The MPD shall not exceed 200 rem during normal life span.
4) Maximum accidental or emergency exposure occuring once in a lifetime should not exceed 25 rem.

Higher dose rates than the laboratory tolerances listed above will be acceptable for the small number of people who will be involved in performing space missions. It is generally agreed that a dose of up to 50 rem received in a burst produces no detectable symptoms. A reasonable single mission dose appears to lie between 5 and 15 rem, depending upon the frequency of such missions considered for each individual.

Although large manned space stations with life span up to 5 years can be considered, duty times for men will probably not usually exceed a period of one month. A dose rate of about 7.5 mrem/hr (5 rem/month) has been taken as a design value for the manned spacecraft. Such effects as attenuation by structures and equipment and space radiation shielding may allow certain reductions in the reactor shield when these factors can be properly evaluated.

C. SHIELDING

1. Radiation From a Bare Reactor

The central problem of shielding is illustrated in Figure 4 which gives the dose rate from a reactor operating at 100 kw thermal as a function of distance from the reactor. Conversely, Figure 5 is a plot of the time required to receive 1 and 5 rem as a function of separation distance from an unshielded reactor operating at 100 kw thermal. The SNAP hydride type of reactor system was used to calculate these dose rates; these curves, however, should be relatively independent of the reactor type. Inspection of these curves indicates that, for a reasonable separation distance, i.e., 100 ft at 100 kwt the dose rate is 5000 rem/hr. From the previous discussion of allowable dose rates, it was seen that a dose rate of about 10^{-3} rem/hr is desired; hence the shield required will have to provide a dose rate attenuation of about 10^{7} to present an acceptable radiation environment for the manned portion of the space station. During operation the radiation from the reactor consists of both neutrons and gamma rays. After the reactor is shutdown the reactor still presents a significant radiation source consisting of decay gamma rays. The dose rate from an unshielded reactor that has been operated at 100 kwt and then shutdown for one hour is shown, vs separation distance, in Figure 6. This source of radiation is a slowly decreasing function of time. Also shown on this curve is the attenuation of the shutdown dose rate provided by a slab of depleted uranium 2.16 in. thick placed between the reactor and spacecraft.

2. The Shadow Shield

In order to obtain a minimum weight shield, it is necessary to employ the "shadow shield" concept. Using this technique shielding is provided around the reactor in such a manner as to cast a relatively radiation-free shadow over the volume occupied by the space station. For the volume outside of this acceptable

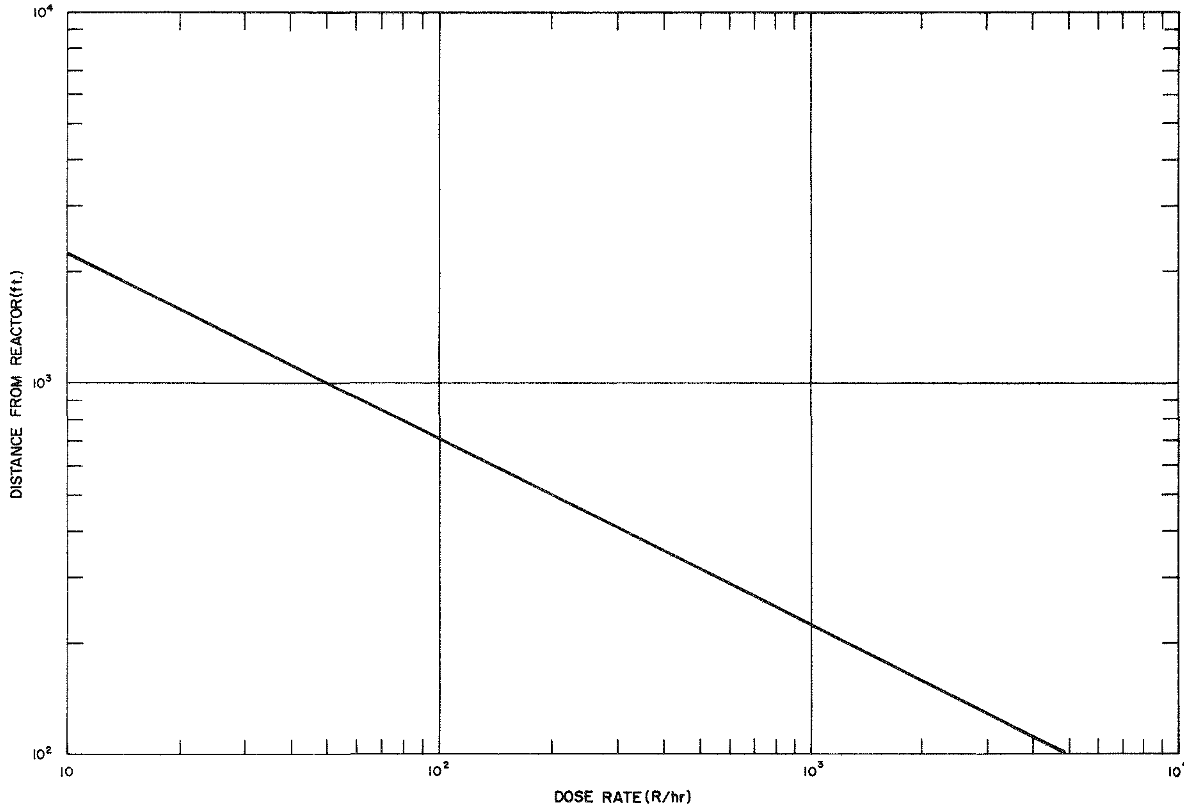

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Figure 4. Radiation Levels from Unshielded Reactor

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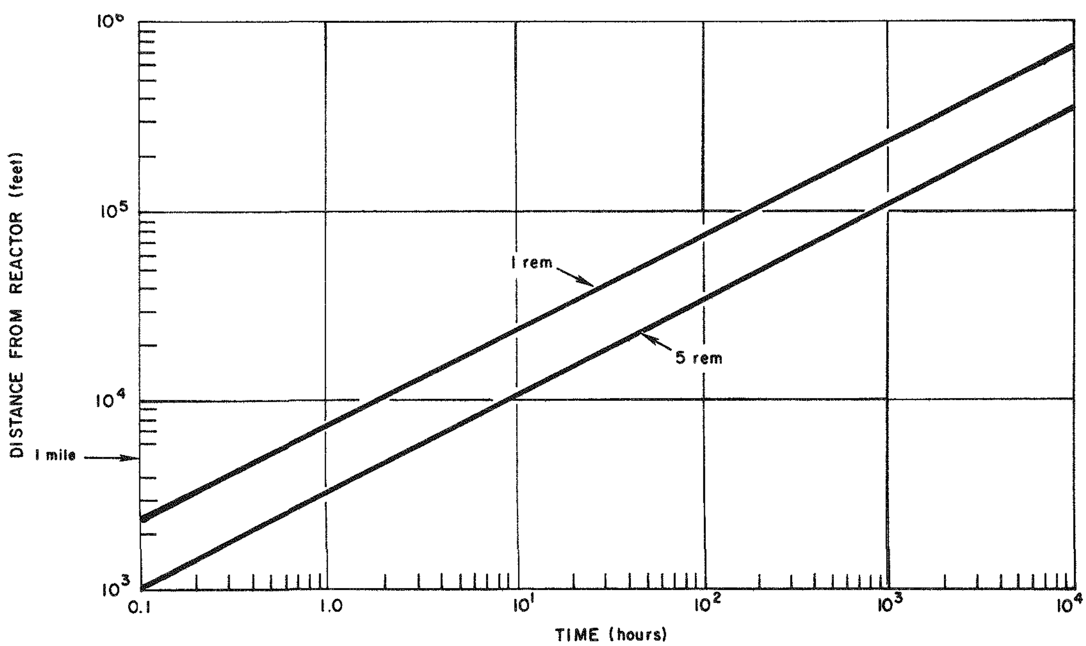

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Figure 5. Maximum Residence Times Outside Shielded Cone with Reactor Operating

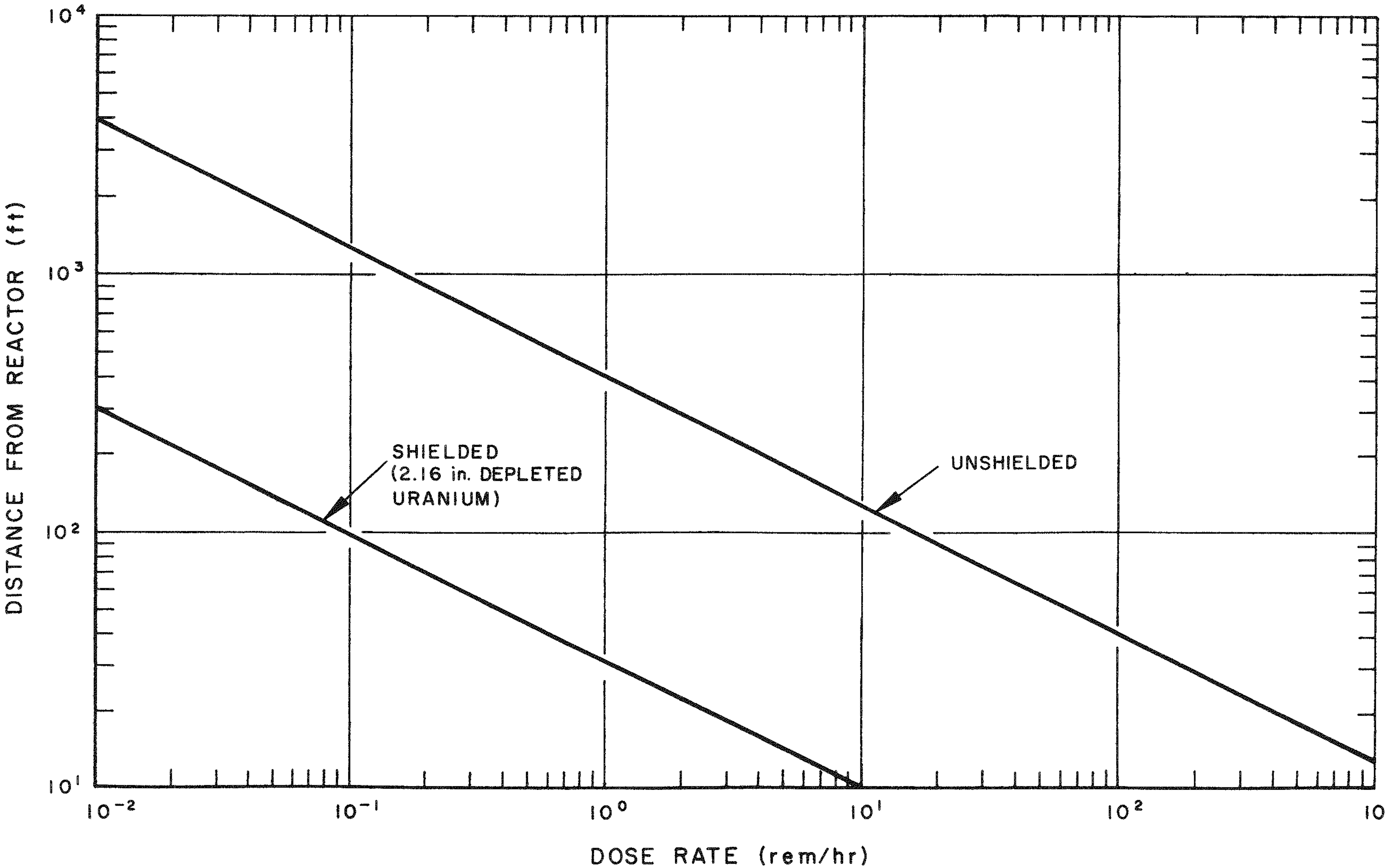

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Figure 6. Dose Rates One Hour After Shutdown

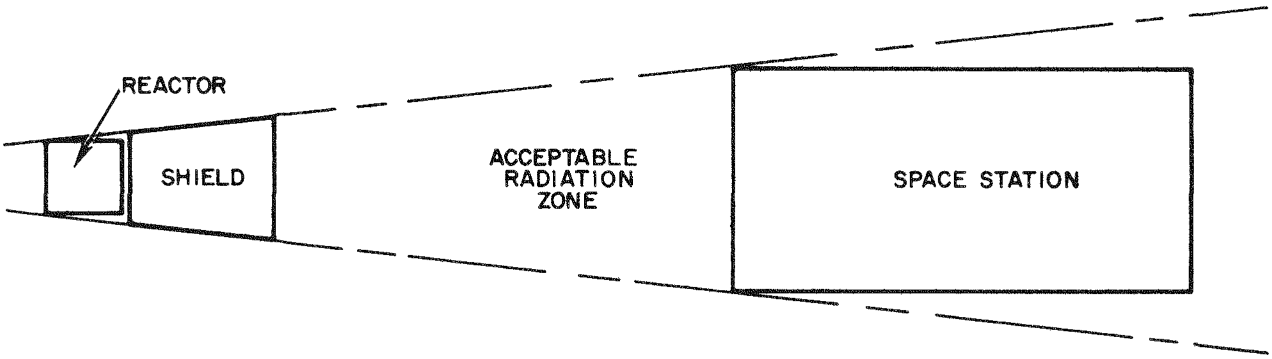
radiation zone, the radiation is essentially that of an unshielded reactor unless other constraints are imposed that make this situation unacceptable. The shadow shield concept is illustrated in Figure 7a, b, for two typical space station configurations.

The most favorable situation, from a weight standpoint, for reactor shielding occurs when the manned spacecraft can be protected by a shadow shield subtending a small spherical angle. This situation is realized when the reactor is separated from the payload by at least two payload diameters. Practically, this separation can best be achieved with a payload of relatively small cross section. This case is represented by the cylindrical station depicted in Figure 7a.

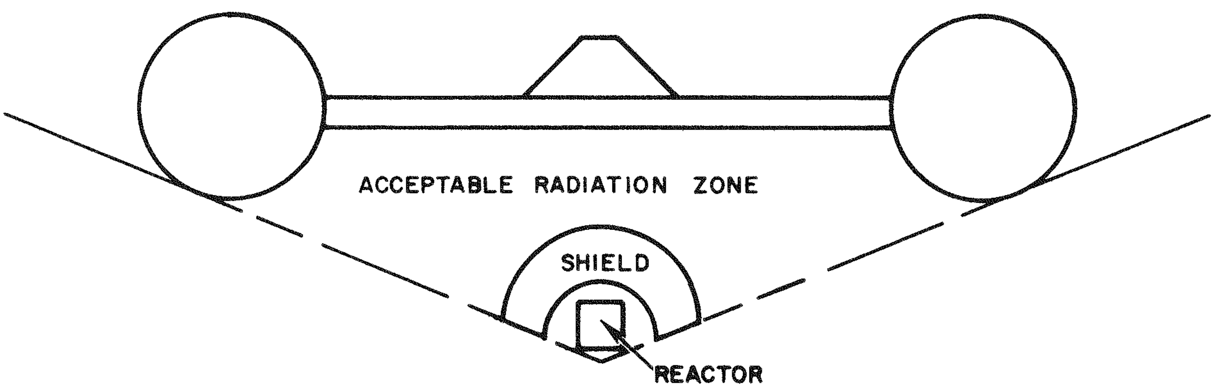
The shadow shield concept provides a zone that permits personnel protection from the reactor radiation and excursions outside of this zone by personnel can be permitted only when the reactor has been shutdown. The shutdown dose rate, too, as shown in Figure 6, may be prohibitive except when large separation distances are involved. When access outside the radiation zone offered by the main biological shield is required, the concept of a shutdown shield is employed. This is a shield that reduces the reactor shutdown dose to a level where access is permissible. Since the shutdown dose is entirely made up of gamma rays this shield is usually composed of a heavy metal such as depleted uranium. The thickness of heavy metal required is dependent upon the acceptable dose rate which is determined by the operational procedures envisioned, required length of time and frequency of access, percentage of the total radiation dosage to be received during the access, and other parameters. The attenuation afforded by 2.16 in. depleted uranium is illustrated in Figure 6.

3. Shield Geometry

The radiation from a reactor consists of both neutrons and gamma rays released in the fission process. As developed in Section VIII, the section on Shielding, a manned-system shield will consist of (1) regions of heavy material for gamma ray attenuation, (2) regions of light material, usually hydrogenous, for neutron attenuation, and (3) hydrogenous heavy material, such as zirconium hydride, to shield against both gamma rays and neutrons. It is important that these materials be placed so that reactions of the neutrons with the heavy atoms do not produce excessive secondary gamma radiation. Yet as spherical geometries are approached, it is necessary to place the heavy material as close to



a. CYLINDRICAL SPACE STATION



b. TOROIDAL SPACE STATION

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Figure 7. Shadow Shield Concept

the reactor core as possible followed by the lighter, neutron shield material.

D. OPERATIONAL REQUIREMENTS

In order to provide a consistent set of assumptions to estimate the performance of a nuclear power system as applied to the manned space station, the following listed criteria are used; these are independent of the geometry or type of station involved.

1) The main biological shield provides a zone of acceptable radiation levels in those portions of the space station occupied by personnel.

2) Shutdown shielding will be installed or available over the remaining portion of 4π space to the extent of providing personnel access for short periods of time to accomplish rendezvous with the space station of unshielded vehicles, maintenance, etc.

3) The reactor and power systems must be capable of repeated startup and shutdown as a corollary of (2) above.

4) The reactor shield and power conversion system must be capable of installation in the space station while in orbit.

5) The reactor and power conversion system (but not necessarily the shield) must be capable of replacement with new units in the space station and safe disposal must be made of the spent or failed reactor and conversion system.

6) Failure of the reactor by any reasonable means must not present an unacceptable hazard to the space station or to the personnel involved.

While the above ground rules can be relaxed in some cases, they describe the general design criteria that a nuclear power plant must meet to offer the space station designer an attractive electrical power supply.

In the following two sections, the design of an 11-kwe SNAP nuclear electrical system for a cylindrical and a large toroidal space station are considered. In both cases the criteria listed above are met.

IV. THE CYLINDRICAL SPACE STATION INSTALLATION

A. DESCRIPTION

The man-rated SNAP 2 power system for the 10 ft diam cylindrical space station depicted in Figure 8 utilizes the 11 kwe advanced SNAP 2 power plant described in Section VII. The major components of this system are listed below.

1) Three independent 4 kwe (gross) mercury power conversion systems.
2) A biological radiation shadow shield that reduces the operating reactor dose rate to 7.5 mr/hr within the shadow cone 50 ft from the reactor.
3) A deployable reactor shutdown shield which can be placed around the reactor to reduce the shutdown radiation 1 hr after operation to a level of 1 r/hr at 30 ft in any direction.
4) Supporting and erecting structure.

A layout of the selected configuration is shown in Figure 9. The power plant can be separated for disposal from the biological shield and attaching structures. After the used power plant package is at a safe distance from the space station it can be disassembled by means of a destruct charge aimed at the reactor. A replacement power plant package may then be brought up by the shuttle vehicle and mated to the biological shield.

1. Shadow Shield

The shielding materials and their thicknesses required for the shadow shield are determined from the data developed in Section VIII, specifically from Figures 24, 25, and 26 of that section. The shield shape is determined by the thickness required and the angle is determined by a line tangent to the 10 ft diam payload and the furthermost corner of the reactor. For the case presented the shadow shield is in the shape of a frustum of a cone with a top diameter of 30.86 in. and a lower diameter of 36.12 in. It is composed of 2.36 in. of depleted uranium, 17.7 in. δ-phase zirconium hydride and 19 in. of lithium hydride. It is contained in a 1/8 in. can of 316 stainless steel. This shield

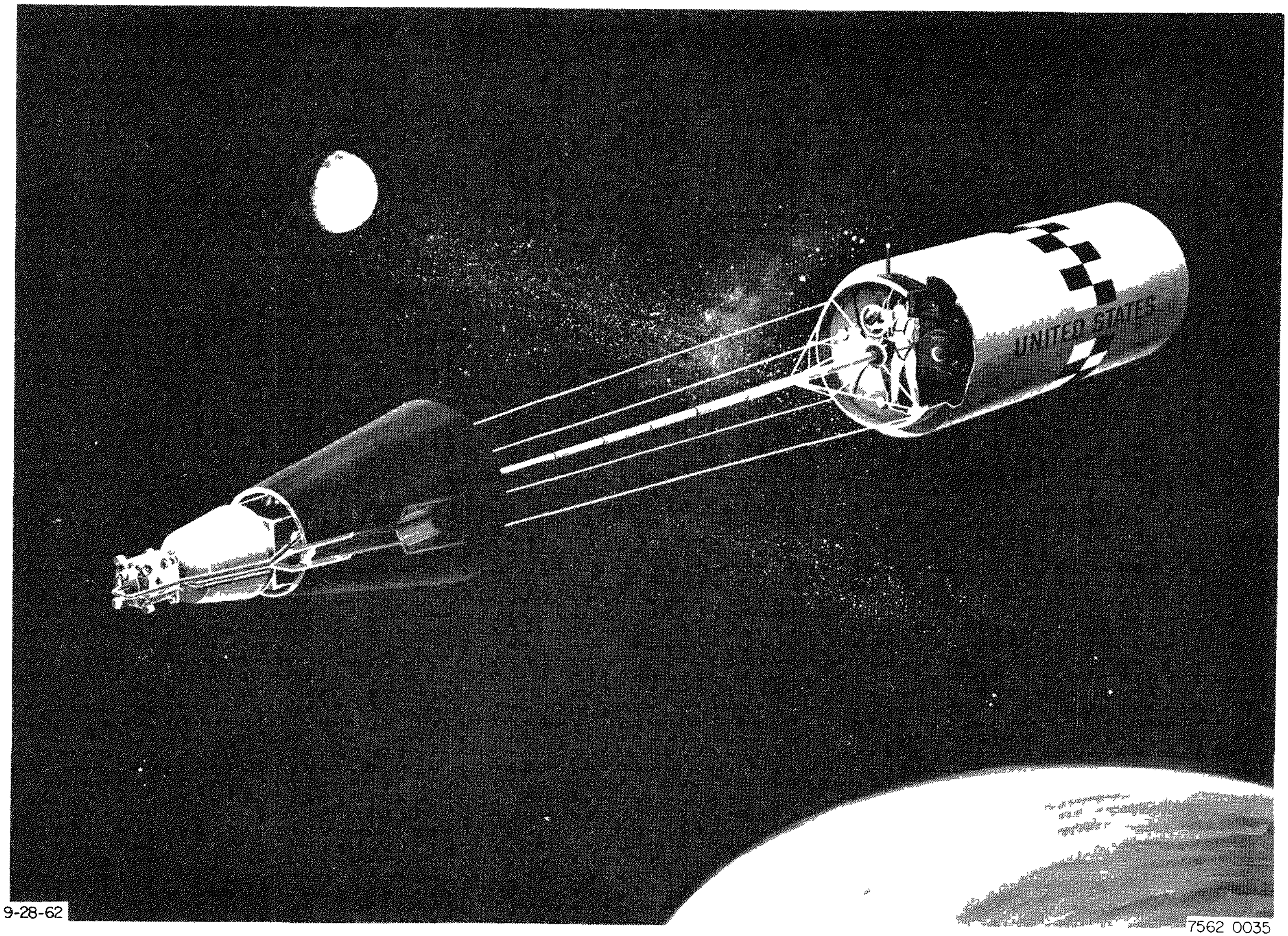

Figure 8. Cylindrical Space Station with SNAP 2 Power Plant

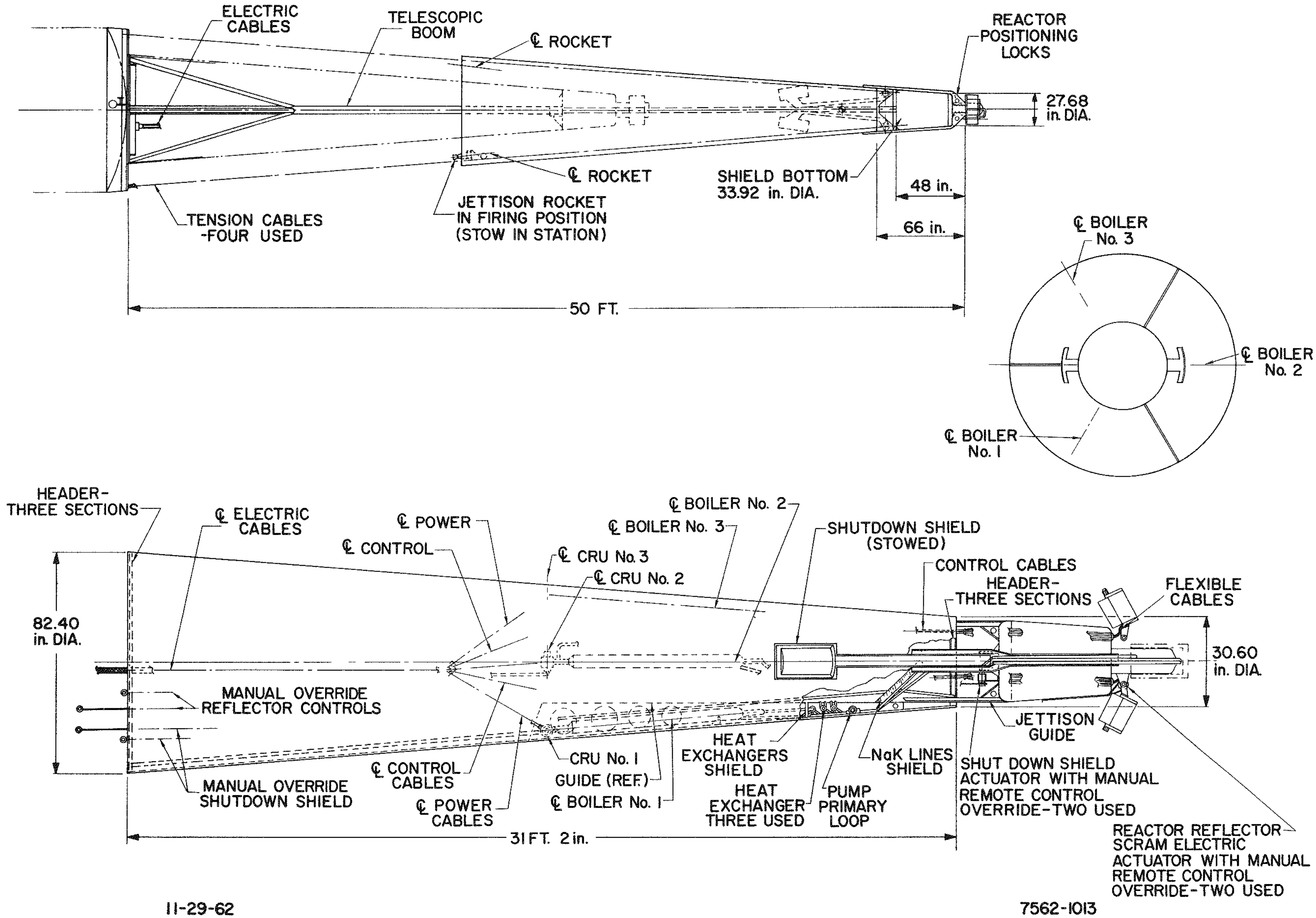

Figure 9. Man-Rated SNAP 2 Power Plant

provides a dose rate in the space station of 7.5 mr/hr during operation. A map of the dose rate levels is given in Section VIII, Figure 21, and includes conditions existing both during operation and 1 hr after shutdown with the shutdown shield in place.

2. Supporting Structures

The main supporting structure is used to absorb the collision shock during the rendezvous operation and to stabilize the bottom tube of the telescoping boom. It is a tubular truss which is locked to the bottom of the shield. It contains the mating ring which matches with the space station structure.

After installation, the locks are released and the power package can be extended 50 ft away from the station.

The telescoping boom is an integral part of the main structure. It consists of 5 tubes of different OD in telescoping arrangement. It will extend from 12 to 50 ft using compressed gas to provide the extension force. Four 1/8 in. cables attached to the shield and to the mating ring will supply the stabilizing reactions.

3. Shutdown Shield System

The shutdown shield is composed of two clam-shell halves of 2.16 in. thick depleted uranium. During reactor operation, the shutdown shield is stored behind the shadow shield to prevent activation due to the neutron flux. Prior to rendezvous with the shuttle vehicle the reactor is shutdown by rotating the control drums out;the reflectors are split away from the core. Then, the shutdown shield is moved from behind the shadow shield and brought to shielding position completely covering the reactor core. The maximum dose rate is now 1 r/hr at 30 ft from the reactor. The shuttle vehicle can then approach the station without excessive radiation.

4. Heat Rejection Radiator

Approximately 350 ft^2 of heat rejection surface area is required for the power system. In this configuration the radiator follows the cone swept out by the shadow shield. All components of the power system must be located within this cone to avoid the scatter of radiation into the payload section. The radiator consists of three separate and independent sections, each handling the heat rejection load from one of the power conversion systems.

B. WEIGHTS

The weight of the system shown in Figure 9 is given in Table I.

TABLE I
WEIGHT SUMMARY FOR CYLINDRICAL STATION
(lb)

Reactor System	320	
Secondary NaK Systems (3)	150	
Power Conversion Systems (3)	1080	
Radiator-Condensers (3)	660	
Total Unshielded Power System		2210
Uranium	950	
Zirconium hydride	1890	
Lithium hydride	460	
Stainless steel containers	500	
Conical Shadow Shield, Total		3800
Shutdown shield		2000
Support structures		1300
Total System Weight		9310

Shadow shield weights for different separation distances are given in Figure 10 for a reactor thermal power level of 165 kw. For cylindrical stations of different diameters, the shielding data in Section VIII may be used to estimate the weight of the required shadow shields.

C. OPERATIONAL PROCEDURES

1. Launch Configuration and Deployment

The nuclear power unit can be launched with the space station as depicted in Figure 11. In order to minimize structural weight the power plant should be launched in a compact configuration. After orbit is achieved the power unit is separated from the spacecraft by a telescoping aluminum boom using tension cables for stability. The boom length has to be determined by evaluating increased structural weight vs decreased shield weights as the reactor-payload distance increases. In addition, the effect on the space station dynamics of this separation distance and of the variation of the weights involved must be considered.

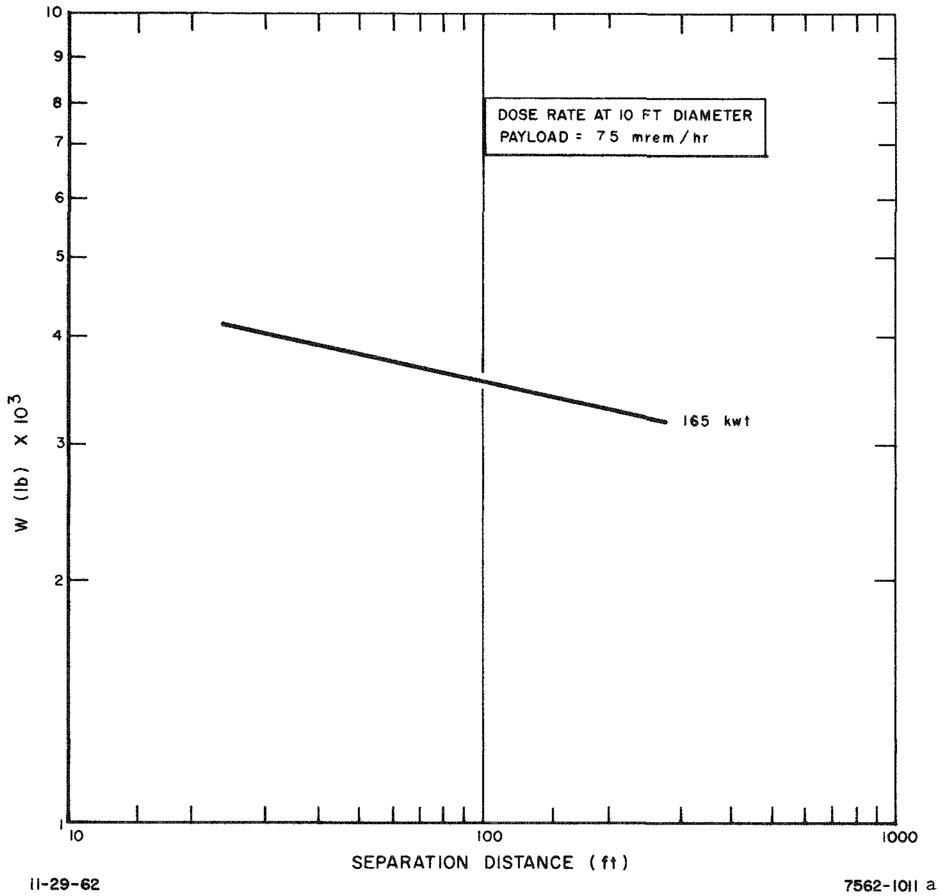

Figure 10. Dependence of Shield Weight on Separation Distance

If the nuclear power plant is to be installed on a space station established in orbit, the first power plant package to be launched will consist of the reactor-power conversion assembly, the biological shield assembly and the supporting and extension structure in the retracted position.

2. Initial Installation

The power package, ready for mating, will be in front of the shuttle vehicle as that vehicle approachs the station. The mating ring will be at the base of the supporting structure and oriented to mate with the corresponding structures on the station. The overall alignment can be made by using auxiliary bumpers and guides.

After mating and alignment, the structures are locked together and the connections are made. If the electrical connectors are incorporated as part of the mating ring, an automatic connection is made. The next operation extends

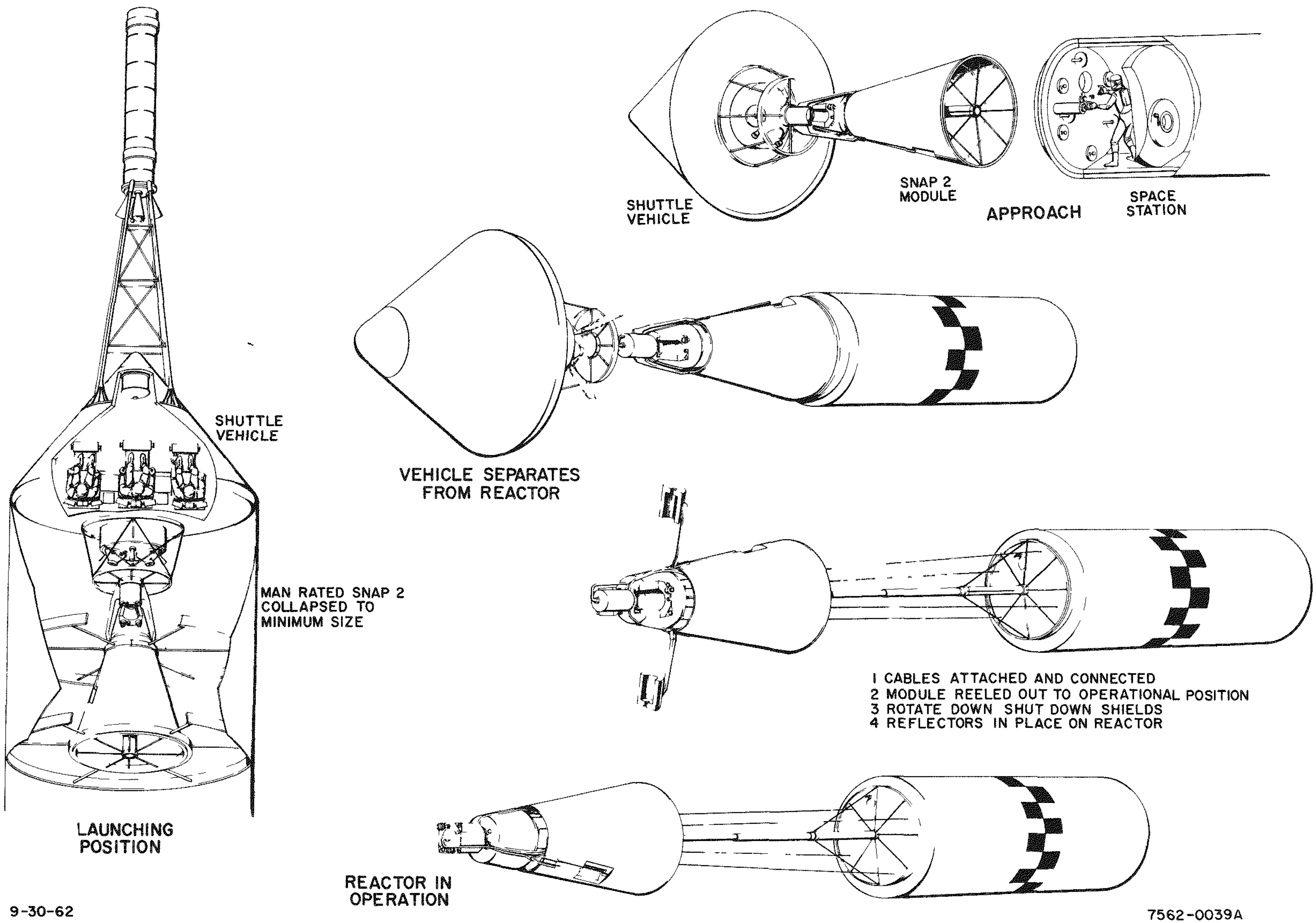

Figure 11. Launch and Installation of Power Plant

the boom to its locked position and rigs the cables to the proper tension. To provide the proper tension, a measured length of cable may be used or the cables may be adjusted from inside the vehicle.

The design criteria of the SNAP systems require maintenance-free operation for one year. Available manpower may be used, however, to increase reliability and safety. Also, certain installation procedures will require use of personnel. It is desirable to operate from inside the space station; therefore, a special compartment, sealed and isolated from the rest of the station and with entrance through an air lock, should be provided. A series of penetrations in the wall facing the reactor will permit the use of special lightweight tools. A man clothed in a pressure suit can operate from inside this compartment and perform all the necessary operations within the doubly protective environment of the suit and the compartment.

To assist in the rendezvous and final mating of the power package to the space station a series of special installation tools to be used to guide, position, index and pin down the two bodies can be used through the wall penetrations. These will considerably ease the mating situation.

Power package replacement equipment, including destruct charges, gas tanks and guidance equipment, may be stowed inside the station away from the heat and vacuum environment. When it is time to exchange the power package, they can be installed in their proper places from inside the compartment by means of telescoping probes through the penetration in the wall.

3. Power Plant Replacement

After the nuclear power plant completes its useful life, about 1 yr, the complete power unit may be released from the supporting structure and shield and separated from the manned spacecraft by small cold-gas rockets attached to the bottom of the radiator cone, Figure 12. This release will allow the massive shield and supporting structure, which should have a useful life of 5 to 10 yr, to remain with the space station to be reutilized for another power plant which can then be installed by a rendezvous shuttle vehicle. Only mechanical and electrical connections will be necessary since the fluid systems of the power plant are completely contained within the detachable unit. In the case of irreparable damage which would prevent its proper operation, the replacement of the system may also be required. The design presented herein provides

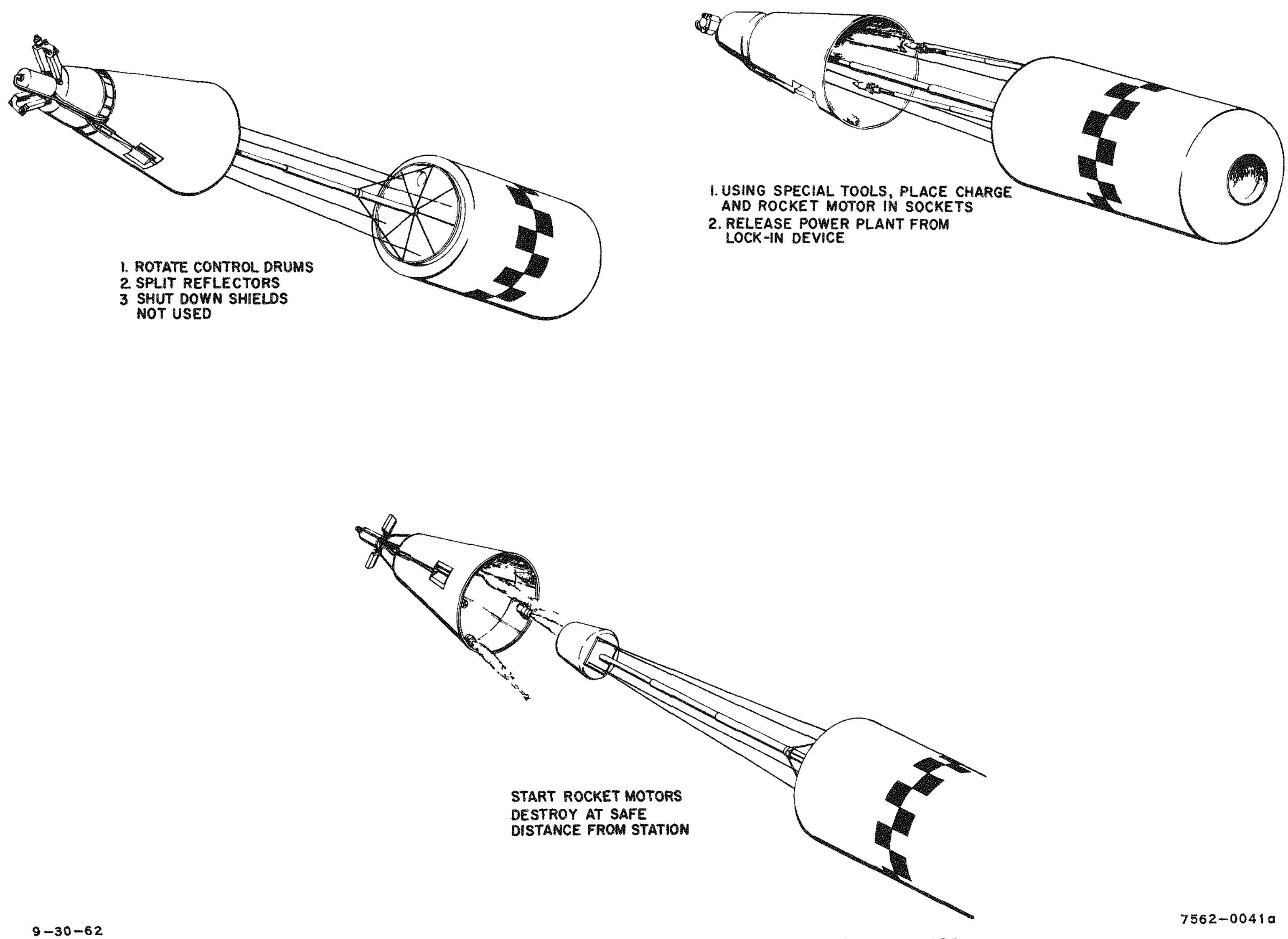

Figure 12. Disposal of Nuclear Power Plant

for the separation and destruction of only the reactor-power conversion system (pcs), leaving the biological shield, shutdown shield, and allied structure attached to the space station.

This separation and destruction is accomplished in the following steps:

a. Scram reactor and split the reflectors away from the core vessel.

b. Disconnect all electrical and mechanical connections.

c. Release and verify all the locks holding the reactor-PCS assembly to the shield.

d. Activate the jets provided at the rear end of the radiators, at which time the radiator-PCS and reactor will be propelled away.

e. Destroy the assembly, after it is at a safe distance (it is destroyed by means of an armour piercing shell which is located at the base of the radiators). Redundancy of systems may be used to increase reliability.

f. Bring up a new reactor-PCS assembly by the shuttle vehicle and align it in place during rendezvous.

g. Guide assembly into place, and lock all the connectors.

The reactor is ready for startup at completion of the final step. For space stations launched into long-lived orbits, the radioactivity in the core will decay to negligible levels before reentry. The first generation of manned stations will probably be in low altitude orbits, and if so, it is desirable for the radioactivity to be dispersed in the upper atmosphere on reentry. Work presently underway by Atomics International (AI), the AEC, the United States Air Force (USAF), and others is intended to determine the requirements for reentry burnup. Burnup of an intact SNAP core is marginal but burnup can be assured by disassembly of the core with an explosive charge similar to the destruct charge utilized for the present USAF-AEC SNAP Flight Test Program. The charge would be stored in an area within the shadow cone of the shield to avoid radiation and heat degradation of the explosives. The activity in a SNAP 2 core is about a factor of 10^{-8} less than that released in a 20 KT nuclear weapon.

4. Rendezvous Operation

Although the shadow shield adequately protects the personnel in the space station, many systems require a rendezvous with shuttle or supply vehicles. Necessity for such a rendezvous is an essential factor in the concept of a 5-yr space station with astronaut duty periods of approximately one-month. Resupply of food, water, fuel and equipment, as well as personnel, is contemplated. Figure 4 illustrated the radiation levels that would be encountered outside the shielded shadow cone from the bare operating reactor at 100 kwt. At 1000 ft only six min would be required to receive a 5 rem dose. Therefore, if rendezvous is to be achieved, the shuttle vehicle must duck into the shadow shielded cone while at about 1000 ft from the space station and must remain in the cone as approach is made to the docking structure. The feasibility of this type of rendezvous has not been investigated but may be too restrictive to present an attractive mode of operation. For this reason a deployable shutdown shield is used which results in the dose rate distribution which was shown in Figure 6. Since the dose rate is only about 1 rem/hr at 30 ft from the reactor, this shield permits the shuttle vehicle to approach from any direction. The rendezvous operation is depicted in Figure 13. The reactor is shutdown 1 hr before the anticipated rendezvous. The reflectors are then split away from the core and the shutdown shield halves rotated and brought up to enclose the reactor core. The shuttle vehicle can then approach the space station and complete its docking procedure.

In the case of a failure of either of the required operations, splitting the reflector halves or deploying the shutdown shield, manual overrides are provided by a system of cables directly connected to the mating ring on the outer surface of the station, and these may be operated by a hand crank from inside the compartment.

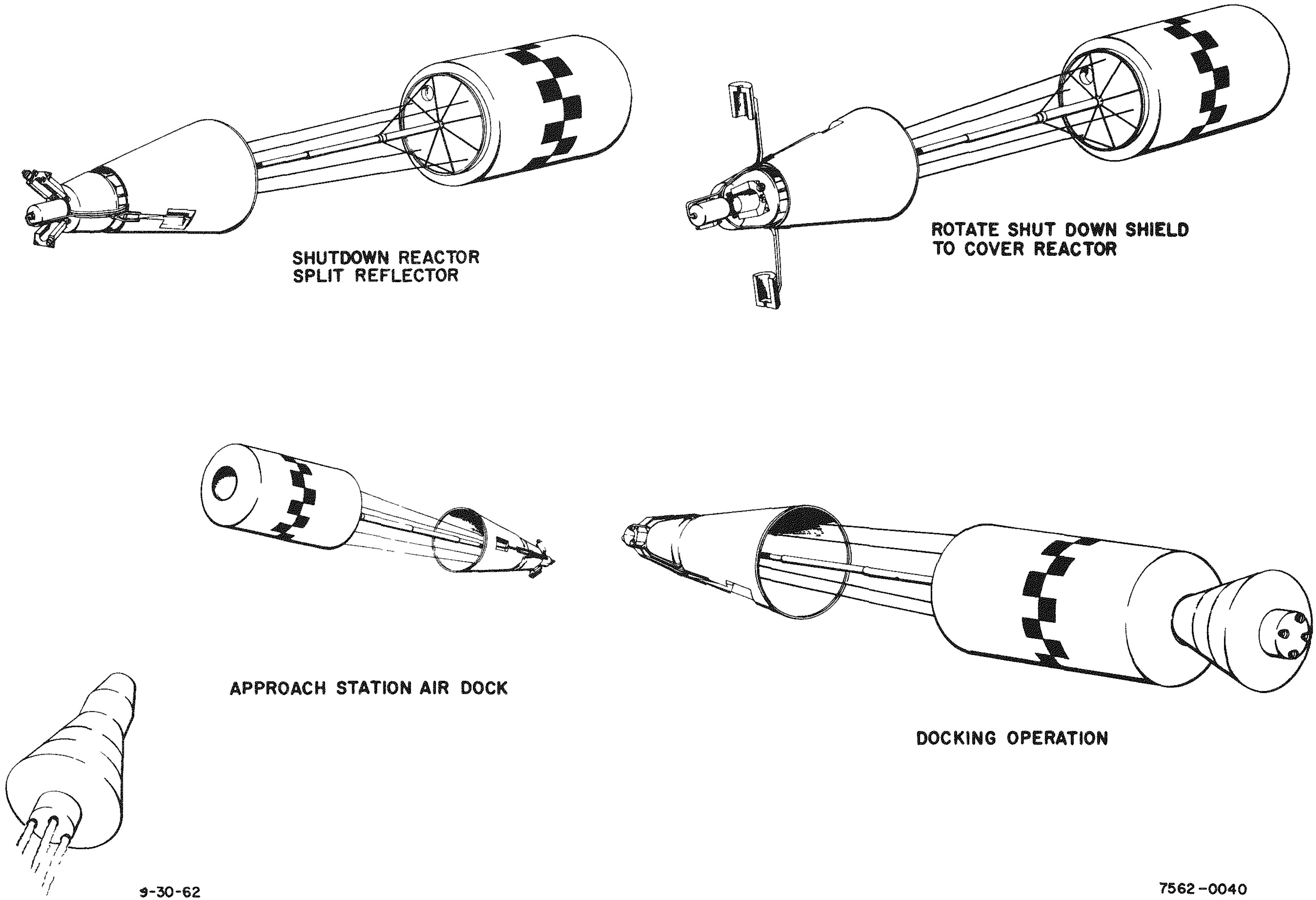

Figure 13. Rendezvous Operation

V. THE TOROIDAL SPACE STATION INSTALLATION

A. DESCRIPTION

One space station configuration that has received a great deal of attention is the toroidal or "doughnut" type space station illustrated in Figure 14. This particular system is patterned after a design developed by the Space and Information Systems Division (SISD) of NAA and described in SID-62-658-1 "Self Deploying Space Station Interim Report." The design is for a system housing a crew of ~20 men, and the electrical power requirements are estimated at about 11 kwe. The space station would be composed of five cylindrical 10 ft diam sections arranged in the shape of a nearly circular hexagon. This volume houses the occupied sections of the station. Three tubular sections provide access from the main portion of the station to a hub section located in the center of the hexagon. The space station is rotated about an axis perpendicular to the plane containing the large cylindrical sections and going through the central hub to provide an artificial gravity at the outer rim of the station. The hub section may be disengaged from the spinning motion to permit rendezvous to occur at this point. When this rendezvous operation has been completed the hub section is accelerated to the same rpm and access to the outer rim is made via the access tubes leading from the hub to the outer rim. The diameter of the space station is ~150 ft.

A layout of the 11 kwe nuclear power system described in Section VIII is shown in Figure 15. For this example the reactor-shield and power conversion system are attached to the central hub extended ~20 ft from the closest access of the hub. Since limited access only is usually required for the hub area (such as zero-g experiments, docking, embarking and disembarking) the design limiting condition for this example then becomes the selected value of the operating dose rate at the rim of the space station. A description of the major power plant components follows.

1. Main Shield

The main shield is approximately a hemisphere surrounding the SNAP 2 reactor and shadowing the extreme edges of the reactor from the rim of the space station. The shield is a composite of 2.36 in. depleted uranium, 15.8 in. borated zirconium hydride, and 16.3 in. lithium hydride. The shield is canned in 1/8 in. thick 316 stainless steel.

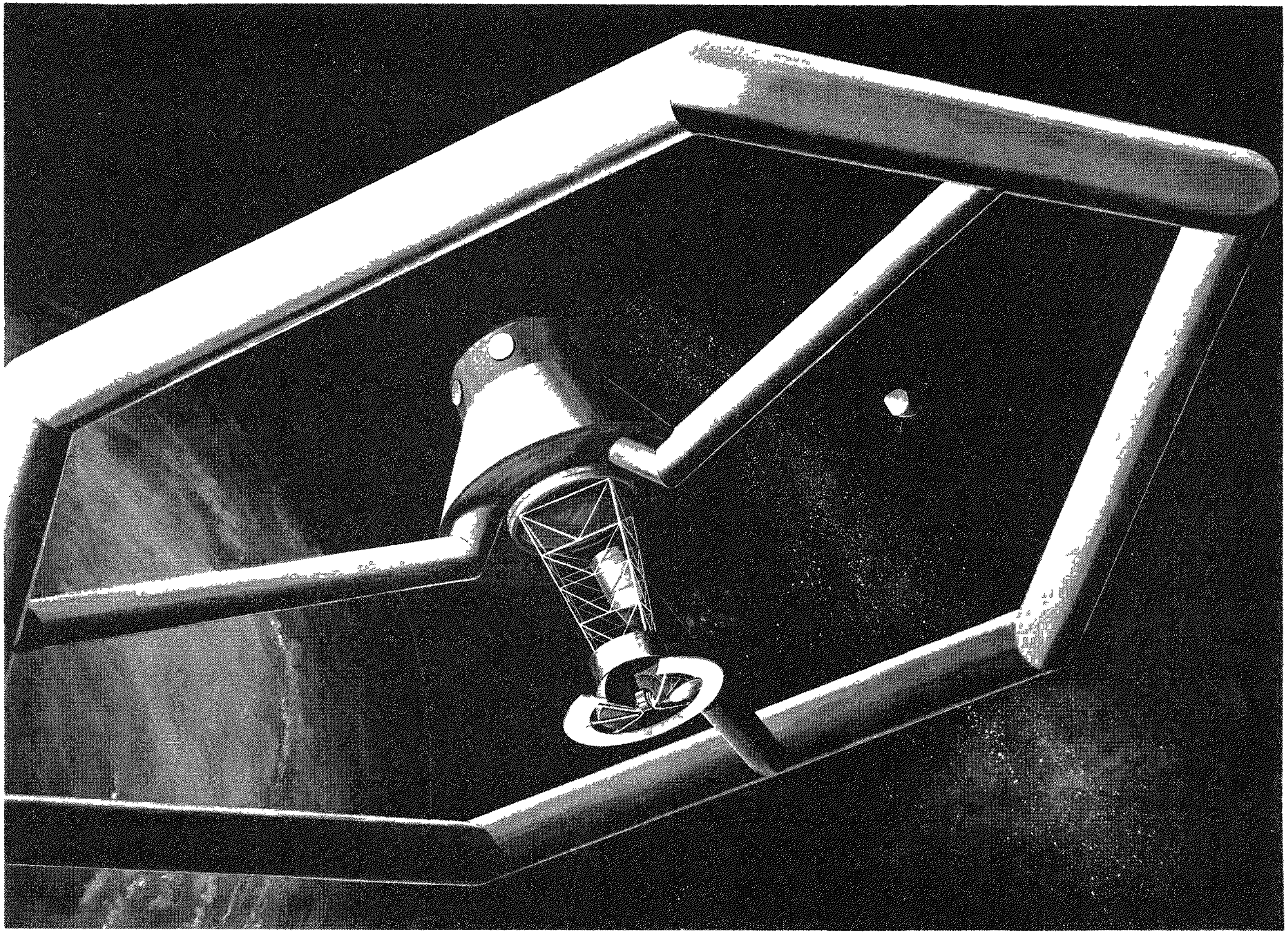

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Figure 14. Toroidal Station Layout, Artist's Concept

SNAP 2 SPACE PLATFORM POWER PLANT

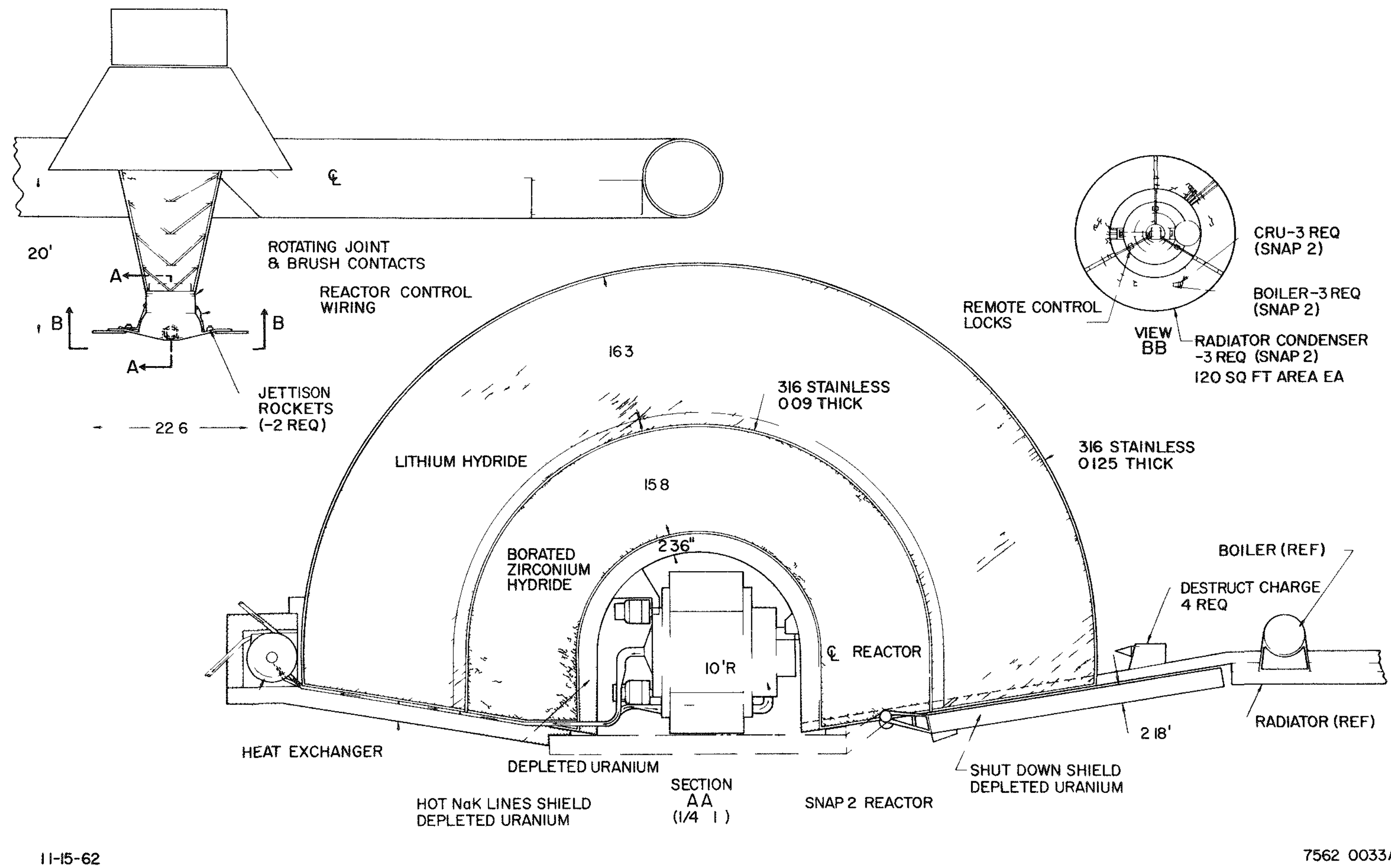

Figure 15. Shielding for Toroidal Station

2. Shutdown Shield

The shutdown shield is a slab of 2.16 in. thick depleted uranium which is rotated into position around the shutdown reactor. During reactor operation neutron activation is minimized by assuring that the shutdown shield is not exposed to the direct beam of the reactor.

3. Radiators

The radiators are pie-shaped rather than conical segments. This shape allows both sides of the radiator to "see" space, thus, minimizing the required radiator area.

4. Support Structures

The power plant system is separated from the hub by a 20 ft high tower structure. This structure need not be of great length since the personnel are usually in the rim area which is 75 ft from the hub.

Figure 16 depicts the dose rate distribution around the space station during and 1 hr after reactor shutdown.

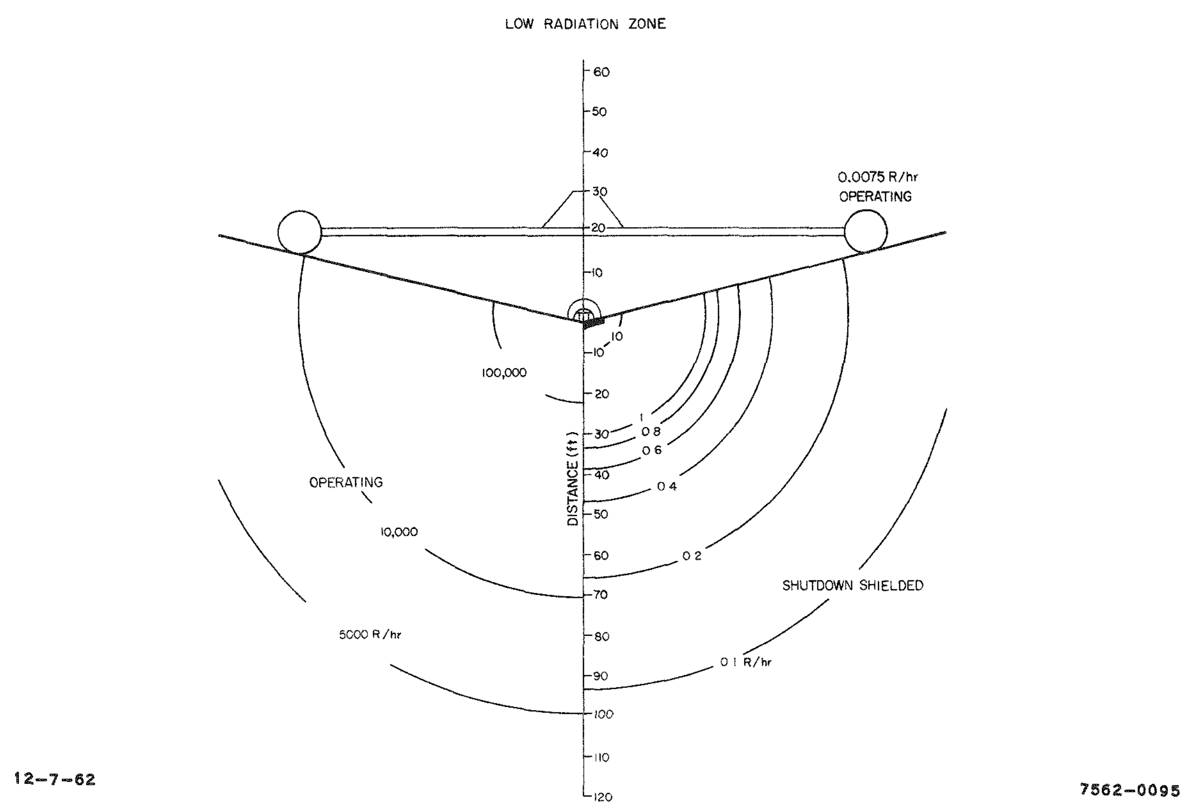

Figure 16. Dose Rate Map for Toroidal Station

B. WEIGHTS

Table II gives the weight breakdown for the power system installed in the toroidal configuration.

TABLE II

WEIGHT SUMMARY FOR TOROIDAL SPACE STATION (lb)

Reactor System	320	
Secondary NaK Systems (3)	150	
Power Conversion Systems (3)	1080	
Radiator-Condensers (3)	660	
Total Unshielded Weight		2210
Uranium, depleted	2440	
Zirconium Hydride	11540	
Lithium Hydride	4280	
Stainless Steel Containers	700	
Biological Shadow Shield, Total		18960
Shutdown Shield		1400
Support Structures		2000
Total System Weight		24570 lb

The weight of biological shadow shield for other separation distances, i.e., space station diameters is shown in Figure 17.

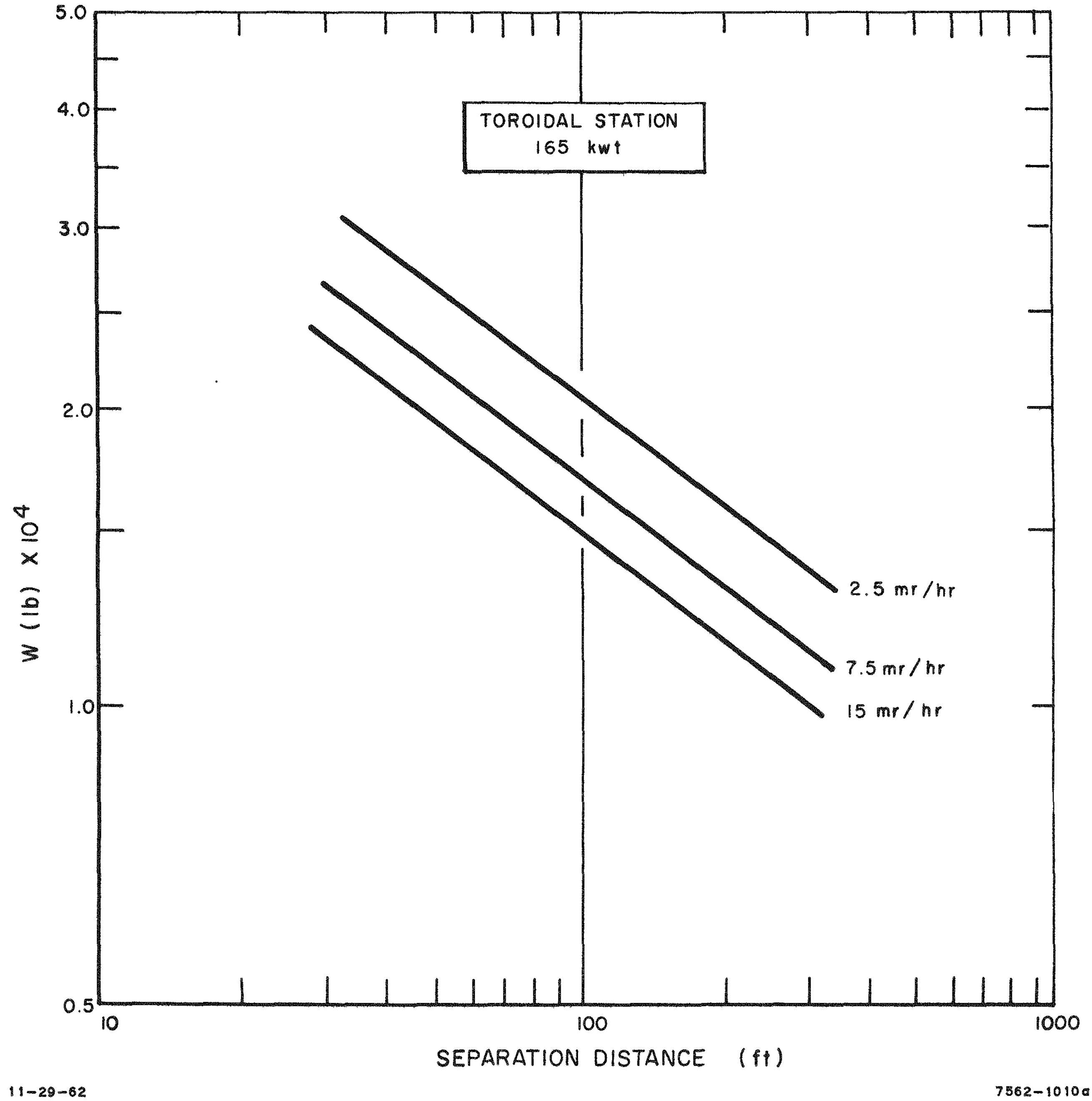

Figure 17. Toroidal Station Shielding Weight

C. OPERATIONAL PROCEDURES

1. Launch Configuration

The nuclear power unit can be launched with the toroidal space station or mated with the station in orbit following a rendezvous. The mating plane is at the base of a 20-ft tower structure which separates the reactor from the hub of the station. The nuclear power plant can be launched with a manned shuttle vehicle as shown in Figure 11.

2. Initial Installation

The shuttle vehicle will approach the toroidal station from below the hub with the mating ring oriented to mate with the corresponding structures on the toroidal station. After mating and aligning the structures are locked together and the electrical and control connections are made.

The possible use of a special sealed compartment, provided with tool penetrations to allow maintenance of the power plant, is discussed in Section IV. This concept is equally applicable to the toroidal station. A man operating from inside a compartment below the hub could assist in the docking and final mating, install accessory equipment on the power plant, and perform certain maintenance functions.

3. Power Plant Replacement

The power plant replacement routine used for the toroidal concept is basically the same as that used for the cylindrical concept described in Section IV. However, the shutdown shield for the toroidal station is in the form of a heavy plate placed below the biological shield rather than the clam shell arrangement used for the shutdown shield of the cylindrical station. The plate is deployed by sliding it into position below the reactor following reactor shutdown for rendezvous. Disposal and destruction procedures are identical with the toroidal and cylindrical concepts. As described in Section IV, the disposal of the reactor at the end of its useful life is accomplished by separation of the nuclear power unit (without shield) using cold-gas rockets and the reactor is destroyed with explosives. Because of the large fraction of the system weight represented by the shield, the capability for reuse of the shield with replacement power units is particularly important in the toroidal concept.

4. Rendezvous Operation

Although the biological shield adequately protects the personnel in the space station, rendezvous with shuttle vehicles will be required and personnel in the vehicle may be exposed to excessive radiation fields. If the shuttle vehicle approaches from "above" the space station it will be adequately shielded. If, however, the vehicle overshoots (routinely or by error), the reactor must be shutdown and, possibly, a shutdown shield deployed. The radiation levels around the space station during reactor operation and following the reactor shutdown with shutdown shield in place, are shown in Figure 16. The dose rate with the shutdown shielded condition is only 1 r/hr at 30 ft from the reactor (outside the biological shield shadow). Thus essentially complete freedom for the rendezvous vehicle is allowed. If the reactor is operated continuously during rendezvous, however, the dose rate is 50 r/hr at as much as 1000 ft and very little time can be spent outside of the biological shield shadow.

VI. NUCLEAR SAFETY CONSIDERATIONS

A. LAUNCH OPERATIONS

The reactor is not operated at any time before or during launch. It therefore represents a negligible radiation source during these periods. Special locks and filler blocks are attached to the reflector control assembly during handling operations to insure that the reactor cannot be brought to criticality and thus generate power. These provisions are removed just before launch but could be removed in orbit for certain manned vehicles.

To bring the reactor to criticality, a squib-actuated pin must be removed and control drums driven in.

The reactor can go critical if immersed in water while fully intact. In the case of a launch abort, the manned craft would be pulled away from the booster by a rocket escape tower. The reactor system would be jettisoned from the manned craft and disassembled by a destruct charge to assure that criticality could not be achieved on immersion in water.

B. ORBITAL OPERATIONS

1. Reactor Operation

The reactor is started after orbit is reached and deployment completed. Shielding has been provided (as discussed elsewhere) to limit crew exposure to 5 rem/month.

Operating experience on the two SNAP 2 ground test reactors — SNAP 2, Experimental Reactor (S2ER) and SNAP 2, Developmental System (S2DS) — has been excellent. In a period of over three years, no major or minor nuclear reactor accident has occurred. Operations experience in space should be equally good. The presence of men to monitor the reactor operations should assure safety. Automatic fast scram systems are, nonetheless, routinely used in ground operation and would probably be equally as useful in space operations to supplement the human operators and relieve them of the requirement for close attendance to the reactor instrumentation.

2. Rendezvous

The rendezvous of a shuttle vehicle with the space station has already been discussed in some detail with respect to the shielding requirements. With adequate shielding the risk of overexposure to radiation is small. The design value for the shutdown shield has been taken as 1 rem/hr at 30 ft. This would allow over 25 hr at that distance before the acceptable emergency dose for radiation would be received. Even if the shutdown shield were unable to be brought up around the reactor, due to some malfunction, about 10 min would be required at 30 ft to receive the emergency dose.

3. Release of Fission Products

For ground operating conditions, the major hazard of nuclear reactors is the release of radioactive fission products into the atmosphere. An interesting consequence of a zero-gravity, high-vacuum environment is that fission product release poses no major danger. This is due to the fact that the fission products released are hot and volatile and therefore quickly dispersed in the absence of atmosphere and gravitational influence. The astronauts would not inhale, ingest, or otherwise be intimately exposed to the radioactive fission products.

C. REACTOR DISPOSAL

After completion of the useful life of the nuclear power unit, the unit will be detached and jettisoned as described previously. For orbits with lifetimes greater than 300 yrs, radioactive decay causes negligible radiation to be present on reentry.

1. Reentry Burnup

For short-lived orbits, the radioactivity must be dispersed in the upper atmosphere. Studies are presently underway to determine if complete reentry burnup occurs for a SNAP 2 reactor. To assure burnup, a destruct charge may be utilized to break the reactor into small pieces.

2. Atmospheric Contamination

The contribution of SNAP reactors to the worldwide atmospheric contamination is quite small as shown in Figure 18. If a 50-kw SNAP 2 reactor is burned up in the atmosphere at the rate of 1/yr continuously until 2020 A.D., and no more nuclear bomb tests are carried out, the Sr^{90} level due to weapons tests will still exceed the reactor induced level by a factor of almost 100.

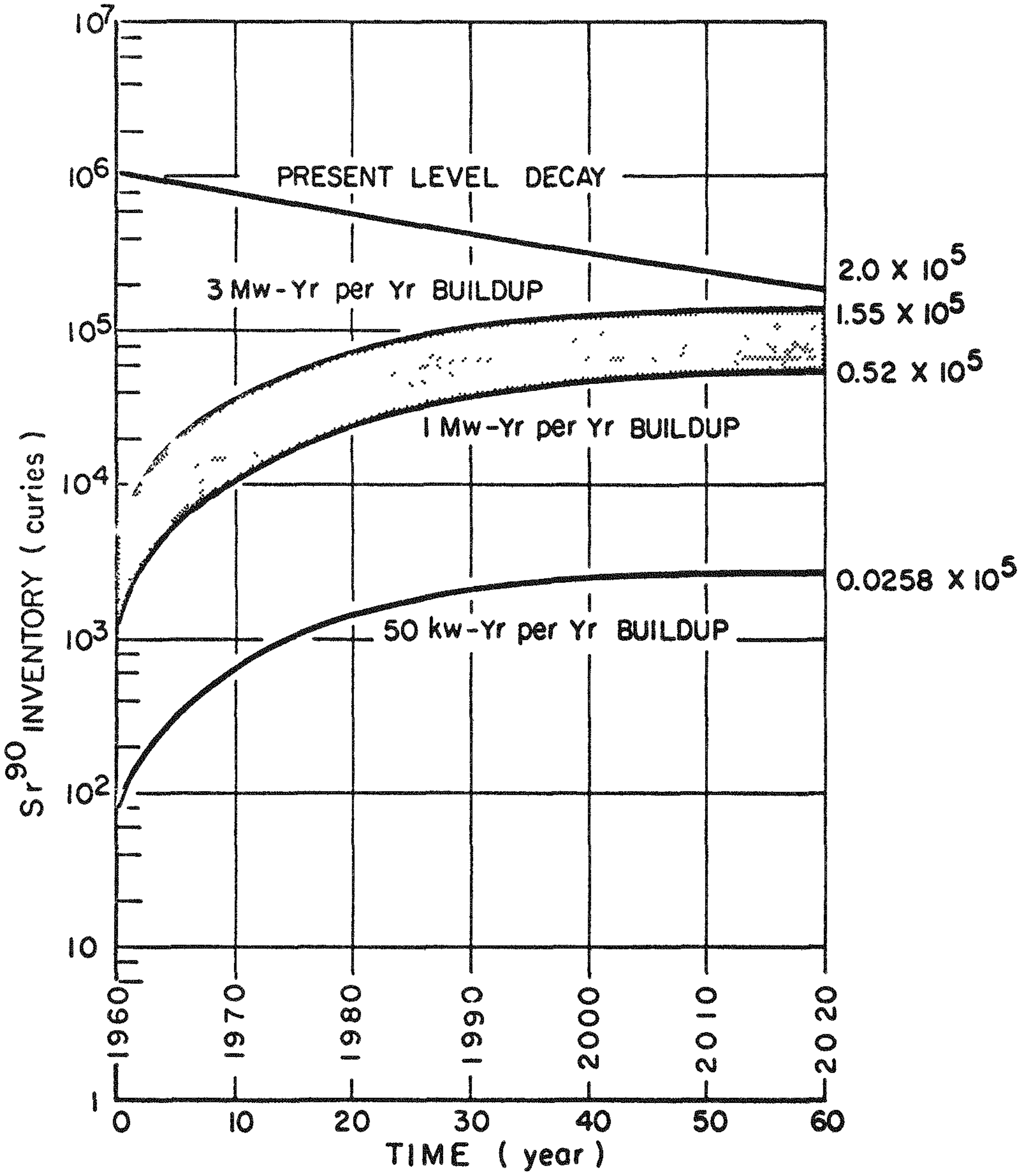

Figure 18. Comparison of Sr^{90} Inventory in Upper Atmosphere From Past Nuclear Tests and Possible Space Programs

VII. THE 11-kwe MULTIPLE SNAP 2 UNIT

The Advanced SNAP 2 11-kwe Power System consists of three 4-kwe (gross) power conversion systems coupled to a single reactor heat source. The reactor is an upgraded version of the SNAP 2 reactor and the power conversion systems utilize the components of the 3-kwe SNAP 2 flight test unit. The SNAP 2 system and the various components are described in Section IX with a brief summary of their development status.

A. SYSTEM DESCRIPTION

A flow schematic of the system is shown in Figure 19. The reactor core is cooled by liquid metal NaK which enters the core at 1040°F and exits at 1240°F. The NaK is then pumped to three separate paralleled NaK to NaK heat exchangers where the reactor heat is transferred to the three independent mercury powered conversion loops. The NaK from the primary heat exchanger is conducted to a mercury boiler where liquid mercury is vaporized and superheated. This superheated mercury vapor is then expanded through a miniature high speed turbine which in turn drives an alternator that delivers 4.0 kwe of gross electrical power at 110 v and 1800 cps. The turbine, alternator, and centrifugal mercury boiler feedpump are located on a single shaft unit called the Combined Rotating Unit (CRU). The expanded mercury vapor is conducted to a direct radiator condenser where the mercury vapor is liquified and returned to the boiler. A portion of the liquid mercury being returned to the boiler is subcooled to provide a low temperature liquid stream for alternator cooling and bearing lubrication. The primary and secondary NaK loop pumps are the canned rotor types and are electrically driven by a portion of the gross alternator output.

Table III lists some of the pertinent system characteristics.

With the exception of the radiator-condenser, the component sizes, dimensions, and shapes are considered a fixed geometry. The radiator shape is allowed to assume geometries dictated by the particular installation configuration with a fixed area required.

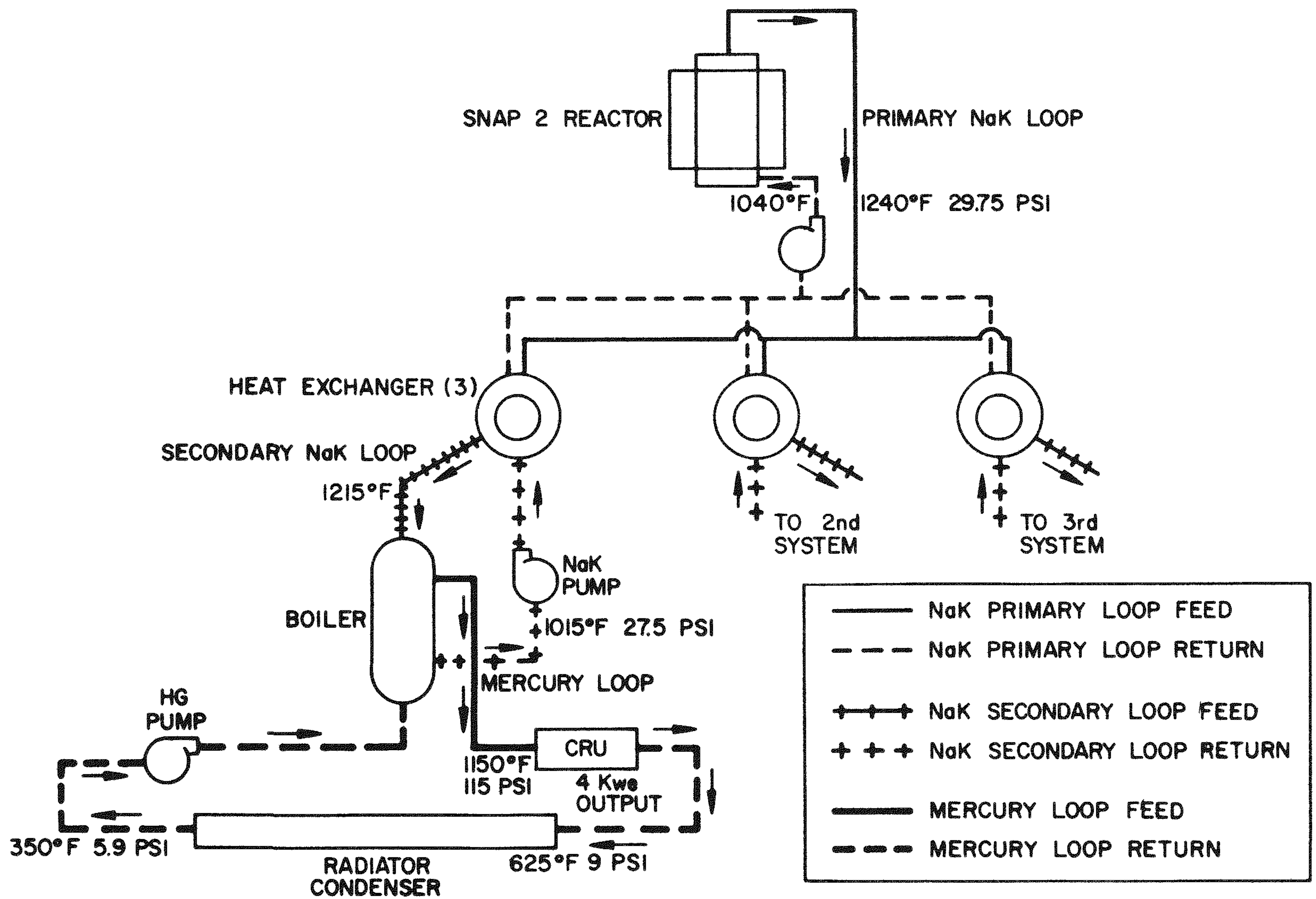

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Figure 19. Advanced SNAP 2 Coolant Flow Diagram

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TABLE III

ADVANCED SNAP 2 POWER SYSTEM

Electrical Power	
3 CRU (net)	11 kwe
1 CRU (gross)	4 kwe
Type	3-phase
Frequency	1800 cps
Voltage	110 ± 5.5 v
Reactor System	
Thermal power	163 kw
Flow (NaK)	216 lb/min
Pressure drop	6 psi
Outlet temperature	1240°F
Inlet temperature	1040°F
Power Conversion System (per CRU)	
Thermal power	50 kw
Turbine inlet temperature	1150°F
Turbine inlet pressure	115 psia
Turbine exhaust pressure	9 psia
Mercury flow rate	20 lb/min
Total mechanical efficiency	37%
Turbine efficiency	53%
Alternator efficiency	84%
Rankine cycle efficiency	8%
Mechanical power-shaft	5.6 kw
Radiator temperature inlet	600°F
Radiator temperature outlet	350°F
Total radiator area	350 ft^2
Radiator pressure drop	3 psi

B. WEIGHTS

A weight summary of this system is presented in Table IV.

TABLE IV

11-kwe SNAP 2 POWER PLANT WEIGHT SUMMARY
(lb)

Reactor System	320
Secondary NaK systems (3)	150
Power conversion systems (3)	1080
Radiator-condensers (3)	660
Total power plant	2210

C. STARTUP AND RESTART

1. Startup

The power system is started after the required orbit is obtained and verified. The NaK pumps (for each loop) are, however, started on the ground (battery powered) to provide 5% NaK flow in order to prevent any NaK from freezing before system startup.

The startup sequence is begun (see Figure 20) by rotating in the reactor safety drums. Simultaneously, the reactor control drums slowly begin stepping inward in response to the low temperature signal from the NaK temperature sensor at the secondary heat exchanger outlet. As the reactor temperature increases, each of the NaK loops and boilers is preheated. When the secondary heat exchanger outlet temperature reaches 1000°F, the Power Conversion System (PCS) startup sequence commences by opening the valves from the injection tank and regulator tank and closing the condenser exit valve simultaneously for all three mercury loops. This sequence causes the injection of mercury into the boiler and bearings. The mercury is vaporized in the boiler and spins the turbine, the shaft of which provides the mount for the mercury pump and alternator. As mercury condenses and collects in the radiator condenser, the pressure and temperature build up to design conditions. The radiator-condenser exit valve is then opened and the excess mercury flows into the regulator tank. At the completion of this step, the system reaches steady-state conditions.

2. Restart

When the system is shutdown the regulator tank is valved off to trap the mercury (130 lb/CRU of the total of 150 lb/CRU is required to start the system). The remainder of the system is then evacuated to space releasing 20 lb of mercury. (See Figure 20.)

To restart, the injection tank is filled with mercury from the regulator tank and the sequence is carried on as previously described.

Twenty pounds of excess mercury per CRU must be provided for each restart in addition to the basic 150 lb/CRU.

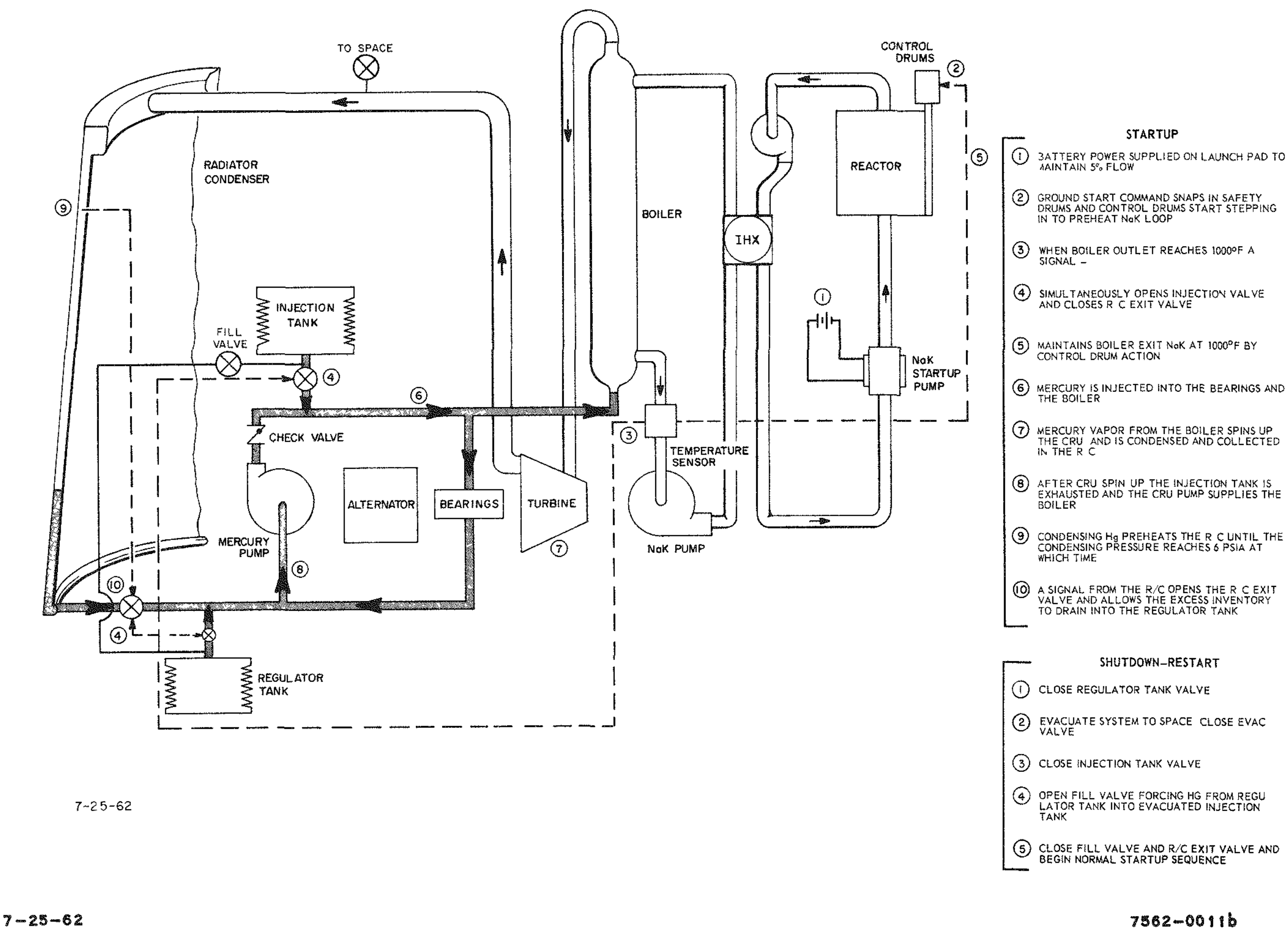

Figure 20. SNAP 2 Orbital Startup and Restart

VIII. SHIELDING

A. SHIELD DESIGN CRITERIA

1. Permissible Radiation Level

While no permissible radiation dose for manned space flight has yet been fixed by any government agency, it is probably conservative to assume that a total integrated dose of 5 rem for a "routine" space flight is permissible, with 25 rem possible under some conditions. In the present study, the various reactor shields have been designed so that the integrated dose in the space station over a period of 4 wk is 5 rem.

2. Shield Configuration

Shields for two different space station designs were analyzed. The first space station design is cylindrical in shape. The reactor is placed along the axis of the cylinder. The shield for this design is a shadow shield in the form of a frustum of a cone. This configuration presents the most favorable geometry for a lightweight reactor shield.

The second space station design is toroidal shaped, and the reactor has been placed at the center of the space station. In this case, the shield weight optimization has been based on the concept of a shield completely surrounding the reactor, although it should be noted that the shield used does not completely surround the reactor, as shown in Figure 15 and the weights are adjusted accordingly.

B. SOURCES OF RADIATION

One of the first steps in designing a shield is the determination of the intensity, energy distribution, and spatial distribution of radiation sources. If the primary source of radiation is a reactor, only neutrons and gamma rays are considered, since these are the most penetrating forms of radiation.[1]

1. During Reactor Operation

The sources of radiation during reactor operation are given in Table V and discussed below. Figure 21 shows the radiation levels with the reactor operating and also one hour after shutdown with a shutdown shield in place.

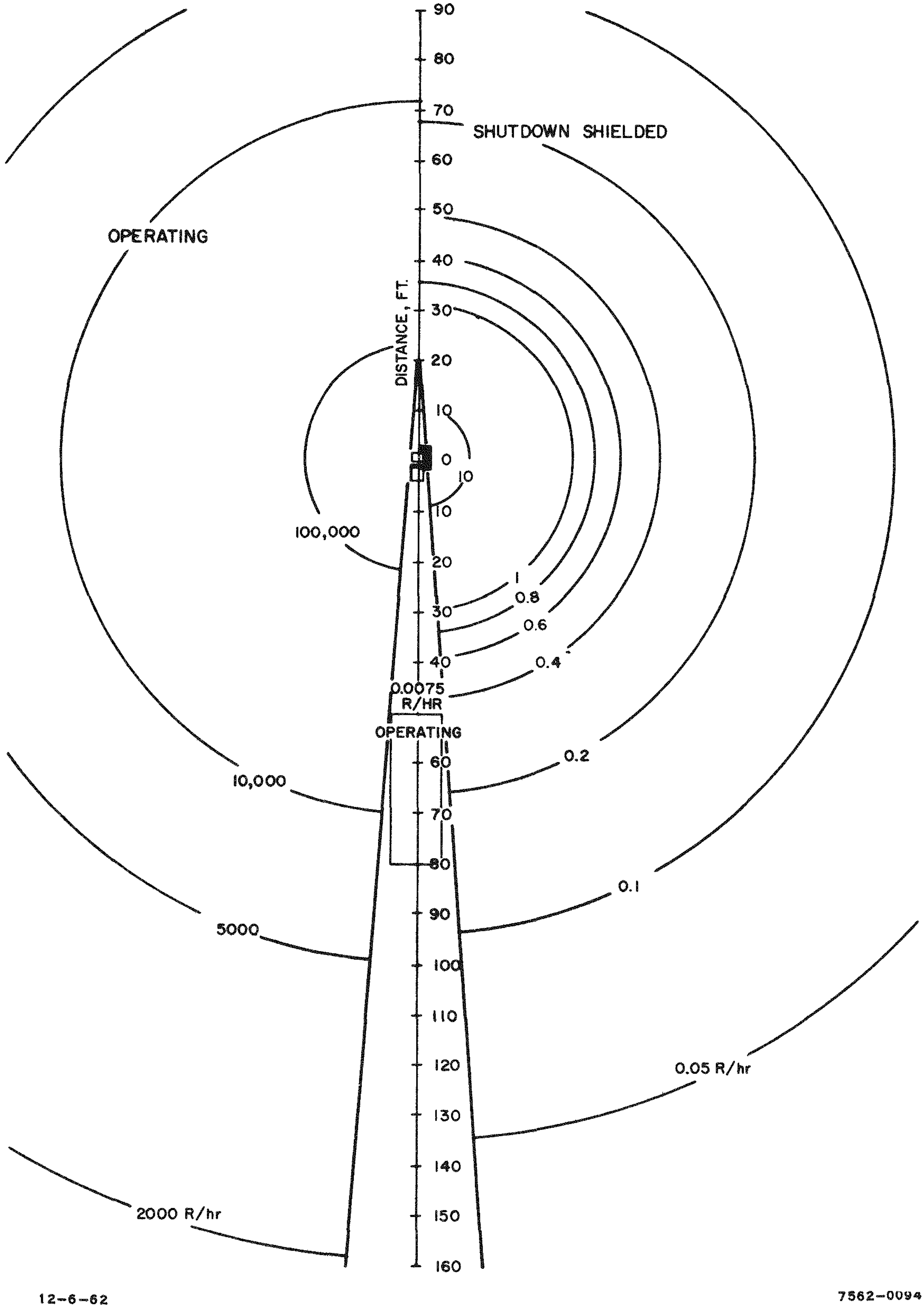

Figure 21. Dose Rate Map for Cylindrical Station

TABLE V

SOURCES OF RADIATION DURING REACTOR OPERATION

Where Produced	Gamma Rays	Neutrons
Reactor core	Prompt-fission gammas	Fission neutrons
	Fission-product gammas	Delayed neutrons
	Neutron-capture gammas	Photoneutrons
	Gammas from neutron inelastic scattering	
	Decay gammas of neutron activated materials in core	
Coolant, structural materials and reactor shield	Neutron-capture gammas	Photoneutrons
	Gammas from neutron inelastic scattering	
	Coolant-activation gammas	
	Structure-activation gammas	

TABLE VI

GAMMA RAY ENERGY SPECTRA RESULTING FROM FISSION OF URANIUM-235 (Mev per Fission)

Average Gamma Ray Energy	Prompt Fission Gamma Ray	Equilibrium Fission Product Gamma Ray	Capture Gamma Ray* in U^{235} in Thermal Reaction	Total
1	3.450	5.16	0.520	9.130
2	2.360	1.737	0.356	4.453
3	1.175	0.322	0.177	1.674
4	0.477		0.072	0.549
5	0.203		0.031	0.234
6	0.136		0.020	0.156
7	0.026		0.004	0.030
Total	7.827	7.219	1.18	16.226

*Capture gamma-ray spectrum is assumed to have the same spectral distribution as that of the prompt fission gamma rays.

a. Gamma Rays

In the computation of the core gamma-ray strength, the most important contributions are the prompt-fission gammas, the fission-product gammas, and the neutron-capture gammas. The gamma-ray contribution resulting from neutron inelastic scattering and from decay of neutron-activated materials in the core is comparatively small; thus it may be neglected in the consideration unless great accuracy is desired. The gamma-ray energy spectra resulting from fission of U^{235} is shown in Table VI. The neutron-capture gamma rays from other materials in the core should be added to the spectrum given in the table. The total gamma-ray source strength is computed by the following equation,

$$S_i = PKU_i + V \sum_j C_{ij} N_j \int_o^{\infty} \varphi(E) \sigma_j(E) dE \quad , \qquad \ldots (1)$$

where

S_i = source strength of core at energy i, in Mev/sec

P = thermal power of reactor, in watts

K = constant, 3.3×10^{10} fissions/sec-watt

U_i = total gamma-ray energy spectra of uranium at energy i, in Mev/fission (see Table VI)

V = volume of core, in cm^3

C_{ij} = gamma ray energy given off at energy i per neutron capture in element[2] j, in Mev

N_j = number of atoms per unit volume of core of element j, in $atoms/cm^3$

$\varphi(E)$ = neutron flux per unit energy E in core, in $neutrons/cm^2$-sec-Mev

$\sigma_j(E)$ = absorption cross section of element j at energy E, in cm^2/atom of element j.

In the above equation, the first term is the contribution from U^{235} and represents a source strength of 16.2 Mev/fission. The second term is the contribution of neutron-capture gamma rays from materials other than U^{235} in the core. This may amount to as much as 2.5 Mev/fission. However, for SNAP reactors, capture gamma rays contribute approximately 0.5 Mev/fission, and these have been added to the fission source with the spectrum given in Table VI. The calculations are thus considerably simplified without introducing large errors in the results.

The gamma rays outside the core result mainly from neutron-capture and from neutron inelastic scattering with shield materials. The secondary gamma-ray source strengths are calculated identically to those given in the second term of Equation 1.

Certain nuclei become radioactive upon neutron absorption and emit gamma radiation. This source of radiation may sometimes be important, as in the case of reactor coolants or structure, thus presenting an additional source requiring shielding. The source strength of activated materials may be determined using the following equation,

$$S_i(E_j) = \left[\frac{N\rho_i f_i E_{ij} n_{ij}}{A_i} \int_E \varphi(E)\sigma_i(E)dE\right] \frac{\left(1 - e^{-\lambda_i t_r}\right)\left(1 - e^{-n\lambda_i t_c}\right) e^{-\lambda_i T}}{\left(1 - e^{-\lambda_i t_c}\right)}, \qquad \ldots(2)$$

where

$S_i(E_j)$ = source strength of radioactive isotope i per unit volume of activated material emitting gamma rays of energy E_j, in Mev/cm^3-sec

N = Avogadro's number, 6.025×10^{23} atoms/mole

ρ = density of material i, in gm/cm^3

f_i = weight fraction of parent isotope of radioactive isotope i in material. This is equal to the product of the weight fraction of element i and the isotopic abundance of the parent isotope of i.

A_i = atomic weight of parent isotope of i, gm/mole

E_{ij} = decay gamma-ray energy of isotope[3] i, in Mev/photon

n_{ij} = fraction of photons emitted by isotope i at energy E_j,[3] in photons/disintegration

$\varphi(E)$ = differential neutron flux, in neutrons/cm^2-sec-Mev

$\sigma_i(E)$ = energy dependent absorption cross section of parent isotope i,[4] in cm^2/atom

λ_i = decay constant of isotope i, in units of inverse time = 0.693/half-life

t_r = amount of time that the parent isotope is being irradiated per cycle

t_c = amount of time for one cycle

n = number of cycles

T = time after material being activated is out of neutron flux.

Let $H_i(E_j)$ be defined as follows:

$$H_i(E_j) \equiv \left[\frac{N\rho_i f_i E_{ij} n_{ij}}{A_i}\int_E \varphi(E)\sigma_i(E)dE\right]$$

For one cycle activation (n = 1), and Equation 2 reduces to

$$S_i(E_j) = H_i(E_j)\left[1 - e^{-\lambda_i t_r}\right]e^{-\lambda_i T} \qquad \ldots(3)$$

This is the basic equation for activation of stationary materials. For many cycle activations, i.e., coolant activation,[5] n is large, and Equation 2 reduces to

$$S_i(E_j) = H_i(E_j)\left[\frac{1 - e^{-\lambda_i t_r}}{1 - e^{-\lambda_i t_c}}\right]e^{-\lambda_i T} \qquad \ldots(4)$$

If $\lambda_i t_c$ (and hence $\lambda_i t_r$) is a small number, i.e., less than about 0.1, Equation 4 reduces to

$$S_i(E_j) \approx H_i(E_j)\frac{t_r}{t_c}e^{-\lambda_i T} \qquad \ldots(5)$$

Equations 4 and 5 are usually used for calculating reactor coolant activities. In this case:

t_r = time coolant spends in core, i.e., in high neutron flux

t_c = coolant cycle time

T = time after reactor shutdown

During reactor operation, Equation 5 reduces to,

$$S_i(E_j) \approx H_i(E_j) \frac{t_r}{t_c} \qquad \ldots(6)$$

Coolant cycle times are usually about one minute, thus for radioactive sodium (Na^{24}), which has a half-life of 15.0 hr, Equation 6 could be used, and the source strength calculated would be quite accurate.

In thermal reactors, most of the activation results from thermal neutrons, for which the following approximation may be used:

$$\int_E \varphi(E)\sigma_i(E)dE \approx \varphi(th)\sigma_i(th) \quad , \qquad \ldots(7)$$

where $\varphi(E)$ and $\sigma_i(E)$ are defined in Equation 2,

$\varphi(th)$ = average thermal flux in the core, in neutrons/cm^2-sec

$\sigma_i(th)$ = thermal neutron absorption cross section of parent isotope i, in cm^2/atom.

In some cases, the fast-neutron activation may be important. In these cases, the radioactive isotopes are produced by (n,γ), (n, p), (n,α), and (n, 2n) reactions.

b. Neutrons

For thermal reactors, like SNAP reactors, using U^{235}, 2.47 neutrons are emitted per fission. The energy spectrum of these neutrons[6] is given in Table VII. It is important to note that this spectrum occurs only during the fission process in the core.

Delayed neutrons and photoneutrons in the core may be ignored, since they represent a very small fraction of the total neutrons produced. Photoneutrons produced out of the core may usually be neglected also since these too, in general represent a small fraction of the total neutrons from the core.

2. After Reactor Shutdown

After the reactor is shut down, since no fissions are taking place, there is no source of neutrons except for the negligible contribution from delayed neutron sources. The only radiation source is fission product decay which results in the emission of beta and gamma rays. For shielding calculations, only

TABLE VII

NEUTRON SPECTRUM FROM FISSION OF U^{235} BY THERMAL NEUTRONS

Energy of Neutrons, E (Mev)	Fraction of Neutrons Above Energy, E
0	1.000
0.5	0.8531
1.0	0.7024
2.0	0.4024
3.0	0.2131
4.0	0.1076
5.0	0.05259
6.0	0.02505
7.0	0.01169
8.0	5.372×10^{-3}
9.0	2.434×10^{-3}
10.0	1.092×10^{-3}

gamma rays are considered. The activity of fission products after reactor shutdown is given by the following approximate equation:

$$C \approx 1.4 P_o \left[(t - t_o)^{-0.2} - t^{-0.2} \right] , \qquad \ldots (8)$$

where

C = activity of fission products at time t days after reactor startup, in curies ($1\ c \equiv 3.7 \times 10^{10}$ disintegrations/sec)

P_o = operating thermal power of reactor, in watts

t = time after reactor startup, in days

t_o = time the reactor has operated, in days

It should be noted that in Equation 8, $(t - t_o)$ is the number of days after shutdown. This equation should only be used when $(t - t_o)$ is greater than 10 sec.

In order to calculate after-shutdown shielding requirements, the gamma-ray intensity and energy spectrum should be known. Figures 22a, b illustrate the rate of gamma-ray energy emitted per watt of reactor power, as a function of time after shutdown, for various energy groups and for a reactor operation time

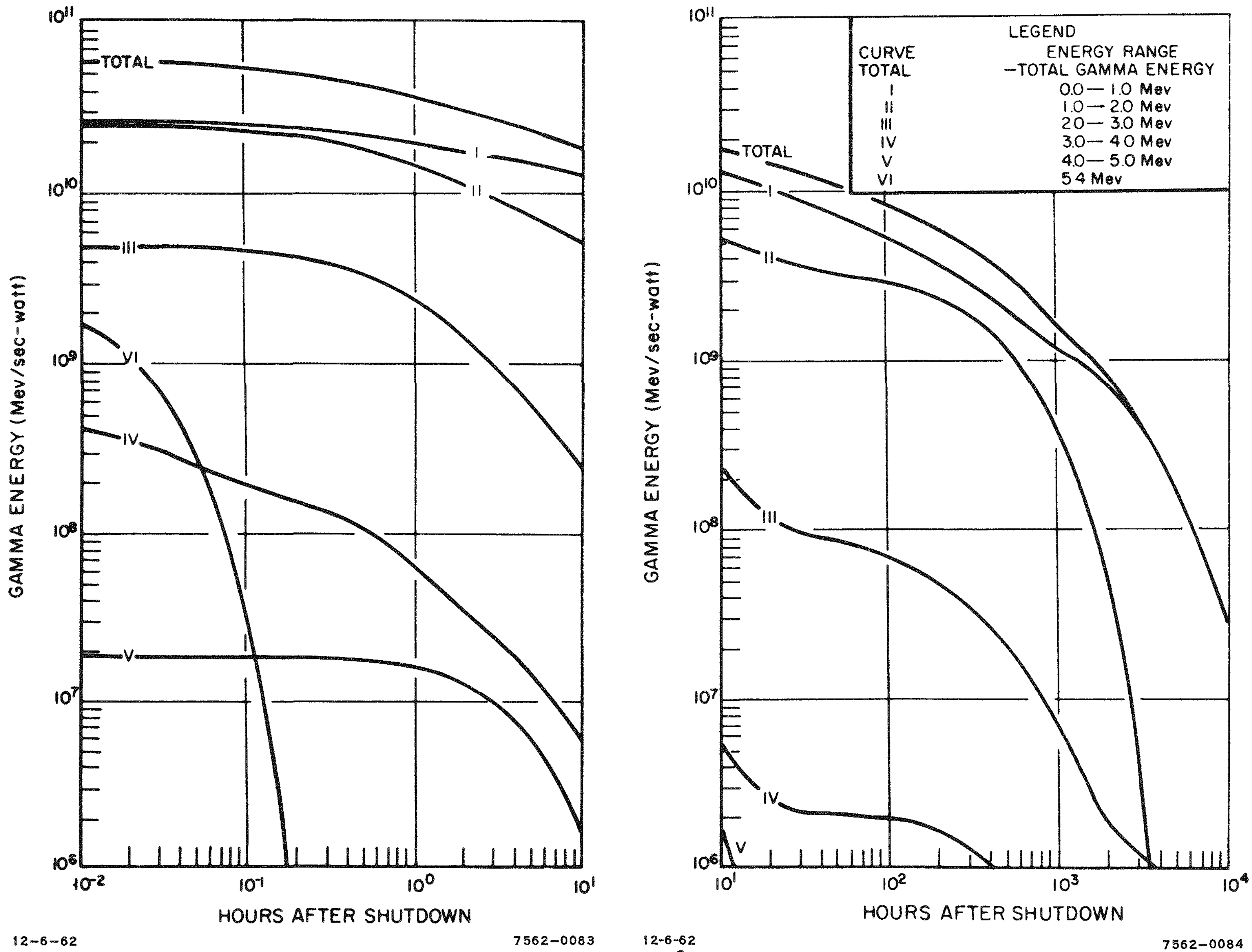

Figure 22. Gamma Ray Energy 10^3 hr Reactor Operation as a Function of Time After Shutdown

of 10^3 hr.[7] It should be emphasized that the gamma-ray energy spectrum does not vary appreciably for operating times from a few days to one year. However, the total gamma-ray energy emitted per watt as a function of time after a shutdown varies with the reactor operation time, as shown in Figure 23. Thus, the gamma-ray energy spectrum in Figure 22 (operating time 10^3 hr) may be used, and the source may be normalized to that in Figure 23, if operating times other than 10^3 are required.

C. SHIELD MATERIAL SELECTION

Great care should be taken in choosing the shielding materials. The temperature behavior and radiation damage characteristics should be considered as well as the shielding properties. Several of the usually ideal shielding materials have been eliminated from consideration because of either poor mechanical properties at the temperature involved or poor radiation damage characteristics.

The shields have been designed using the following method of approach to the problem:

1) Use a few inches of a high density (> 10 gm/cm^3) material in the beginning of the shield to efficiently attentuate core gamma rays.

2) Continue with a gamma-ray shield which has an intermediate density but a high fast neutron removal cross section plus a very low thermal neutron capture cross section for the production of high energy capture gamma rays.

3) Finally, complete the shield with a low density material containing a high fast neutron removal cross section.

In the shields presented here, depleted uranium clad in borated steel was used for the high density shield material. This layer is followed by a region of borated zirconium hydride, forming the intermediate density shield, and, finally, by a region of lithium hydride for the low density shield. (See Table VIII.)

TABLE VIII
SHIELDING MATERIALS

Material	Density (g/cm^3)	Fast Neutron Removal Cross Section (cm^{-1})
Uranium (depleted)	18.7	0.170
$ZrH_{1.8}$	5.55	0.154
LiH	0.710	0.110

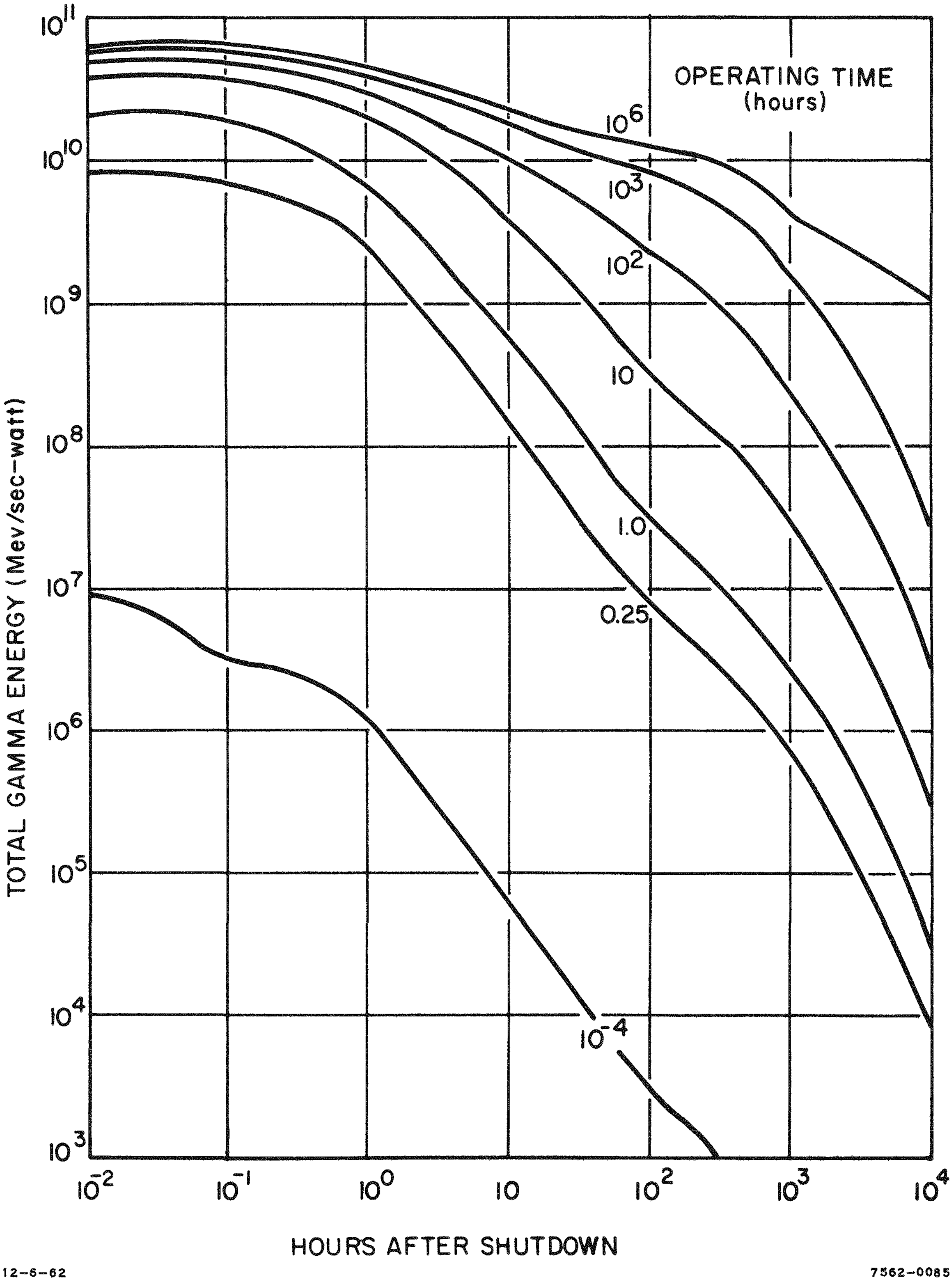

Figure 23. Total Energy for Various Reactor Operation Times

Depleted uranium is an excellent material for the high density shield (density = 18.7 gm/cm^3) and can withstand relatively high temperatures. It also has good attenuation properties for high energy gamma rays accompanying fission in the core and it has a high fast neutron removal cross section, (0.170 cm^{-1}). Furthermore, it has a thermal neutron absorption cross section and resonance integral less than that of tungsten, thus it generates less secondary gamma rays.

The zirconium hydride used in the calculations has the formula $ZrH_{1.8}$ and a density of 5.55 gm/cm^3. It is assumed that the zirconium hydride contains 1% boron by weight and that, as a result, thermal neutron capture gamma rays produced in the zirconium are reduced to such a level that they are only a small fraction of the core gamma rays. This is a reasonable assumption since the thermal neutron microscopic absorption cross sections for the zirconium and for the boron are 0.0065 cm^{-1} and 2.35 cm^{-1}, respectively. As required, the zirconium hydride also has a high fast neutron removal cross section (0.154 cm^{-1}) and, since it contains hydrogen, it effectively slows down intermediate energy neutrons to thermal energies where they are readily absorbed by the boron. The fact that the zirconium also has a fairly good cross section for inelastic scattering helps in slowing down high energy fission neutrons (greater than 5 Mev).

Lithium hydride was chosen for the low density (0.710 gm/cm^3) shield. It has a high effective neutron removal cross section (0.110 cm^{-1} per weight) and it exhibits good thermal and radiation stability.

D. OPERATING SHIELD DESIGN

Two basic shield configurations were considered: a spherical shield partially or fully enveloping the reactor, and a conical shadow shield. The conical configuration is used for applications in which the payload (or manned compartment) size is small, relative to the distance between the payload and the reactor. The spherical configuration is used for cases in which a conical shadow shield would not suffice, such as a large toroidal manned space station with the reactor near the center of the wheel.

1. Optimization of Materials

Three shield materials are used: depleted uranium, borated zirconium hydride, and lithium hydride. Section VIII-C described the bases for the selection of these three shielding materials. A large number of detailed calculations were performed by IBM 7090 computer to determine the optimum thicknesses of the three materials, based on minimum total shield weight, for various dose rates and payload-reactor separation distances. The results of this analysis indicated that two major simplifications could be effected in the design of radiation shields utilizing these materials:

a) A constant uranium thickness of 6 cm may be used in the shield design since this uranium thickness results in a minimum total shield weight for both configurations which were studied.

b) For any specific dose rate and separation distance, use of the zirconium hydride and lithium hydride thicknesses which were calculated for the spherical configuration results in a total weight for the conical configuration which is within 5% of the weight given by the use of the thicknesses calculated for the conical shape. Thus, it is possible, within the accuracy required for a preliminary design, to use the same material thicknesses for both basic configurations.

2. Shielding Material Thicknesses

Figures 24, 25, and 26 are plots of zirconium hydride and lithium hydride thicknesses required to give a minimum weight shield design for a payload dose rate of 7.5 mr/hr, for reactor power levels of 100, 200, and 300 kwt. In order to obtain a shield weight based on these curves, it is necessary to determine the shield configuration required which is normally dependent upon payload size, shape, and distance from the reactor. The graph is entered with the payload-reactor separation distance to obtain the optimum zirconium hydride and lithium hydride thicknesses. Then, with the use of the material densities given in Table VIII, the weight of uranium, zirconium hydride, and lithium hydride may be calculated.

3. Determination of Thicknesses

To determine the required material thicknesses for dose rates other than 7.5 mr/hr, the following relationship is used:

$$S_2 = S_1\left(\frac{D_1}{D_2}\right)^{1/2} , \qquad \ldots(9)$$

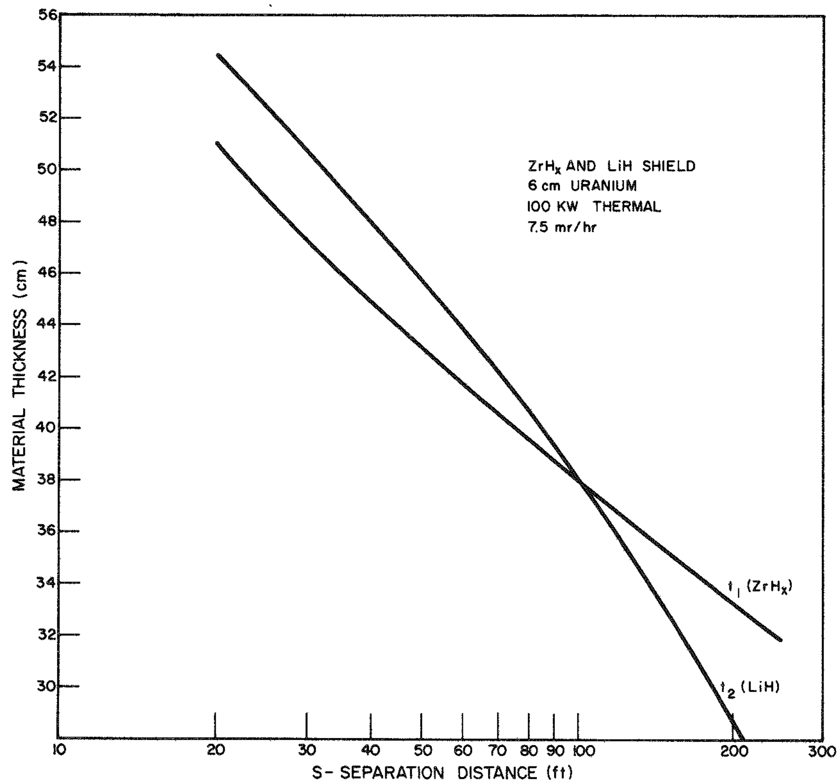

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Figure 24. Material Thickness vs Separation Distance (100 kwt)

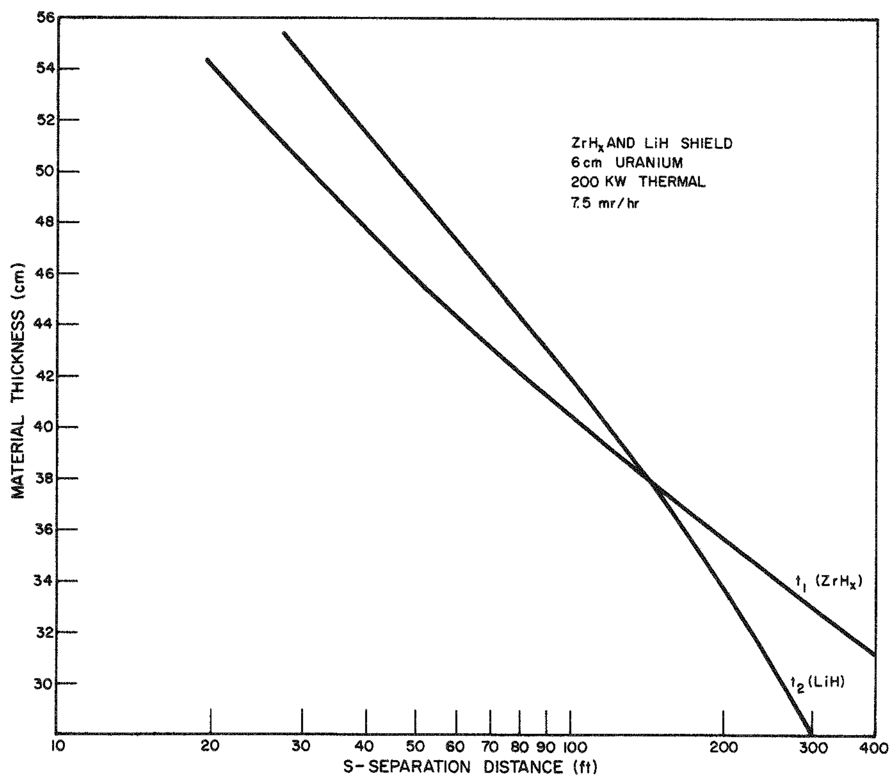

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Figure 25. Material Thickness vs Separation Distance (200 kwt)

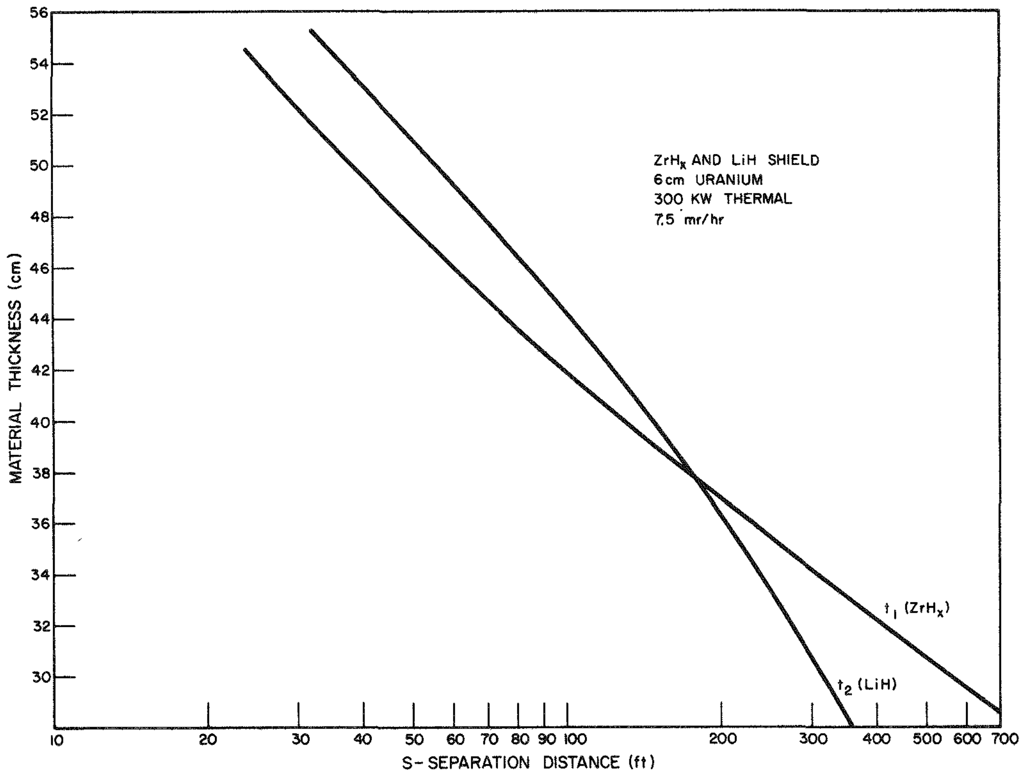

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Figure 26. Material Thickness vs Separation Distance (300 kwt)

where

S_1 = the actual separation distance, ft

D_1 = the required dose rate, mr/hr

D_2 = 7.5 mr/hr

S_2 = value of separation distance used to obtain material thicknesses from the graph.

Example: With a reactor power level of 100 kwt, an actual separation distance of 50 ft, and an allowable dose rate of 15 mr/hr, Figure 24 would be entered with

$$S_2 = 50\left(\frac{15}{7.5}\right)^{1/2} = 70.7 \text{ ft} \quad ,$$

Thus, it is found that, for a minimum weight shield with the above conditions, 6 cm of uranium, 40.5 cm of borated zirconium hydride, and 42 cm of lithium hydride should be used.

To obtain optimum material thicknesses for reactor power levels other than those shown in Figures 24, 25, and 26, the following expression may be used:

$$S_2 = S_1\left(\frac{P_2}{P_1}\right)^{1/2} , \qquad \ldots(10)$$

where

P_1 = the actual power level

P_2 = the power level for which the graph is drawn.

Example: For a reactor power level of 165 kwt, a separation distance of 100 ft, and a dose rate of 7.5 mr/hr, Figure 25 would be entered with an equivalent separation distance of:

$$S_2 = 100\left(\frac{200}{165}\right)^{1/2} = 110 \text{ ft}$$

The required material thicknesses are found to be 6 cm of uranium, 39.8 cm of borated zirconium hydride, and 41 cm of lithium hydride.

E. SHUTDOWN SHIELD

When the reactor is shut down following operation, the radiation source consists of decay gammas from the reactor core and gammas due to activation of the NaK primary coolant which has passed through the core during operation. Thus, the shielding material required for the shutdown condition consists entirely of a high density material, such as depleted uranium or tungsten.

The shielding required for reactor operation includes sufficient heavy material to reduce the shutdown dose rate in the normally manned areas to an insignificant level. Therefore, a shutdown shield is required only to shield those regions which are not normally manned but which must be shielded during special operations, such as rendezvous of a manned vehicle with the station.

Figure 27 shows the thickness of uranium required to reduce the shutdown dose rate to 1 r/hr, at various separation distances, for the cases in which the operational power level is 100, 200, and 300 kwt. The uranium thickness required for a dose rate other than 1 r/hr may be found in the same manner as described in the section above for the operating dose rates.

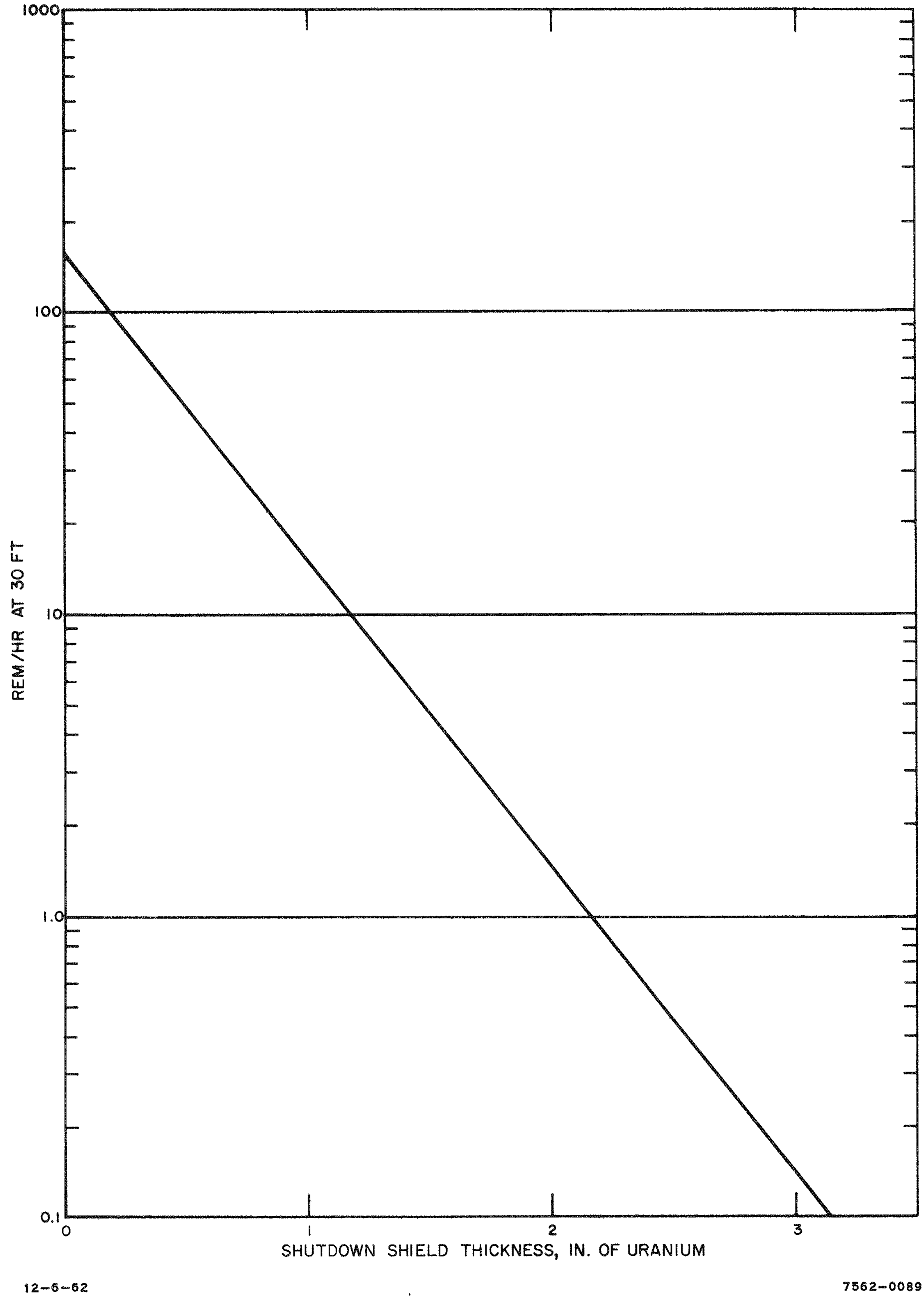

Figure 27. Shutdown Shield Thickness vs Separation Distance

F. SHIELDING PROCEDURES

The preliminary shield analysis presented is based on an idealized spherical configuration. No attempt was made to reduce weight by taking credit for directional effects that would be considered in a final design. Basic assumptions used in this analysis are as follows:

1) The core was assumed to have a uniform neutron and gamma-ray source and core self-absorption was considered.[8]

2) The core gamma-ray source strength was calculated to be approximately 17 Mev/fission, and it was assumed this gamma-ray energy is carried exclusively by 3 Mev photons.

3) A gamma-ray dose buildup factor of 20 was used in the calculations. This corresponds to approximately 20 mean-free paths for any material.[9] Approximately 20 mean-free paths are necessary to reduce the dose rate from core gamma-rays to 7.5 mrem/hr at 75 ft.

4) The fast neutron dose rate was calculated using an approximate form of a double exponential expression for fast neutron attenuation in a hydrogenous shield.[10] This expression may be used in the present shield design since both zirconium hydride and lithium hydride are hydrogenous media. However, for deep penetrations, i.e., greater than 10 mean-free paths, only the second term in the attenuation kernel of the double exponential expression for fast neutrons becomes important. As a result, in the present calculations, only the second term was used.

5) The secondary gamma-ray source born in the depleted uranium shield was incorporated in the calculations by multiplying the core gamma source strength by the ratio of the gamma-ray dose rate including uranium secondaries to the dose rate without uranium secondaries. It should be noted that this ratio was obtained from experimental data,[11] and Figure 28 shows this ratio as a function of depleted uranium thickness.

6) Gamma rays resulting from fast neutron inelastic scattering in zirconium were explicitly considered. It was assumed that secondary gamma rays from degraded neutrons can be suppressed by proper placement of boron.

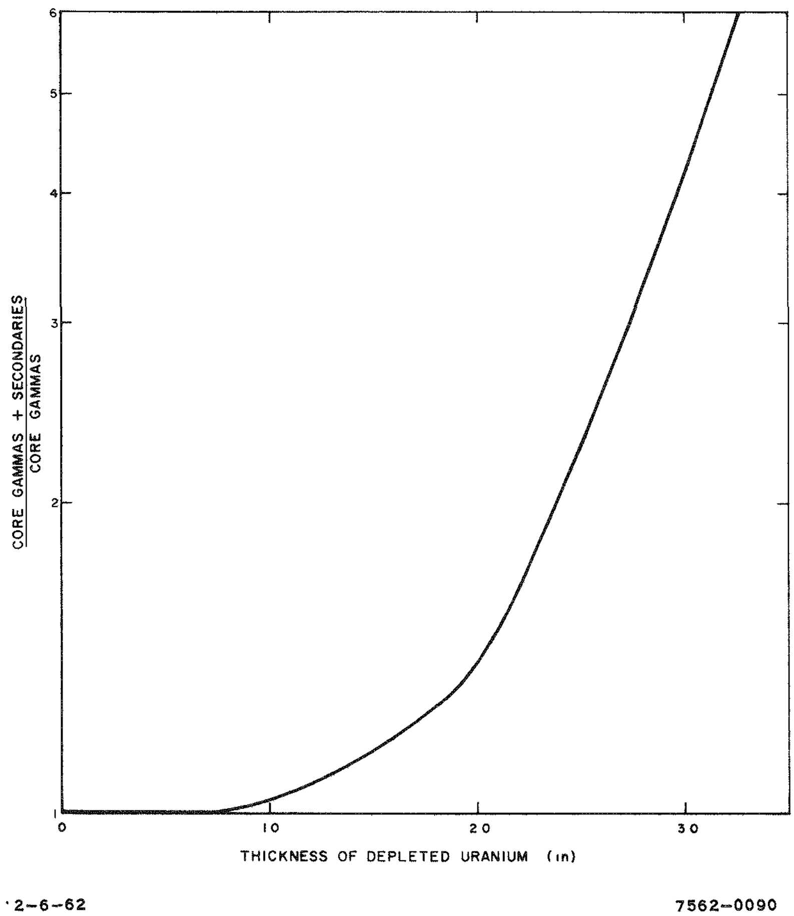

Figure 28. Ratio of Core Gammas Plus Uranium Secondaries to Core Gammas

The resulting expressions for the dose rates from core gamma rays (including secondaries from depleted uranium), core neutrons, and gamma rays from fast neutron inelastic scattering in zirconium are as follows:

$$D_n = \frac{VFPN}{4\pi R^2 A} \exp -\left(\Sigma_c Z_n + \sum_i \Sigma_i t_i\right) \quad , \qquad \ldots(11)$$

$$D_g = \frac{SFPB_g G\acute{U}(T)}{4\pi R^2} \exp -\left(\mu_c Z_g + \sum_i \mu_i t_i\right) \quad , \qquad \ldots(12)$$

$$D_i = \frac{3}{16\pi} \Sigma_I EMB_i \varphi_o \left(\frac{a}{R}\right)^2 e^{-\mu_2 t_2} \left(\frac{e^{-\Sigma_1 t_1} - e^{-\mu_1 t_1}}{\mu_1 - \Sigma_1}\right) \quad , \qquad \ldots(13)$$

$$D = D_g + D_n + D_i \quad , \qquad \ldots(14)$$

where

D_g = dose rate from core gamma rays and depleted uranium secondaries, mrem/hr

D_n = dose rate from core fast neutrons, mrem/hr

D_i = dose rate from inelastic gamma rays generated in zirconium, mrem/hr

D = total dose rate at detector, mrem/hr

S = core gamma-ray source strength, 17 Mev/fission

V = effective number of neutrons produced per fission,[9] 2.47

F = conversion factor, 3.3×10^{10} fissions/sec-watt

P = thermal power level of reactor, watts

R = distance from center of core to detector, cm

B_g = dose buildup factor for 3 Mev gamma rays

G = dose conversion factor for 3 Mev gamma rays

$$1.45 \times 10^{-3} \frac{\text{mrem/hr}}{\text{Mev/cm}^2\text{-sec}}$$

N = dose conversion factor for fast neutrons

$$50 \text{ n/cm}^2\text{-sec} = 7.5 \text{ mrem/hr}$$

A = constant used in double exponential expression for fast neutron attenuation,[9] 8.27

T = thickness of depleted uranium shield, cm

$U(T)$ = ratio of gamma-ray dose rate including depleted uranium secondaries to the dose rate without depleted uranium secondaries, for T cm of depleted uranium (see Figure 28).

μ_c = linear gamma-ray absorption coefficient of core at 3 Mev, cm^{-1}

Σ_c = neutron removal cross section of core, cm^{-1}

Z_g = effective self-attentuation distance for gamma rays in core, cm

Z_n = effective self-attentuation distance for fast neutrons in core, cm

μ_i = linear absorption coefficient for 3 Mev gamma rays of material i outside core, cm^{-1}

Σ_i = fast neutron removal cross section for material i outside core, cm^{-1}

T_i = thickness of material i, cm

Σ_I = microscopic cross section for production secondary gamma rays in ZrH, cm^{-1}

E = energy of secondary gamma rays generated, 1.5 Mev/capture

M = dose conversion factor for 1.5 Mev gamma rays

$$1.76 \times 10^{-3} \frac{\text{mrem/hr}}{\text{Mev/cm}^2\text{-sec}}$$

B_i = dose buildup factor for 1.5 Mev gamma rays[8]

φ_o = fast neutron flux incident on ZrH shield, n/cm^2-sec

a = distance from center of core to beginning of ZrH shield, cm

μ_1 = linear absorption coefficient for 1.5 Mev in ZrH, cm^{-1}

μ_2 = linear absorption coefficient for 1.5 Mev in LiH, cm^{-1}

Σ_1 = fast neutron removal cross section for ZrH, cm^{-1}

t_1 = thickness of ZrH shield, cm

t_2 = thickness of LiH shield, cm

An optimization study was performed in order to get a minimum weight. The following shield thicknesses were varied: depleted uranium, zirconium hydride, and lithium hydride. It was found that 6 cm of depleted uranium gave a minimum weight.

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IX. THE 3-kwe SNAP 2 SYSTEM

A. GENERAL

The SNAP 2 nuclear APU system consists of three major subsystems – the reactor heat source, the power-conversion unit, and the radiator-condenser heat rejection system. A drawing (Figure 29) illustrates the basic flight test installation. Energy is produced in the nuclear reactor by the fissioning of U^{235}. A liquid-metal (NaK-78) heat transfer fluid is circulated through the reactor core and to the NaK-to-mercury boiler-superheater by a rotating permanent-magnet pump. In the boiler-superheater, the reactor heat is transferred from the primary reactor coolant to the mercury working fluid of the Rankine power-conversion cycle. The reactor heat converts liquid mercury into superheated vapor which is expanded through a turbine. The resulting mechical power output of the turbine is converted to electrical power by the alternator. The mercury-vapor exhaust from the turbine is condensed in the radiator-condenser, which is part of the outer skin of the space vehicle. Because of the space environment, the cycle rejection temperature must be maintained by radiative heat rejection only. The mercury condensate is returned to the boiler by a boiler-feed pump. SNAP 2, therefore, incorporates the major components of a conventional nuclear electric plant except that the cycle working fluid is mercury instead of water, and the cycle heat is rejected by radiation to space instead of a conventional water heat sink.

A generalized cycle schematic of the complete SNAP 2 system is shown in Figure 30. Specifically, the system performance values for the flight system are tabulated in Table IX.

B. REACTOR SUBSYSTEM

The SNAP 2 reactor, as shown in Figure 31, employs a homogeneous fuel moderator of zirconium hydride containing U^{235}. For minimum weight, the reactor is reflected by beryllium and controlled by variation of the effective reflector thickness by means of angular rotation of two semicylindrical beryllium drums. Each fuel element is clad in a thin-walled steel tube for liquid-metal exclusion. The steel-clad tubes are internally coated to prevent hydrogen loss from the fuel-moderator material. The core is contained in an approximate 9-in. -diam-core vessel, with the beryllium radial reflector outside the vessel.

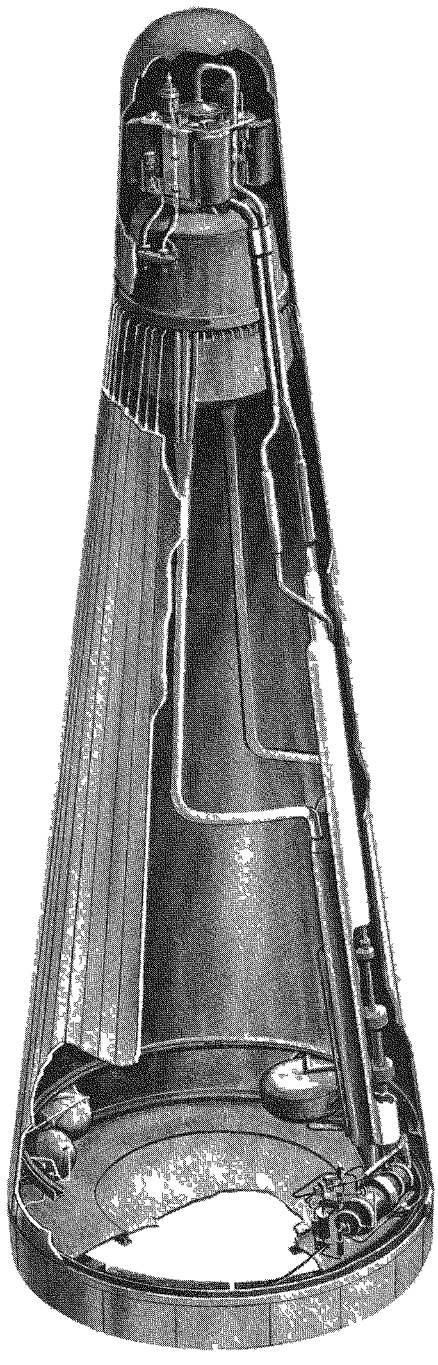

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Figure 29. SNAP 2 APU Configuration

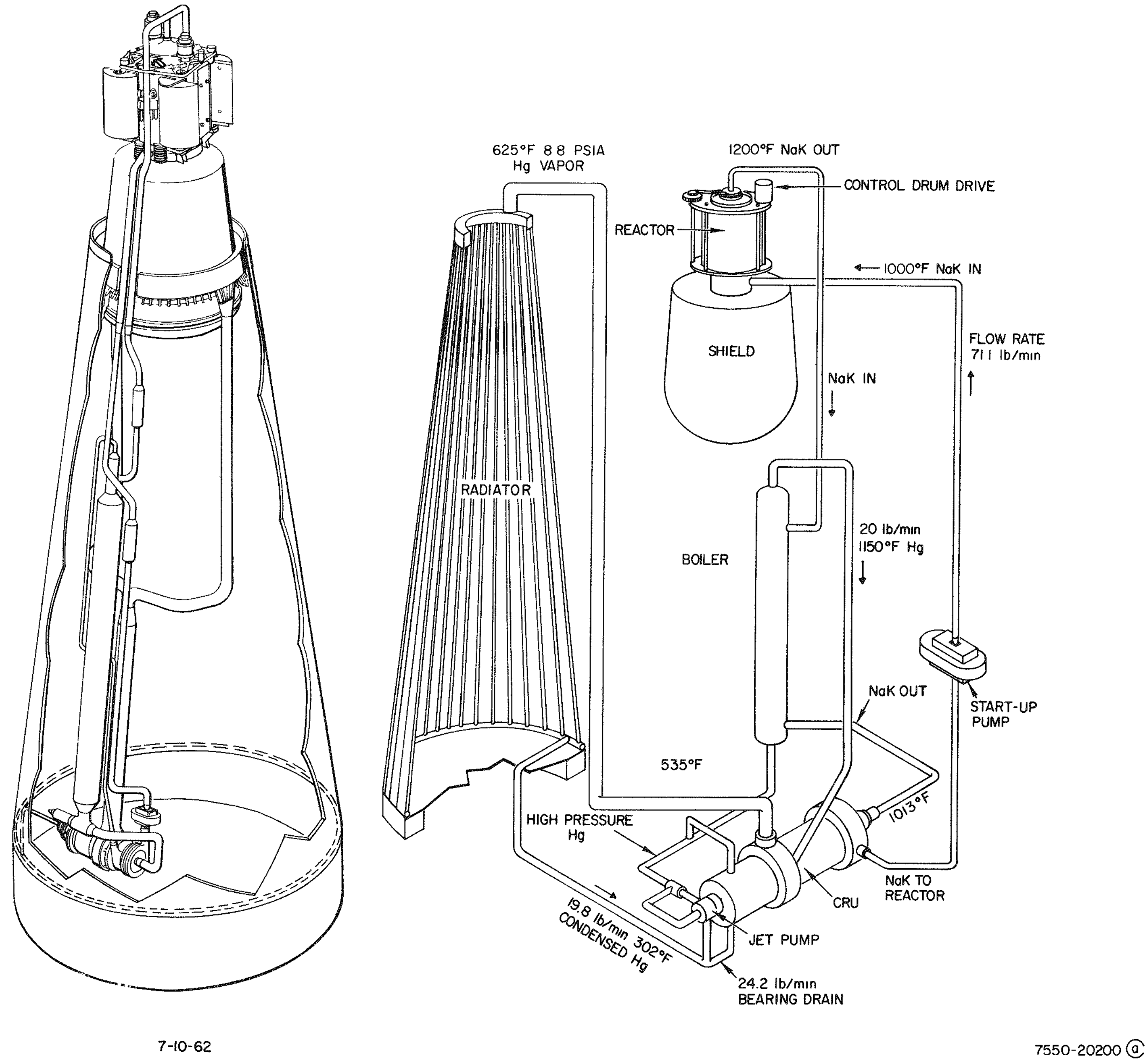

Figure 30. SNAP 2 Schematic

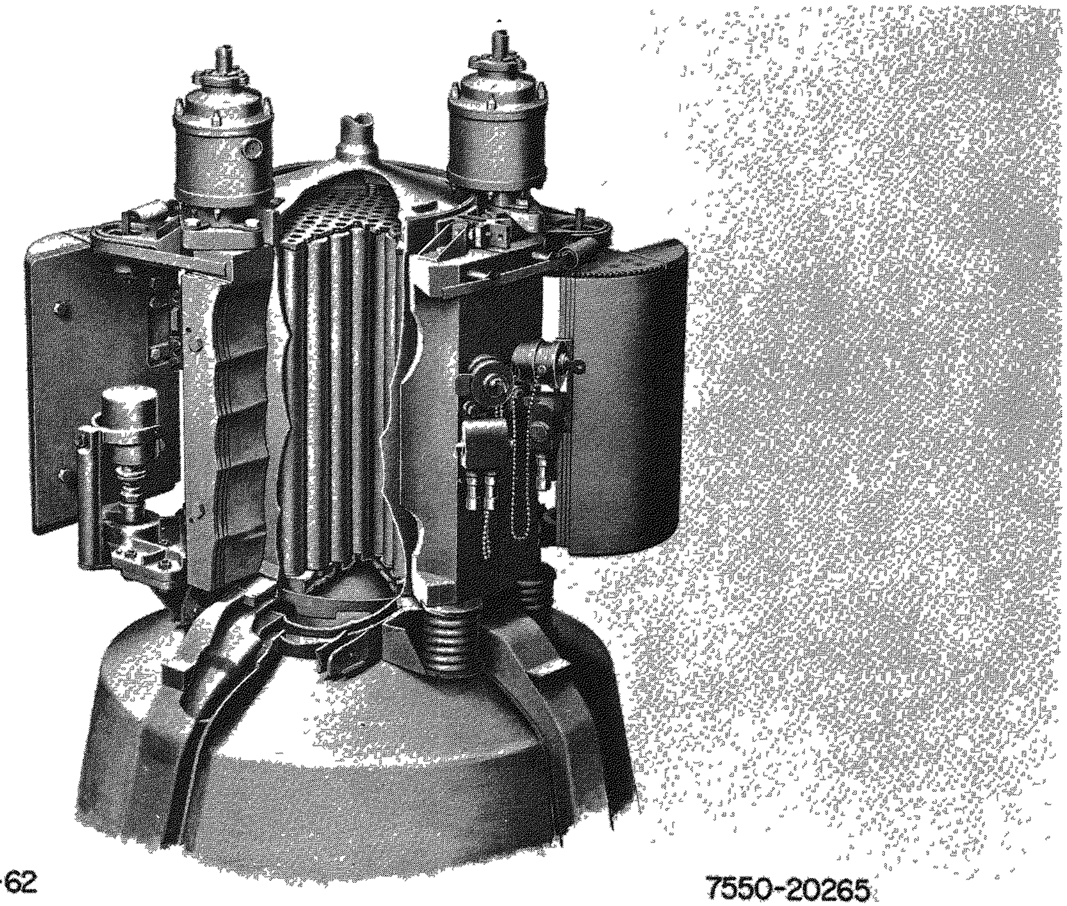

Figure 31. SNAP 2 Reactor

TABLE IX

SYSTEM PERFORMANCE

System thermal power		
Thermal power generated		54.37 kw
Reactor	53.17	
Shield	0.70	
Reflector	0.50	
Thermal power to PCS		50.32 kw
Reactor power	53.17	
Direct line losses	- 3.75	
Parasitic heater input	0.25	
NaK pump input	0.65	
Turbine shaft power		5.56 kw
Thermal power to PCS	50.32	
Preheat from CRU cooling and drains	2.00	
Mercury losses in CRU	- 0.57	
Turbine exhaust	-46.19	
Alternator power output		3.43 kwe
Turbine shaft power input	5.56	
Mercury pump	- 0.13	
NaK pump	- 0.65	
Bearings and seal	- 0.70	
Alternator (84% efficiency)	- 4.08	
Electrical output		3.00 kwe
Alternator power output	3.43	
Reactor control	- 0.02	
Instrumentation	- 0.01	
CRU speed control	- 0.10	
Degradation	- 0.30	
Heat rejection		51.37 kw
Radiator-condenser	49.25	
NaK lines to space	0.16	
Reactor to space	0.50	
Reflector to space	0.70	
Shield to space	0.50	
Electrical and instrument compartment	0.26	

TABLE IX (Continued)

System thermal power (Continued)	
System efficiencies	
Net system efficiency (3.0 kwe)	5.5%
Rankine cycle efficiency (3.43 kwe)	6.8%
Shaft mechanical efficiency (3.43 kwe)	33%
System typical operational characteristics	
Reactor inlet temperature	1000°F
Reactor outlet temperature	1200°F
NaK primary system flowrate	71.1 lb/min
NaK primary system pressure drop	2.0 psi
Hg inlet temperature to boiler	530°F
Hg superheated vapor temperature	1150°F
Hg pressure turbine inlet	115 psi
Hg pressure turbine outlet	9.0 psi
Hg condensing pressure (min)	6.0 psi
Hg subcooled temperature	351°F
Hg pump inlet temperature	423°F
Hg pump outlet pressure	525 psi
Hg pump flowrate	44 lb/min
Hg system flowrate	20 lb/min

The reflector is completely separable from the core for safe reactor shutdown and handling. The thermal output is removed by the flow of NaK-78 axially through the core within the interstitial passages between the fuel elements. The coolant enters the core at 1000°F and exits at 1200°F.

The development of the SNAP 2 reactor is complete. Two SNAP 2 reactors have been operated. The SNAP Experimental Reactor (SER) went to power in October of 1959. It operated for 6,035 hr without failure including a 1000-hr continuous run at SNAP 2 conditions. This reactor was followed by the SNAP 2 Demonstration Reactor (S2DR) shown in Figure 32. This reactor incorporates flight type fuel elements, bearings, and other materials. It began operation in April 1961, and is still performing successfully. It has been used for various physics measurements and has operated for extended periods.

C. POWER CONVERSION SUBSYSTEM (PCS)

The PCS converts the thermal energy of the primary reactor system to usable electrical power. A major portion of the PCS development has been subcontracted to Thompson Ramo Wooldridge Co. All the PCS rotating components are mounted on a single common shaft which is called the CRU. The individual components of the rotating shaft include:

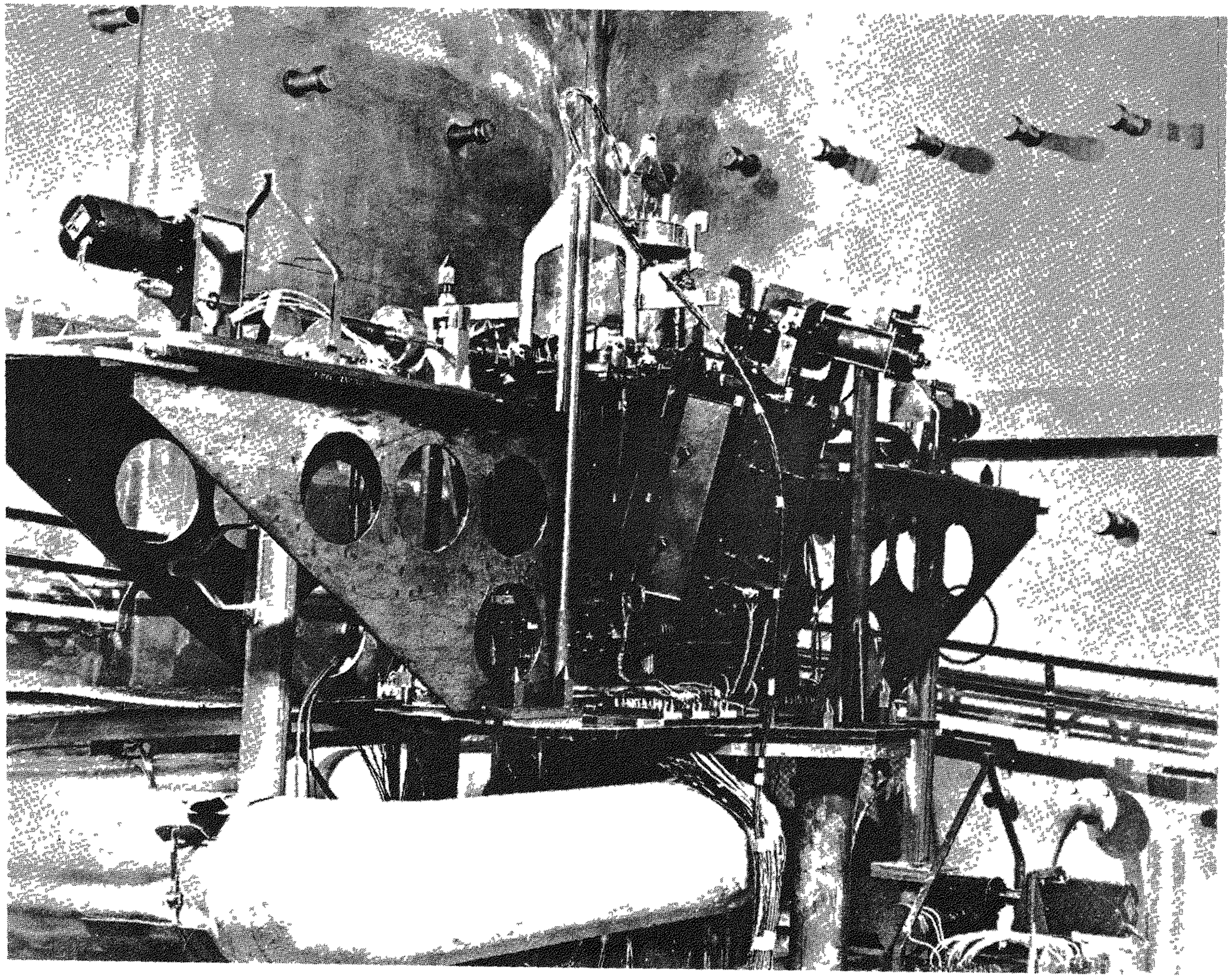

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Figure 32. SNAP 2 Demonstration Reactor (S2DR)

1) The rotating permanent-magnet NaK pump, the operation of which is similar to that of a conventional EM pump with the exception that the moving magnetic field is provided by a rotating magnet.

2) The mercury turbine, a two-stage axial-flow impulse machine.

3) The alternator, a permanent-magnet machine with a sealed stator; the alternator delivers about 3.6 kw at 80 v and 2000 cps.

4) The mercury pump, a conventional but miniature centrifugal ejector type combination pump, supplying pressurized mercury to the boiler and to the bearings.

These are shown in Figures 33 and 34.

All the rotating components — the NaK pump, turbine, alternator, and mercury pump — are mounted on the shaft, which rotates at 40,000 rpm. The shaft is supported by liquid-mercury-lubricated journal and thrust bearings. The entire assembly of rotating machinery is enclosed within a hermetic housing which prevents the loss of the mercury working fluid. Speed control is maintained by a parasitic type load control.

Associated with the CRU is the NaK-to-mercury boiler-superheater. Liquid mercury is preheated, boiled, and superheated within parallel tubes. The NaK flows around the tubes within this annulus.

D. PCS DEVELOPMENT STATUS

Basically, the PCS development is broken down into five major areas:

1) CRU

2) Boiler

3) PCS

4) Parasitic load control

5) Mercury corrosion

A continuing developmental effort on each of these areas has been expended since 1958.

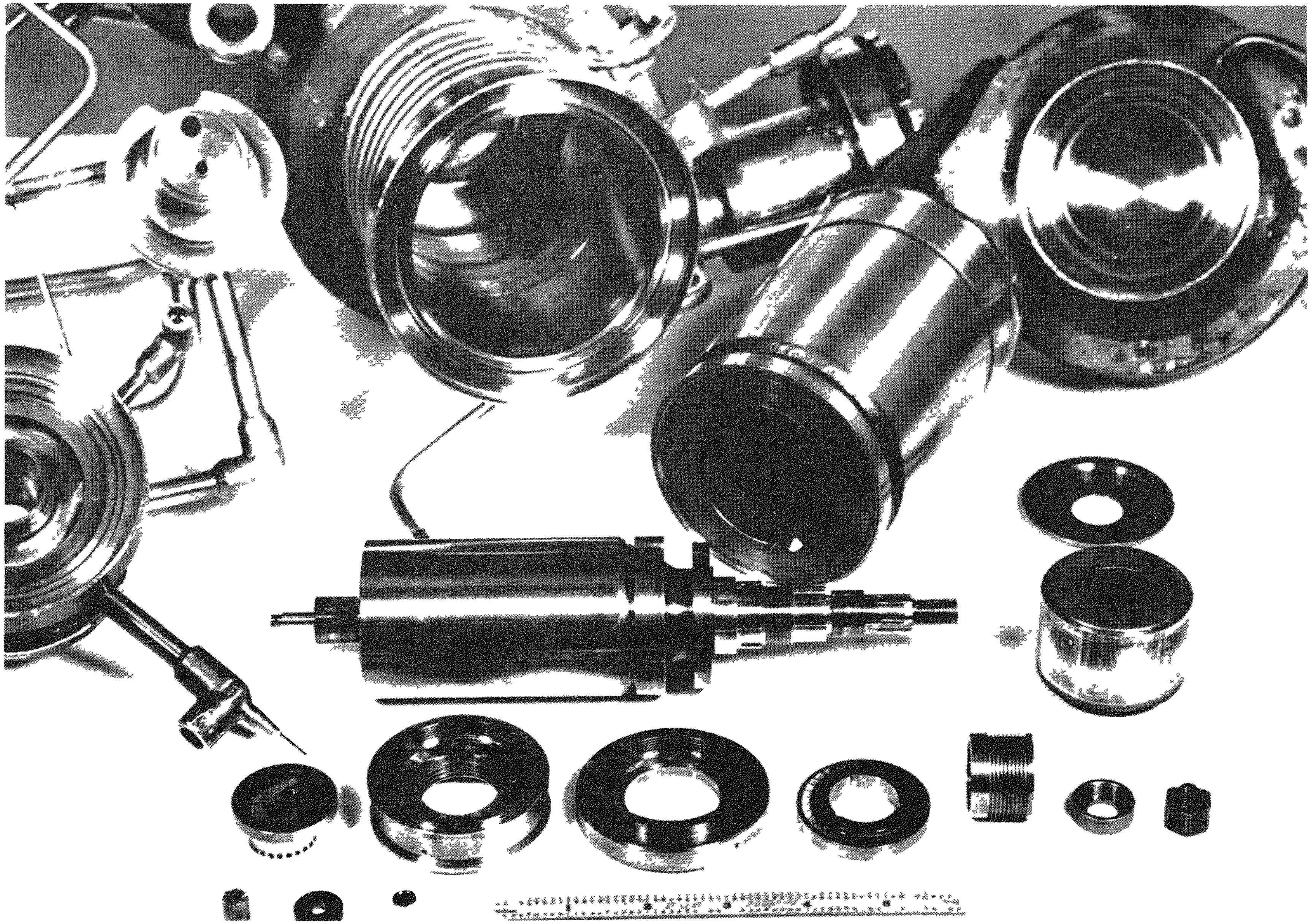

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Figure 33. CRU Components

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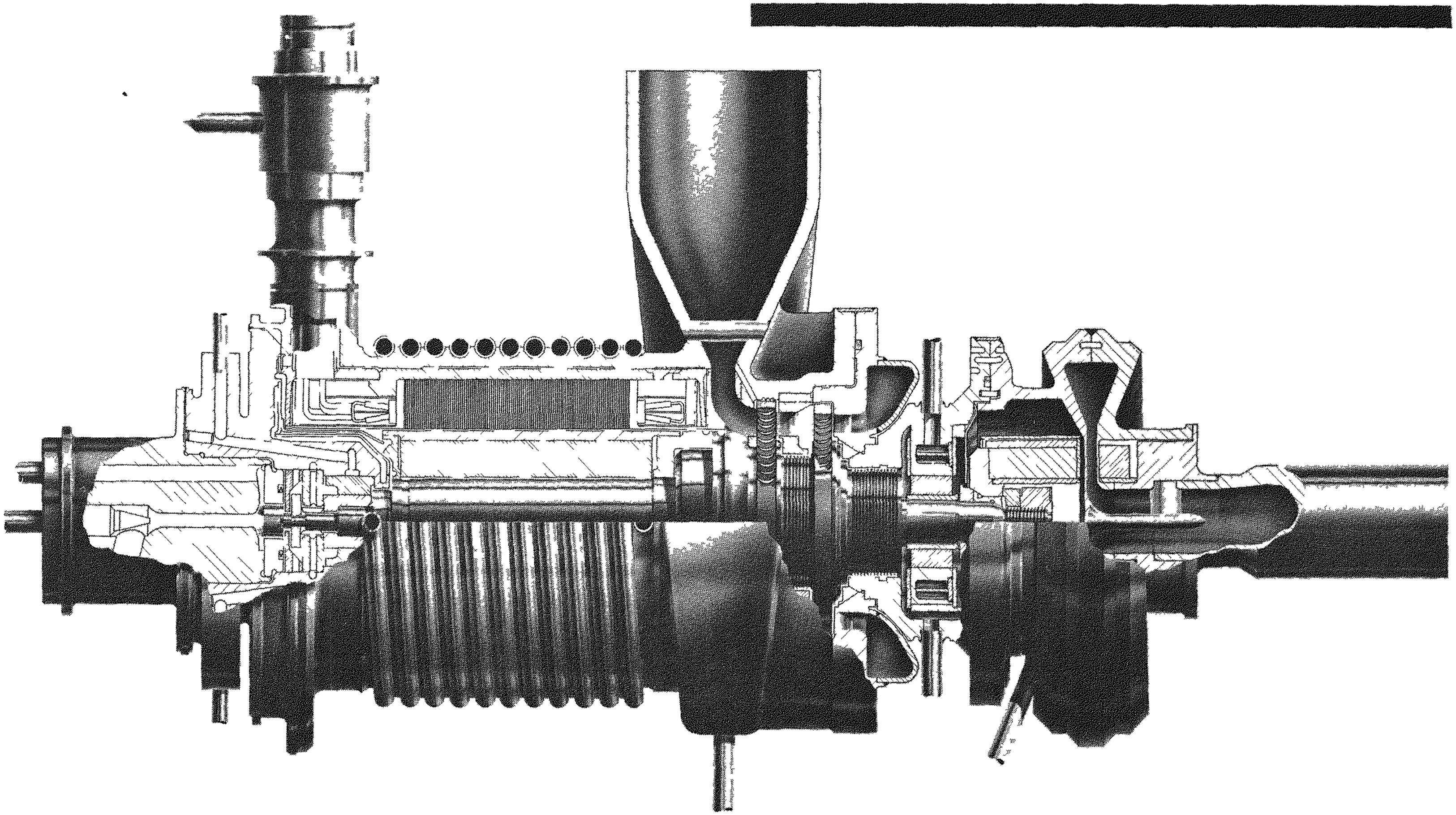

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Figure 34. SNAP 2 CRU

1. CRU

The objective here is to combine the turbine, alternator, mercury pump and NaK pump components into a 36,000 to 40,000 rpm single shaft assembly, utilizing liquid mercury lubricated journal bearings and a thrust bearing. The assembly housing is welded into a hermetically sealed support structure. The basic CRU concept was formulated in 1958. Since that time, the addition of a shaft-mounted EM-type NaK pump has been the only basic modification to the design criteria. During the period of time, four CRU models have been designed, fabricated, and tested. Presently, the fifth model, CRU-5, is now in the preliminary design stage.

E. HEAT REJECTION SUBSYSTEM

The cycle rejection heat is radiated to space by a combined radiator-condenser, which also forms part of the outer structural skin of the space vehicle. Mercury condensation takes place at about 600°F and 6 psia within a number of small-diameter parallel tapered tubes. The tubes are installed within a honeycomb structure to which is bonded a high thermal conductivity fin (copper). The fin, in turn, radiates the heat of condensation and subcooling to space. The total area necessary to radiate the ~50 kw is 120 ft^2.

The above major subsystems combine into an integrated nuclear APU power plant in which the radiator-condenser acts as the basic structural member. The overall APU package is about 181 in. in length.

The weight breakdown for the SNAP 2 flight test application is shown on Table X. Present weight evaluation indicates that the SNAP 2 flight test weight will be achieved. The weights shown are preliminary estimates based, for the most part, on prototype component testing. The specific weights associated with the startup system and with the shield have the highest degree of flexibility.

1. Radiator-Condenser (RC)

Two parallel approaches to the radiator-condenser development are underway, the basic effort on the steel honeycomb concept and the backup effort on the aluminum-steel concept. These two development efforts will be in process until early 1963, at which time the flight design concept will be resolved. Both radiator-condensers are being designed to the same performance objectives.

TABLE X

APU WEIGHT ALLOCATION

Reactor-reflector assembly		258
Primary NaK system		50
Mercury piping system		10
Boiler		75
CRU module		45
Radiator-condenser		220
Startup system		70
Mercury inventory		75
PCS control		45
Electrical, wiring and insulation		37
Structure		55
Destruct charge		10
Diagnostic instrumentation		50*
Contingency		50
	APU	1050
Shield		300
	APU with shield	1350

*100 lb of instrumentation now under consideration

The present radiator-condenser design is based on a total radiation surface area of 120 ft^2. Of this area, about 100 ft^2 are utilized for condensing heat rejection and the remainder for subcooling the liquid mercury. Approximately 50 kwt is rejected by the RC, which acts as the primary structure for the APU and as such provides support for the reactor, shield, PCS, and system piping. The ejectable nose fairing supplied by the vehicle contractor also mounts on the upper RC structure.

2. Honeycomb Concept

The present honeycomb design is established utilizing a 15 PH-7 Mo material core, facing sheets, and condensing tubes. The rectangular tapered tubes are bonded to the interior of the honeycomb core. A continuous copper fin is brazed to the exterior facing of the honeycomb facing. Copper is used as the high thermal conductivity fin bonded to the outer facing to establish a high fin effectiveness. Sprayed on the copper fin is a specially tailored high emissivity coating (ϵ = 0.90 and solar α = 0.30). In the present design, tapered tubes are utilized for condensing and subcooling the mercury turbine exhaust vapor. To maintain high vapor velocity in the low quality region and to reduce the weight of liquid mercury holdup, tapered tubes have been chosen. Structurally, the present RC design is more than adequate to meet the environmental launch conditions.

3. Aluminum-Steel Concept

An accelerated effort on the aluminum steel concept has been started in order to qualify it for the flight design consideration. The present design calls for utilizing rectangular tapered tubes of HS-25 bonded to an 1100-type aluminum skin. The highly emissive coating is sprayed directly on the aluminum skin. Internal structural rings and supports are required to handle the launch loads.

F. PRIMARY SYSTEM DEVELOPMENT

The primary system program consists of developing the following components.

1) Expansion compensator
2) Flow trimmer
3) Expansion joint
4) Orbital startup pump
5) Thermoelectric pump
6) Parasitic load heater
7) Ground test heater

The above components plus the reactor core, boiler, and associated piping make up the SNAP 2 primary NaK system. Each of the components must be developed and then qualified and endurance tested. Following is a brief resume of the component description and status of development.

1. Expansion Compensator

This component compensates for NaK volume expansion at temperature and provides a coolant system pressurization capability. A welded bellows type construction is provided, plus the use of auxiliary springs. Several prototype units have been fabricated and tested for performance and environmental characteristics. Endurance tests at temperature on prototype units are being run.

2. Flow Trimmer

A pipe crimping concept has been selected to adjust to the NaK design flow requirements. The trimmer, which is adjusted at the time of acceptance testing, permits a pressure drop range of 0.07 to 2.0 psi. The unit has been tested for performance and endurance.

3. Expansion Joint

Special piping expansion joints have been developed to compensate for temperature induced dimensional changes. A metal bellows design approach is also being used here. Developmental and endurance testing have been accomplished successfully.

4. Orbital Startup Pump

A separate NaK pump is necessary for ~5% flow circulation prior to orbital startup. This flow prevents the NaK from freezing by utilizing the heat capacity of the system and also establishes the desired conditions for orbital mercury injection and system startup. A dc conduction pump driven from a 1-1/2 v power source has been selected. Extensive testing on this pump has been accomplished.

5. Thermoelectric Pump

As a backup to the CRU shaft-mounted NaK pump, a thermoelectric pump is currently under development. The hot junction of this pump will be the boiler outlet NaK (~1000°F) or possibly the boiler inlet NaK (~1200°F). The cold junction is the mercury preheat to the boiler at about 530°F. Fabrication of the first prototype units is underway.

6. Parasitic Load Heater

Parasitic load heaters (~1.7 kw each) are being developed to absorb the electrical output of the alternator at times when the useful load is less than design maximum. Two separate heaters are utilized; one located in the reactor inlet line and one in the reactor outlet line. Developmental testing has, to date, demonstrated feasibility.

G. CONTROL AND STARTUP

Two independent systems are necessary to control and regulate the SNAP 2 APU system at design conditions. Reactor control is achieved by maintaining the NaK boiler outlet or reactor inlet temperature at a fixed level of about 1000°F. A reactor control drum actuator controller inserts steps of reactivity if the boiler outlet temperature drops below the specified level. The boiler outlet temperature, rather than the reactor outlet temperature, is utilized to hold the driving primary system at mercury boiling temperature differential. Accordingly, any primary system degradation will show up as a reactor outlet temperature increase.

The CRU is controlled by maintaining a constant frequency alternator output. A variable parasitic type resistance load can be varied directly as a function of frequency, thereby establishing a constant speed for the shaft components. A rough control of the PCS subsystem is provided through use of a constant flow device in the boiler liquid mercury inlet line. This flow regulator, in conjunction with the turbine nozzle throttling characteristics, will provide a stabilizing effect to the APU system.

1. Orbital Startup

Startup concepts for SNAP 2 have been continuously under evaluation since 1958. Today, a reasonably reliable approach is being followed as the basic design. In evaluating the various startup schemes, the following criteria must be evaluated:

a) High inherent reliability, which implies a minimum of extra control components

b) Operation both on ground and in space

c) A minimum number of zero gravity or orbital environment effects or perturbations

d) Recycle capability for ground testing

e) Low system weight

f) Operational parameters to be compatible with component characteristics or capabilities

g) Minimum number of new components to be developed

The present concept which best fits the above criteria is described as follows. Just prior to launch, the battery powered NaK startup pump is energized to develop ~5% primary flow. This circulating flow prevents local NaK freezing when the system is in orbit and prior to preheat. After launch and orbital attainment, and once the proper orbit and associated satellite orbital lifetime has been verified, a command signal is relayed through the Agena Command and Communication System to eject any unneeded destruct systems.

To start the preheat phase, the two safety control drums are inserted by firing the release squibs, establishing the reactor at a nominal - $0.30 shutdown. The control drums are then rotated in at a specified rate, causing the reactor to become critical and generate ~4 kw of thermal power. This power, with the prescribed primary flowrate, establishes the 1000°F inlet and 1200°F outlet temperature approximating the steady-state operating conditions. This preheat phase is expected to take about 4 hr.

The actual injection phase is initiated by injecting the mercury, using N_2 pressurization tanks, into the bearings and the boiler. The mercury begins to boil at a steadily increasing pressure level. Mercury vapor is expanded through the turbine and exhausted to the radiator-condenser. Because the radiator-condenser is about 100°F, the vapor condenses initially at a very low pressure.

The heat of vaporization is thus divided between raising the temperature of the RC and by emissive radiation to space. A valve at the RC outlet prevents outflow until a 6-psia pressure is achieved. As the RC accumulates the condensed liquid mercury, the effective condensing heat rejection area is decreased. Eventually, the condensing pressure reaches a value greater than 6 psia, at which time the outlet valve is opened and the liquid inventory flows back into the system. The final RC liquid level establishes itself at a point where the energy radiated from the condensing heat rejection area equals the inlet thermal energy of the turbine exhaust vapor. A schematic of the above steps is shown in Figure 35.

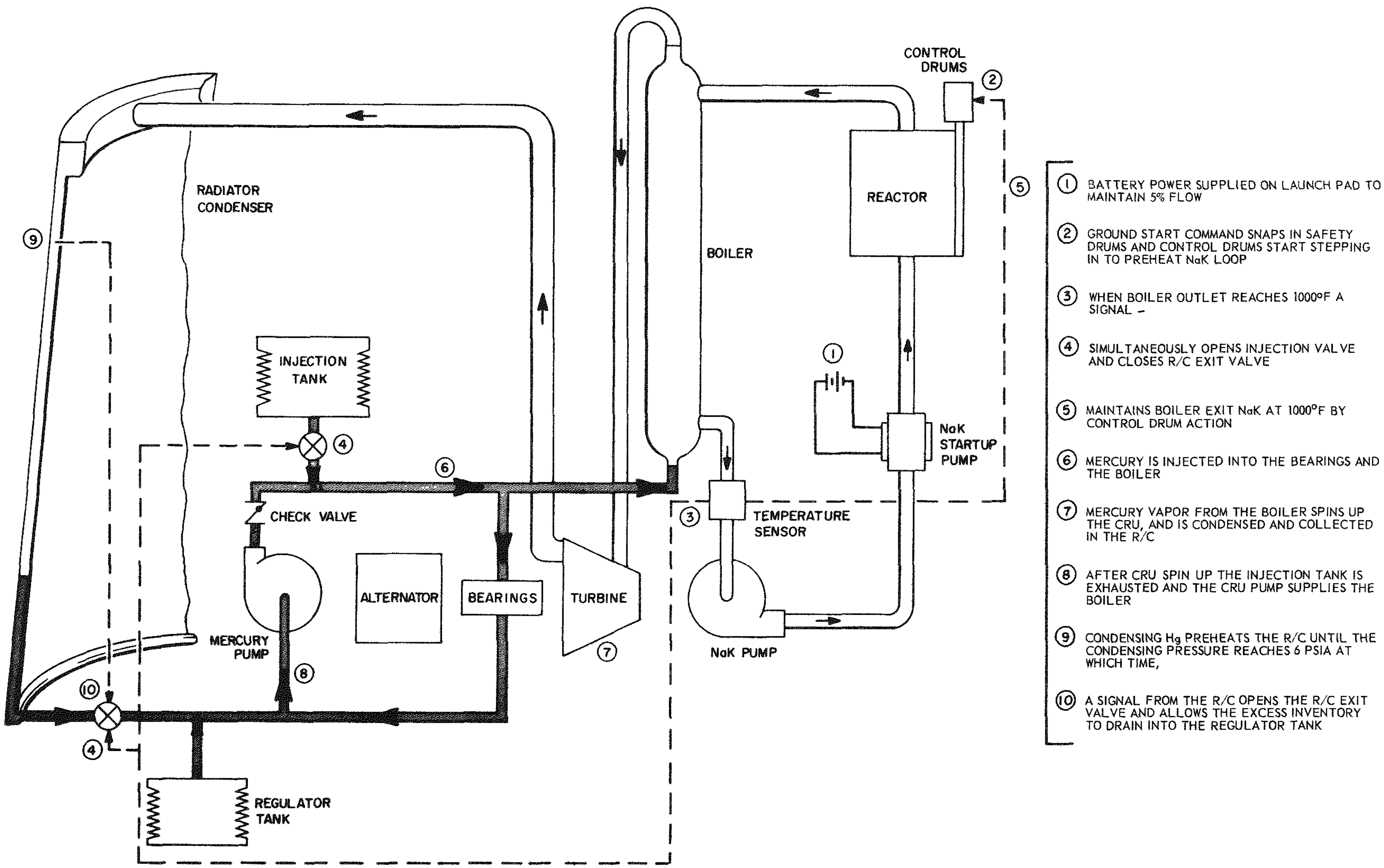

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Figure 35. SNAP 2 Orbital Startup Schematic

During the above phase, the expansion of the vapor through the turbine has torqued the CRU up to design speed. Once stabilized steady-state operation has been achieved, the alternator power is connected to the payload. This overall injection plus stabilization phase takes about 1 hr, of which about 5 min of actual mercury injection time is required.

The primary problems associated with orbital startup of the SNAP 2 system are listed as follows:

a) Mercury pump characteristics as a function of inlet pressure during startup

b) Condensing parallel tube flow distribution and stability in the RC during transient phase

c) Control of liquid mercury inventory in a zero-gravity field prior to establishment of normal operational flow and pressure distributions

d) Temperature transients on the CRU and RC

e) Weight reduction

The primary effects and/or characteristics of startup will be established in the S2PSM-1 system test and S2PSM-3 system orbital simulator.

H. TEST SYSTEMS

System tests include S2PSM-1, a nonnuclear thermal performance test of the APU flight configuration (Figure 36); S2PSM-2, a structural development test (Figure 37), and S2PSM-3, an orbital startup development system (Figure 38).

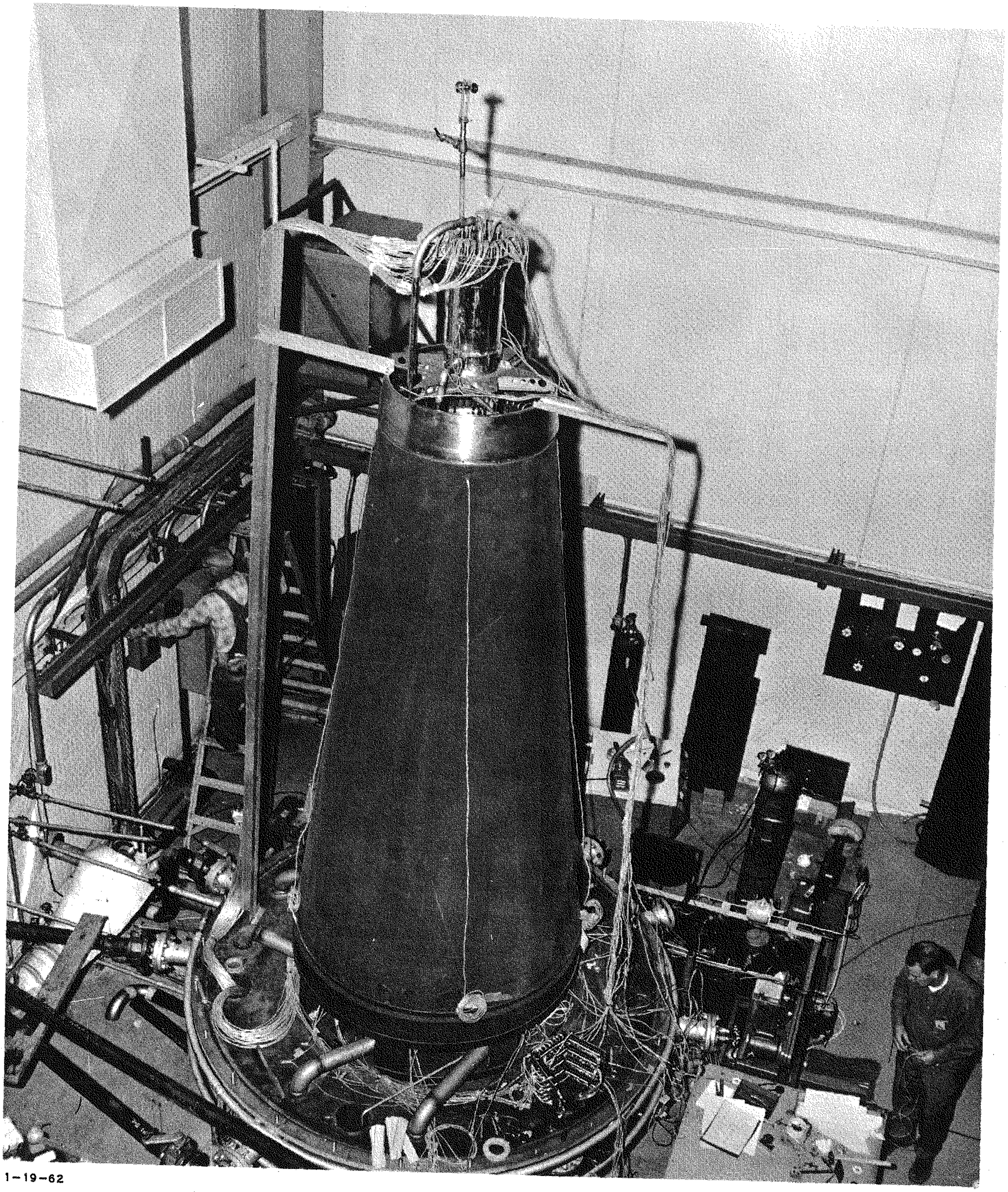

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Figure 36. S2PSM-1

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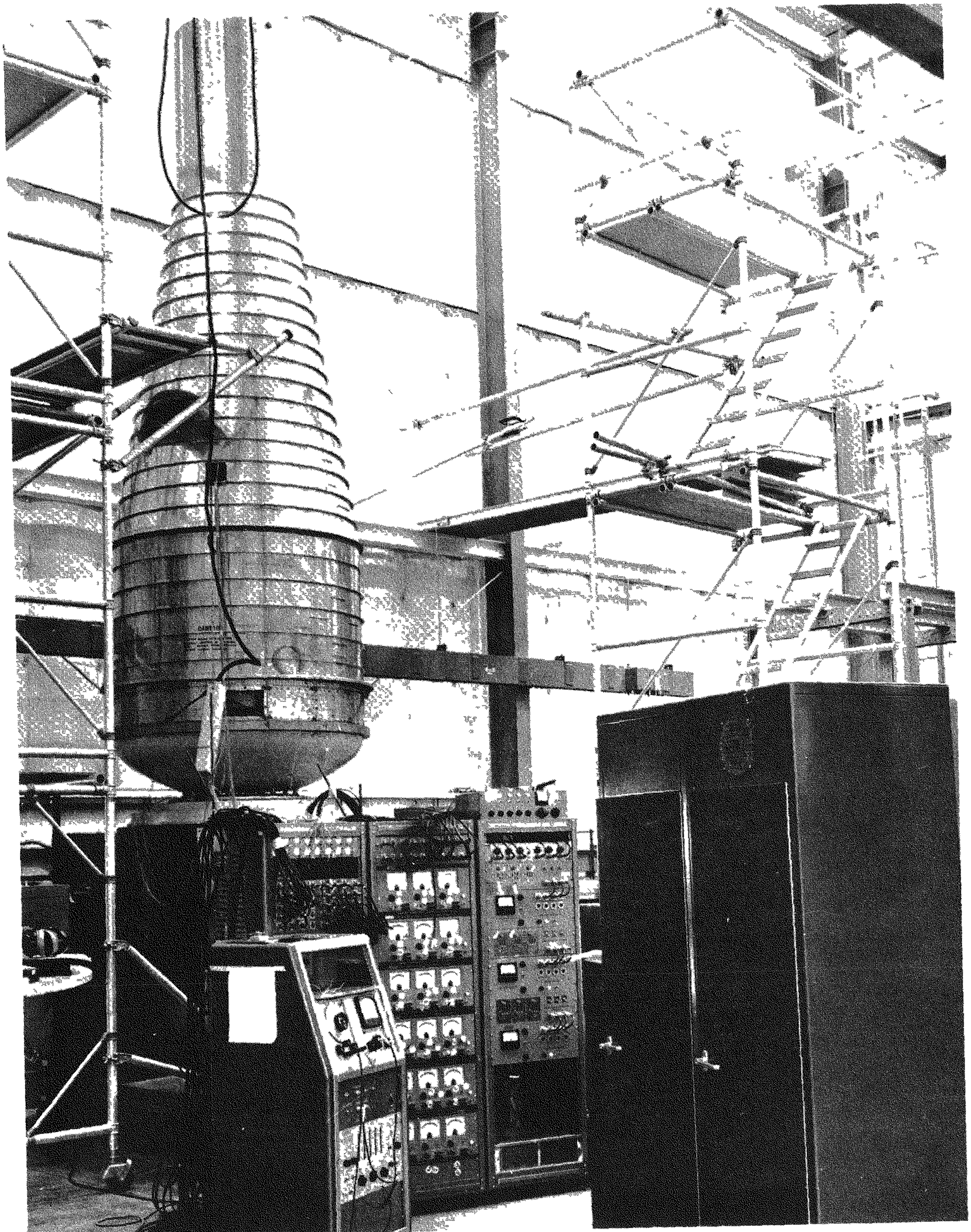

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Figure 37. S2PSM-2

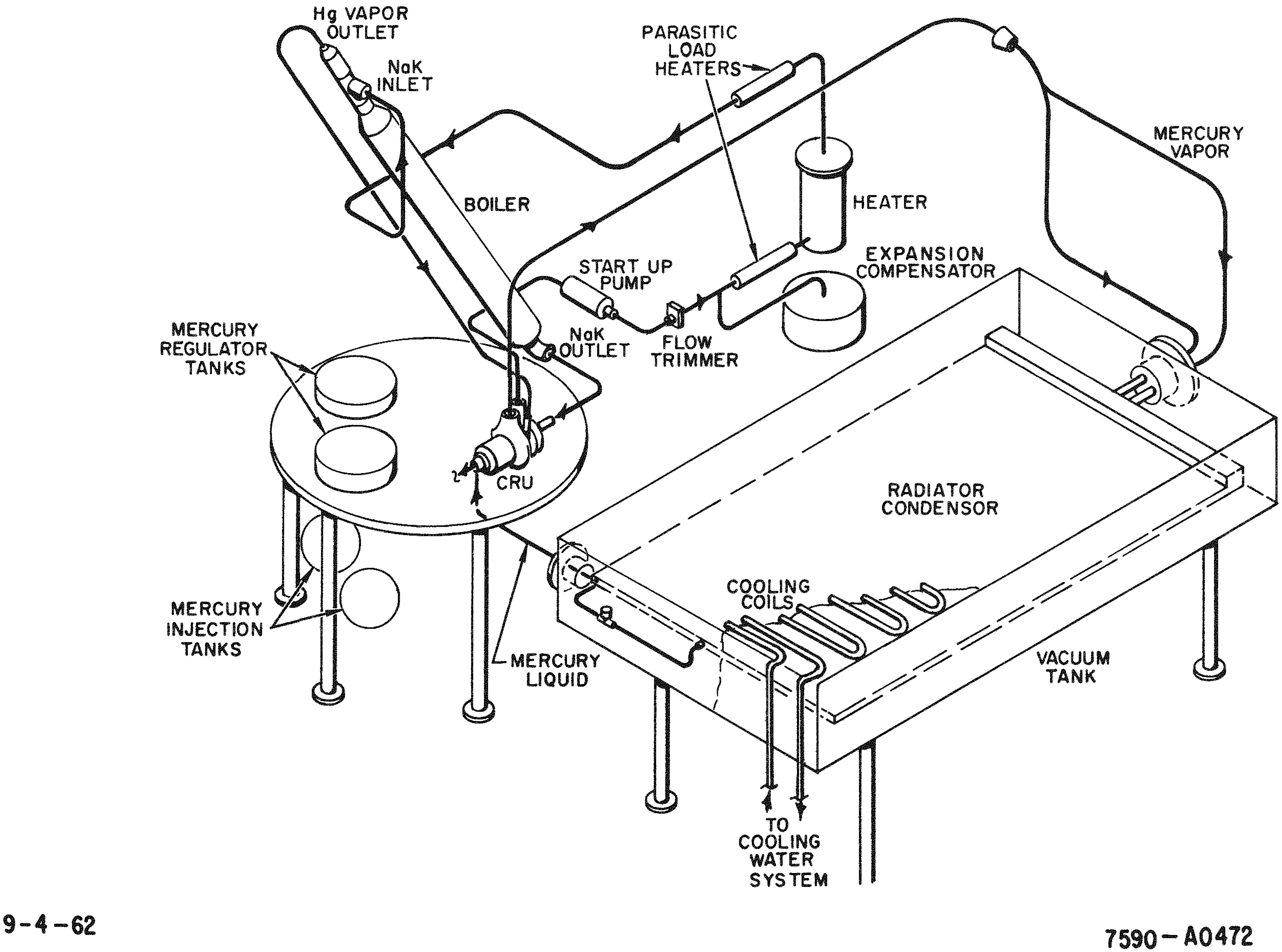

Figure 38. Orbital Startup Development System