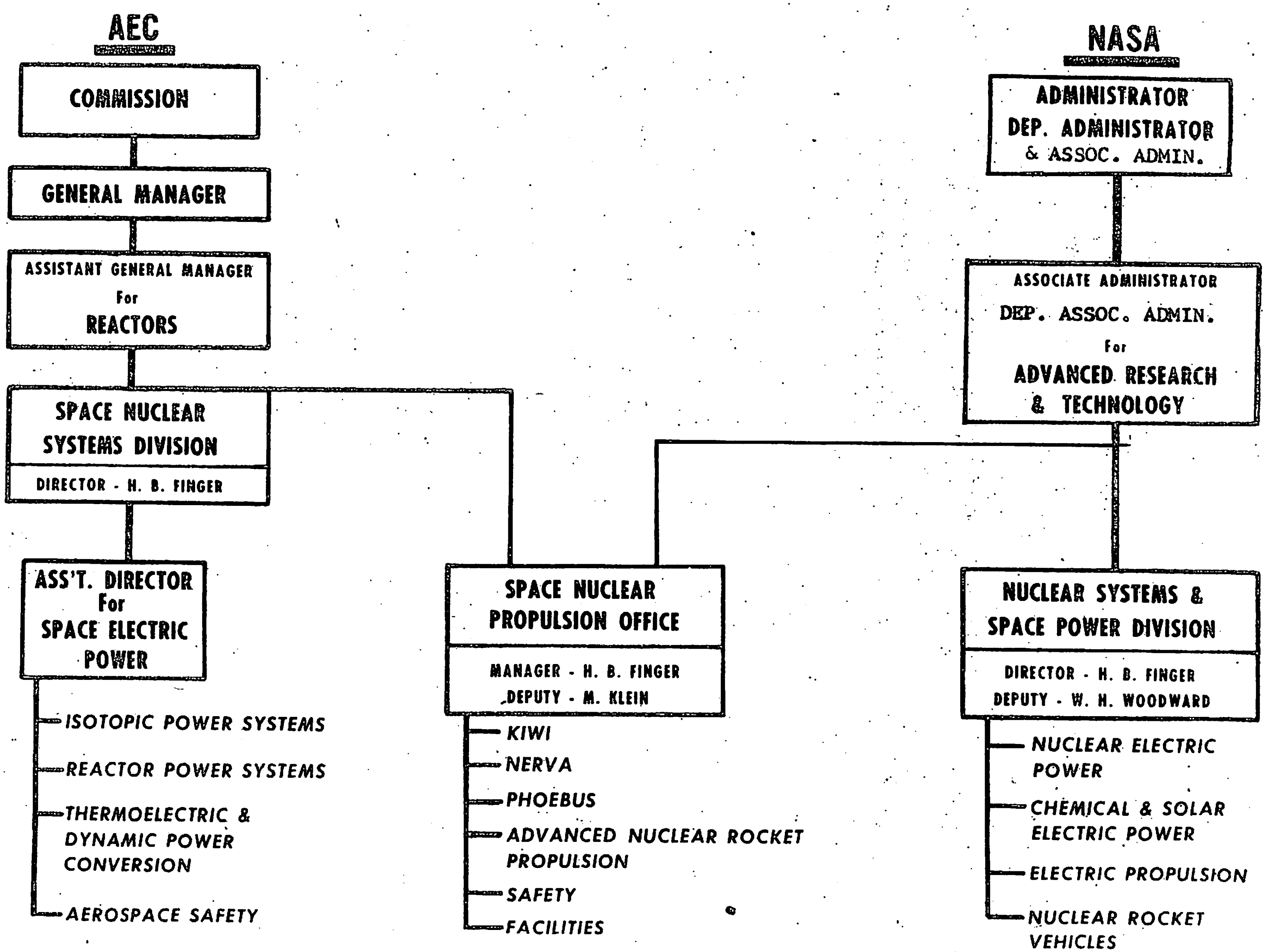
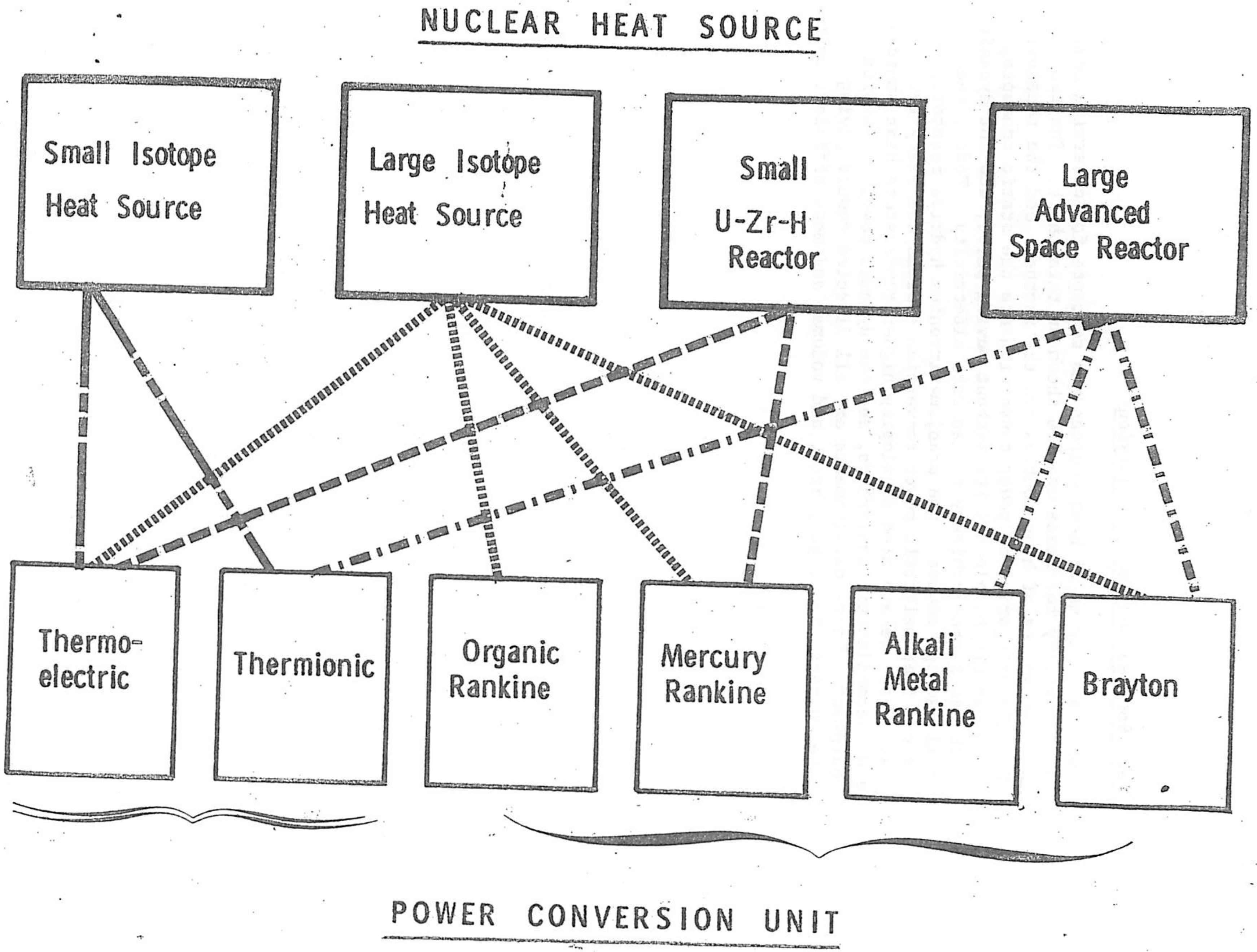
SUMMARY AND INTRODUCTION

ORGANIZATION OF SPACE NUCLEAR SYSTEMS PROGRAMS



Heat Sources and Power Conversion

The several types of heat sources and concepts for coverting heat to electricity for space use are shown on this chart. The two main categories of heat sources are the isotopes and the reactors. The main categories of power conversion are the static concepts, which convert to electricity without moving parts, and the dynamic, which use turbo-machinery to generate electricity. Today, the small isotope sources, the uranium-zirconium hydride reactors, and the thermoelectric power conversion systems are ready for space use. We are also developing higher temperature heat sources and thermoelectric converters, and the dynamic power conversion equipment. These developments are all directed toward giving more compact (in weight, area, and volume) and more efficient systems.


NUCLEAR HEAT SOURCE
Small Isotope Heat Source
Large Isotope Heat Source
Small U-Zr-H Reactor
Large Advanced Space Reactor
Thermo-electric
Thermionic
Organic Rankine
Mercury Rankine
Alkali Metal Rankine
Brayton
POWER CONVERSION UNIT

PROBABLE POWER RANGE OF SYSTEMS OF PRINCIPAL INTEREST

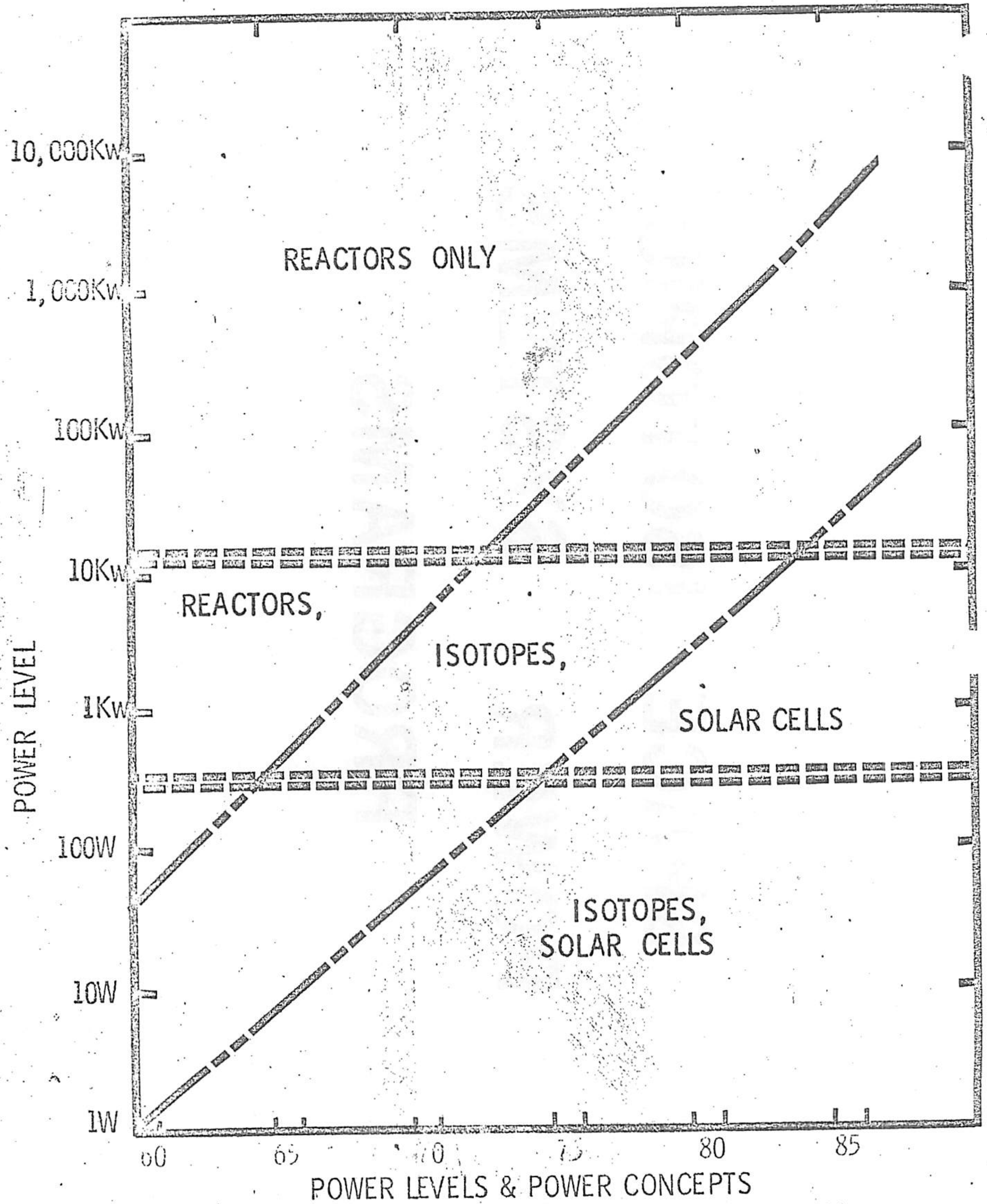
	PROBABLE POWER RANGE IN KW				
SYSTEM	0.01-2.0	2 - 10	10-25	25-50	>50
SC (Solar Cells)	✓	✓	✓	✓	
ITE (Isotope T/E)	✓				
IB (Isotope Brayton)		✓			
RTE (Reactor T/E)			✓		
RR (Reactor Rankine)				✓	✓
RTI (Reactor Thermionic)					✓
RB (Reactor Brayton)					✓

SPACE POWER SYSTEMS

SYSTEM	CONCEPT/ FUEL	POWER	WATTS/ #	DESIGN LIFETIME	TEST STATUS
SNAP 3A	ITE/Pu^{238}	2.7 w	0.6	1 yr.	Launched 6/61, unit still powering radio 11/61 unit shorted out 6/62.
SNAP 9A	ITE/Pu^{238}	25 w	1.0	5 yrs.	Launched 9/63, 12/63, powered satellites 4/64 aborted during launch.
SNAP 11	ITE/Cm^{242}	25 w	1.0	90 days	Surveyor Reqt. cancelled 1965. Fueled test at Oak Ridge 1966.
SNAP 19	ITE/Pu^{238}	30 w	1.0	5 yrs.	Electrically heated unit tested. Launch scheduled late 67.
SNAP 27	ITE/Pu^{238}	56 w	1.8	5 yrs.	Electrically heated tests start fall 1966. Flight unit delivery 4/67.
SNAP 29	ITE/Po^{210}	400 w	1.0	90 days	Hardware development being started. Tentative launch 1969.
SNAP 10A	RTE/U^{235}	500 w	.7*	1 yr.	Launched 4/65, operated 43 days until satellite failure. Ground test 10,000 hrs.
SNAP 8 TE	RTE/U^{235}	to 25 kw.	2-3*	1-2 yrs.	Components under development; use by early 1970's.
SNAP 8	R HgR/U^{235}	35 kw.	4*	10,000 hrs.	System under development for use mid-1970's.
ADVANCED ISOTOPES	I---	to 11 kw.	8-10*	-	Component research started.
ADVANCED REACTORS	R---	to mega-watts	50*	-	Component research started.

* Unshielded

SPACE ISOTOPIC POWER SYSTEMS PROGRAMS

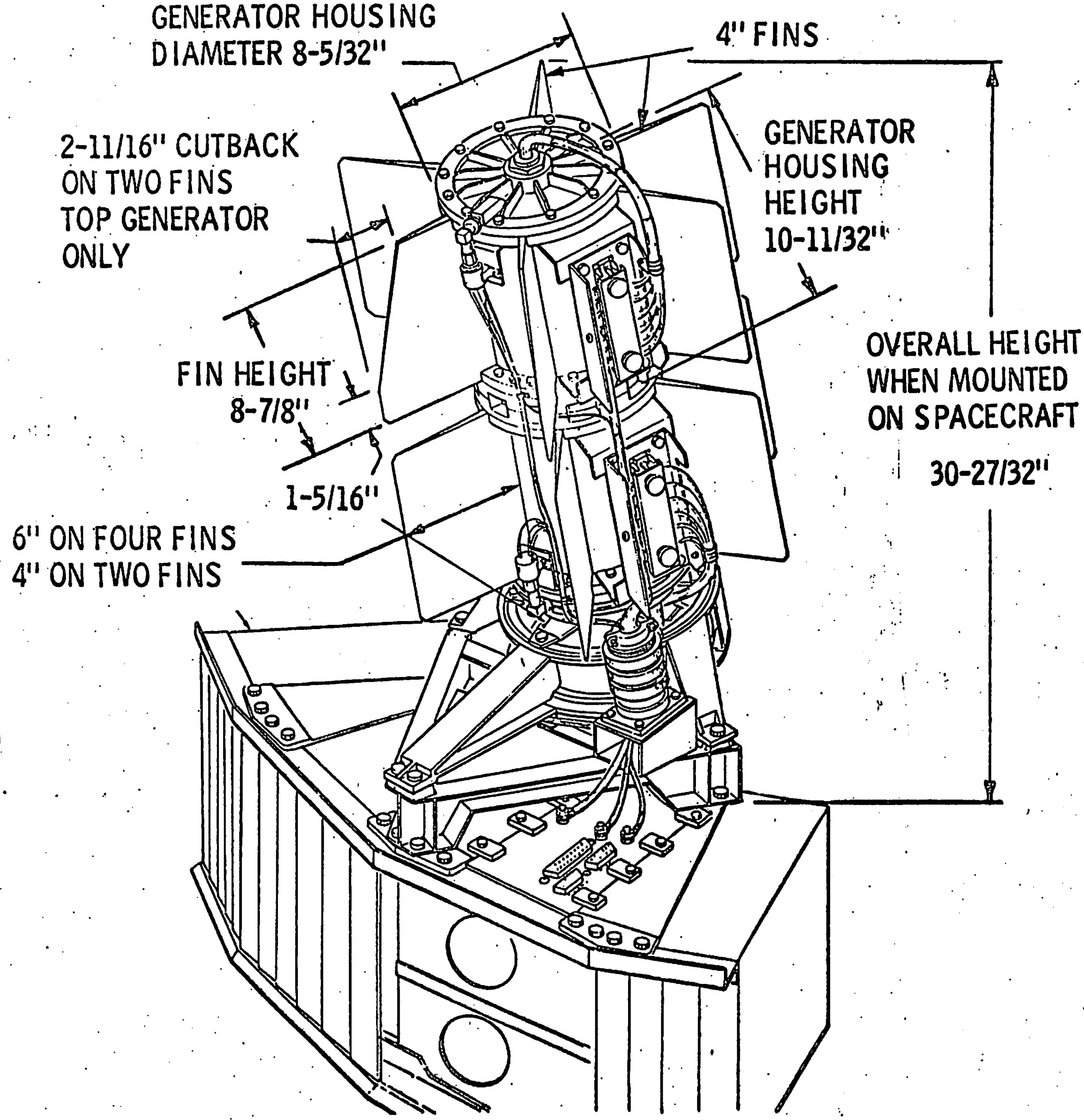


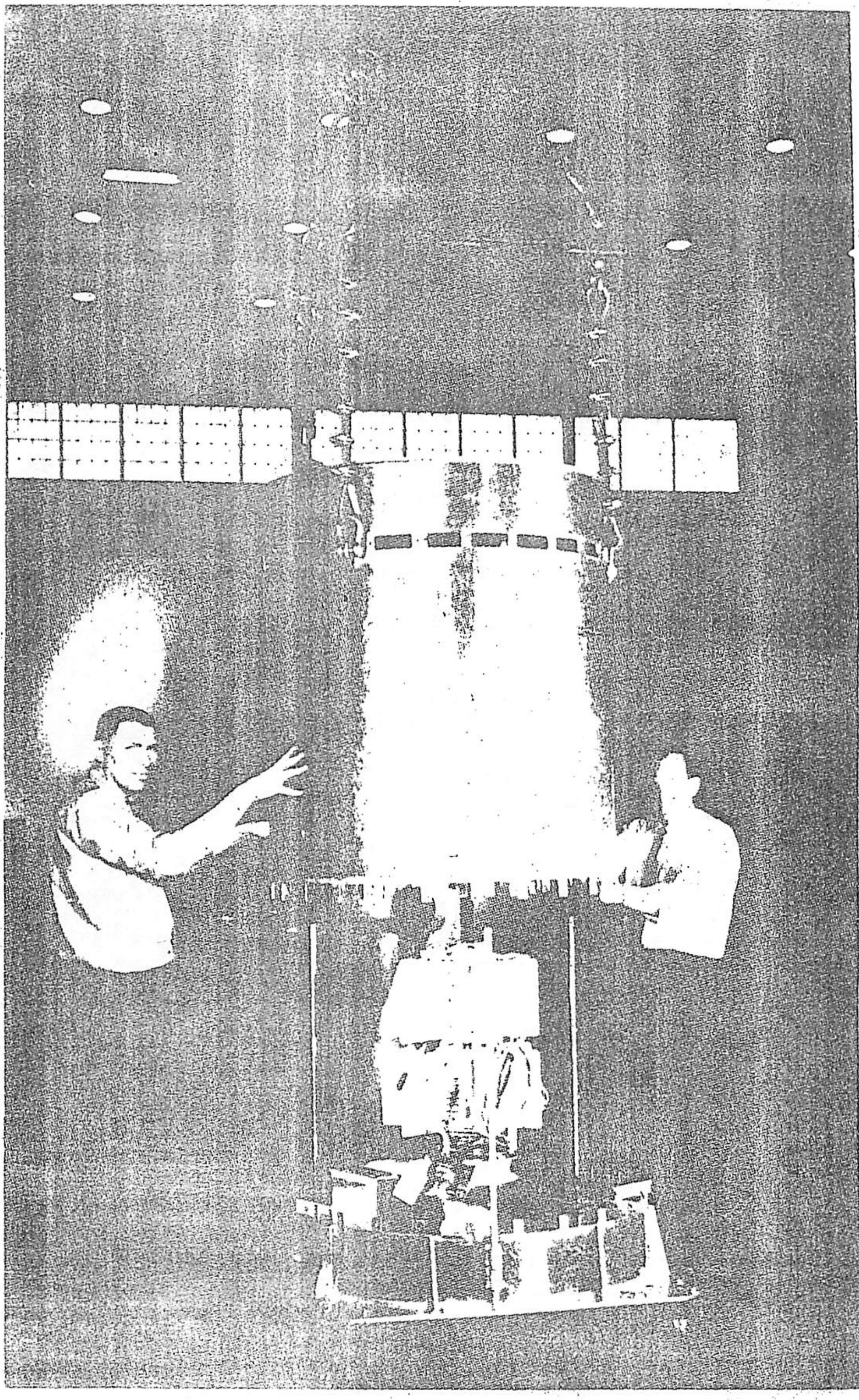
POWER LEVELS & POWER CONCEPTS
Requirements by Calendar Years

SNAP 19 (PHASE III) MAJOR MILESTONES

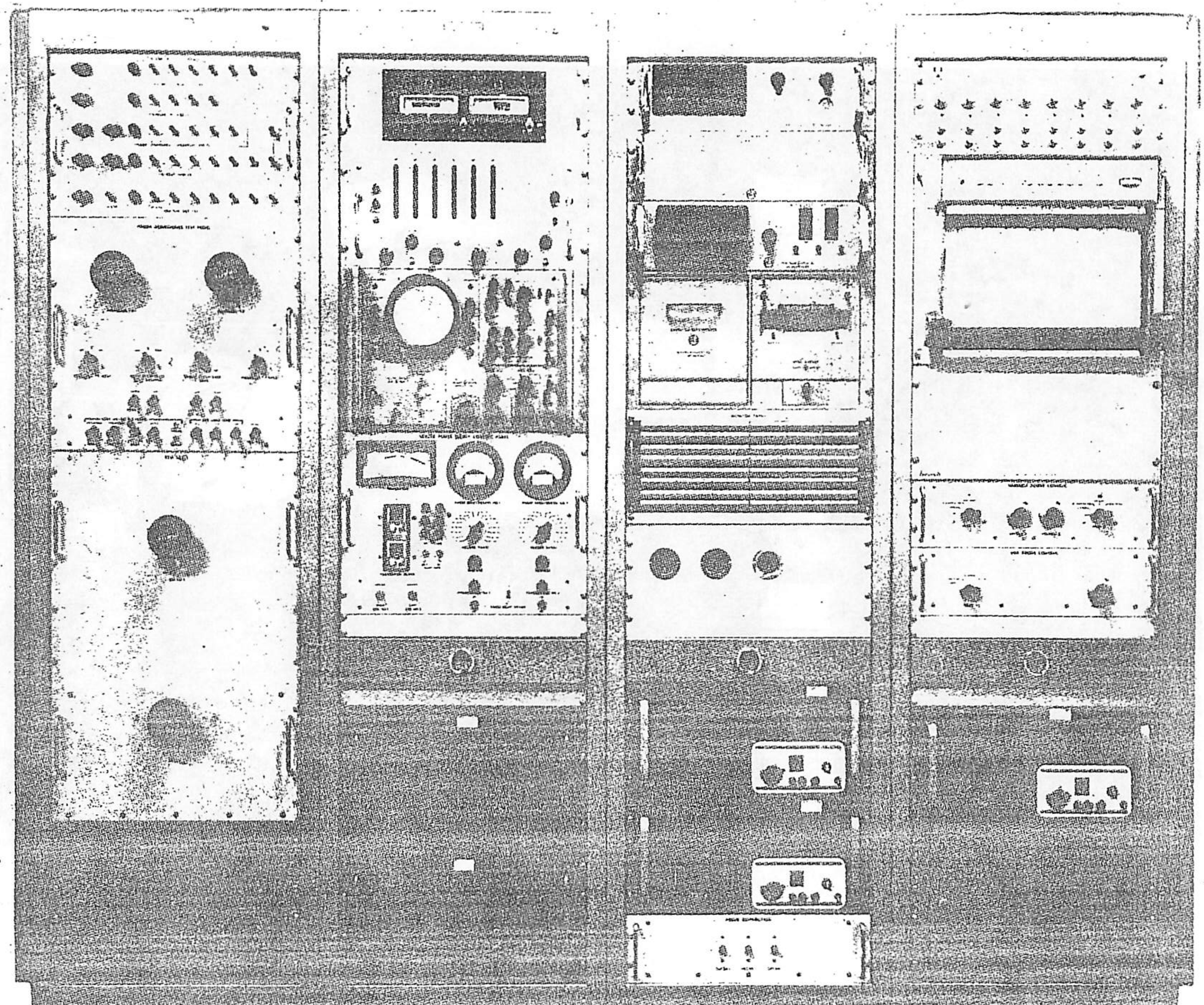
	SCHEDULED DATE
ELECTRICALLY HEATED PROTOTYPE (ENGINEERING MODEL) SHIPPED TO GE	7-5-66
FIRST SET FUEL CAPSULES SHIPPED FROM MOUND LABORATORY	7-23-66
GENERATOR SUBSYSTEM NO. 4 FUELED	8-12-66
SECOND SET FUEL CAPSULES SHIPPED FROM MOUND LABORATORY	9-23-66
AEC SYSTEM NO. 4 (FUELED PROTOTYPE) QUALIFICATION TESTING COMPLETE	10-1-66
AEC SYSTEM NO. 5 (ELECTRICALLY HEATED PROTOTYPE) AVAILABLE AS ENGINEERING MODEL BACKUP	10-14-66
SAFETY EVALUATION REPORT (SAR) DRAFT SUBMITTED	12-31-66
AEC SYSTEM NO. 6 (FLIGHT MODEL FUELED) FLIGHT QUALIFIED	1-6-67
AEC SYSTEM NO. 6 SHIPPED TO GE	2-1-67
AEC SYSTEM NO. 7 (FLIGHT BACKUP-FUELED) AVAILABLE FOR DELIVERY	4-1-67

SNAP 19 PRINCIPAL DIMENSIONS





SHIPPING CONTAINER AND HANDLING EQUIPMENT



GROUND SUPPORT TEST CONSOLE

SNAP-27 GENERATOR

PROGRAM STATUS 7/1/66

The SNAP-27 lunar surface generator development program is aimed at providing a highly reliable 50 watt(e) power source for the NASA Apollo Lunar Surface Experiments Package (ALSEP) program. The program was initiated in August 1965 on a generator technology development basis. Based on a NASA requirement to use SNAP-27 on ALSEP, the program was re-directed in scope to meet the NASA need.

To date several major milestones have been achieved.

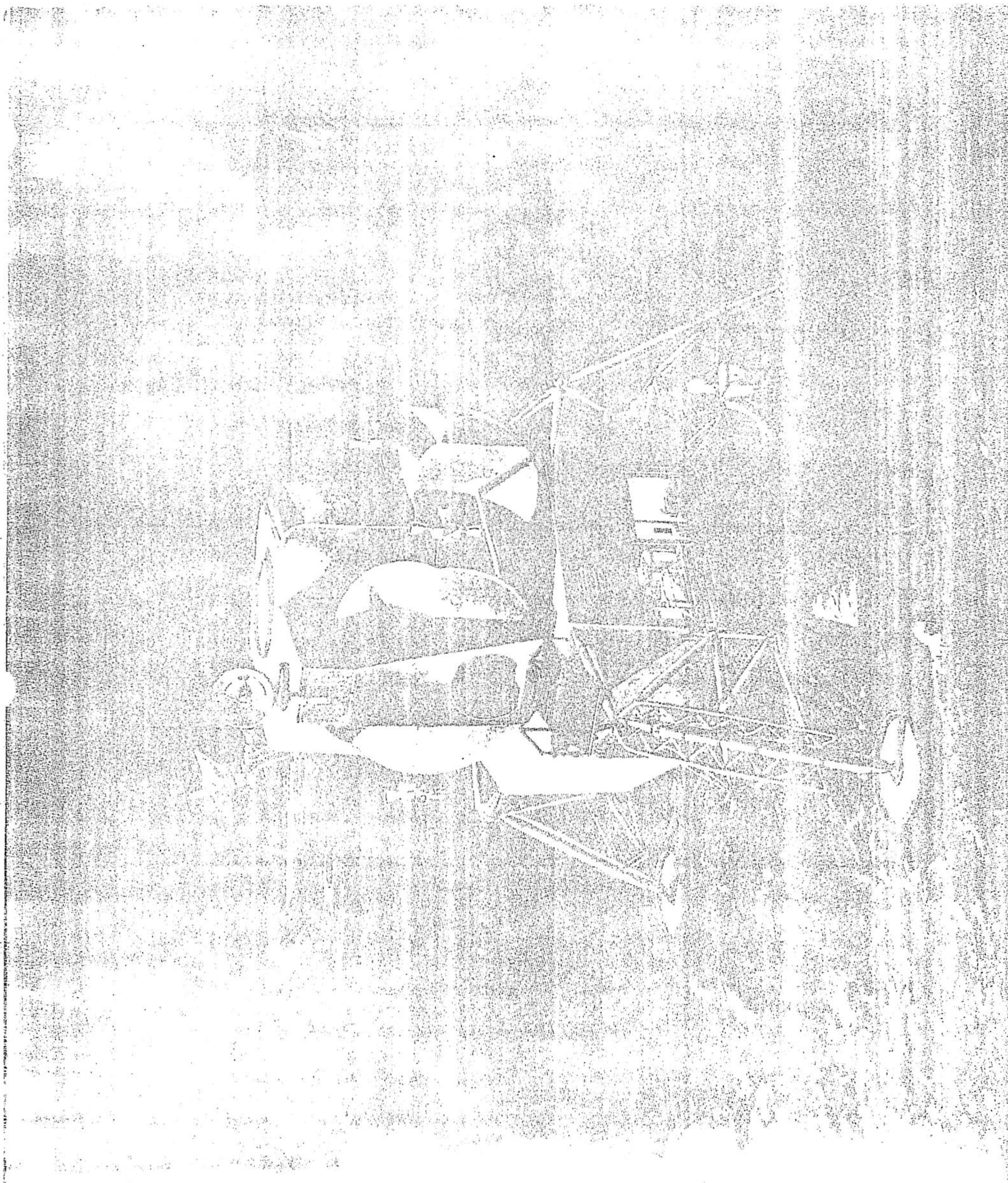
1. Hermetic closure of the Iconel 101 (hot frame) to beryllium (cold case) has been demonstrated.
2. Four ten (10) couple lead telluride thermoelectric modules using flight hardware type material have been placed on test.
3. Final design of the engineering development generators has been released.
4. Precision machining of the all beryllium cold frame, outer case and fins has been accomplished.
5. High temperature brazing of the beryllium fins to the outer case has been verified.
6. Application of a good adhesive emissivity coating to the outer case has been achieved.
7. Design and aerodynamic tests have been completed on the separate fuel cask which will carry the fuel capsule.

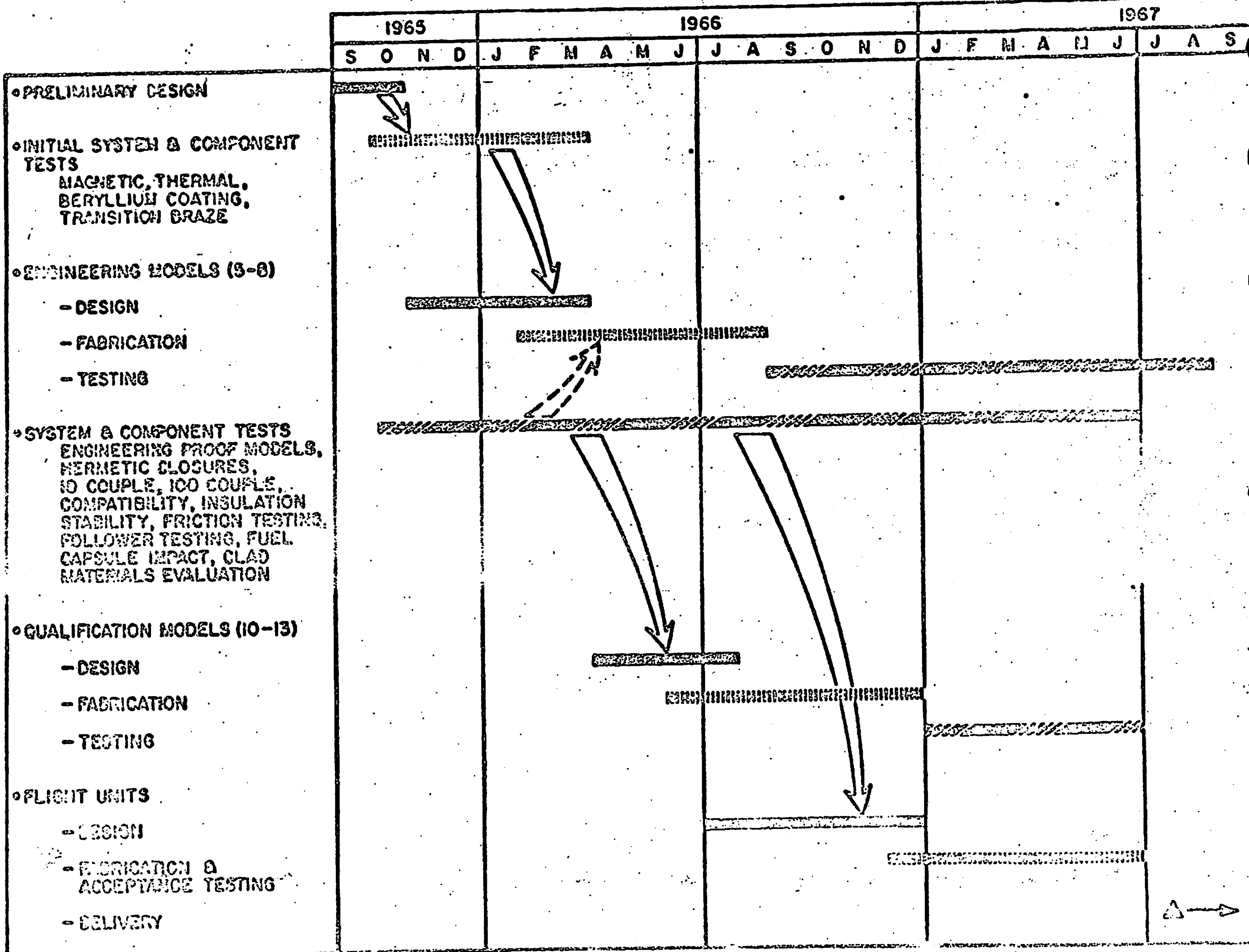
Delivery of the first electrically heated and fueled system is scheduled to NASA in January 1967. Also, delivery of a flight system and back-up flight schedule is planned for July 1967.

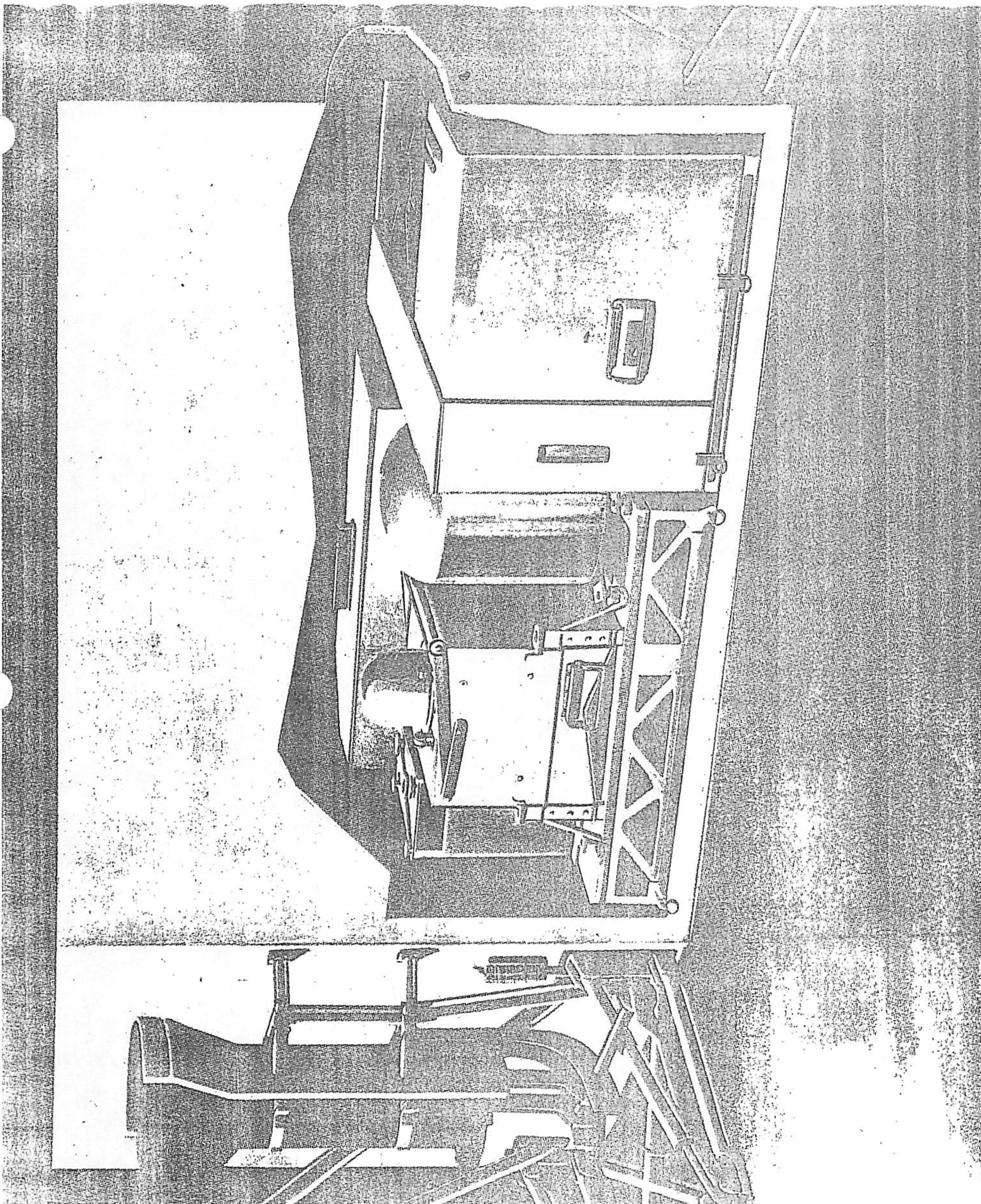
Because of the potential use of this type of generator in earth orbital missions, a study is in progress to determine the reentry characteristics and behavior of a fueled SNAP-27 generator system to withstand orbital reentry heating thereby enabling the fuel to reenter intact.

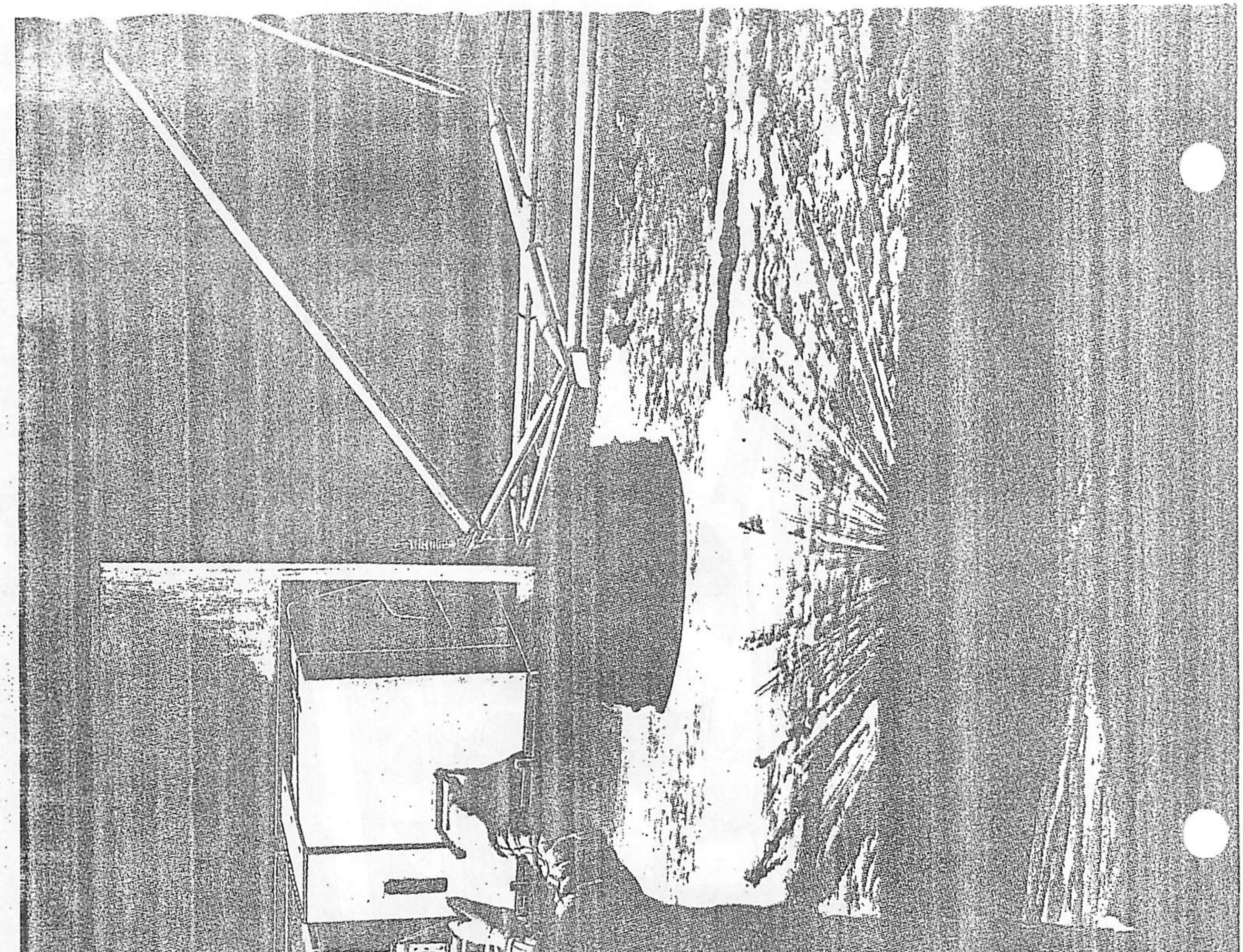
SNAP-27 POWER SUPPLY DESIGN AND PERFORMANCE CHARACTERISTICS

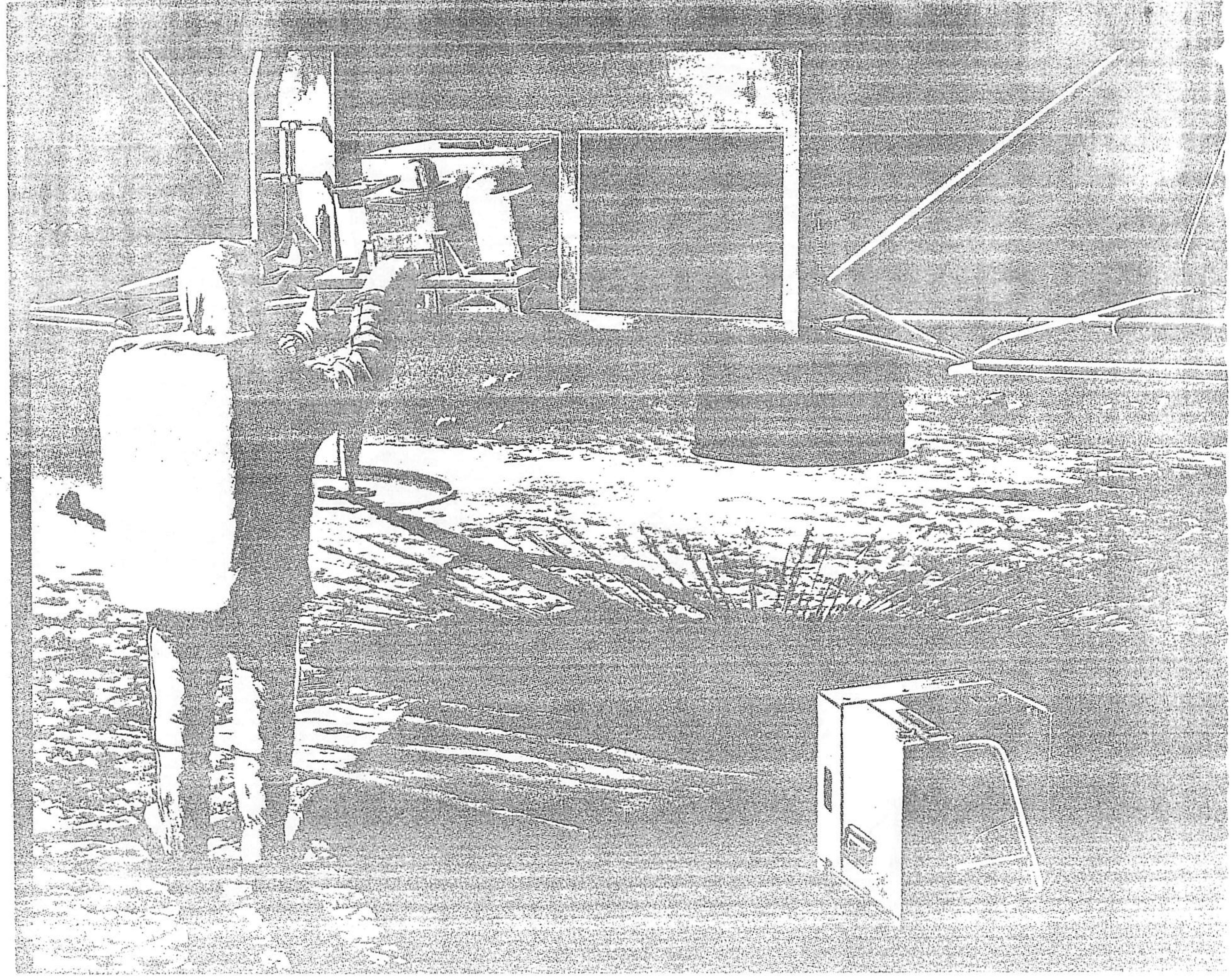
Design Life	One year lunar operation preceded two years earth storage; .995 reliability
Generator Performance	
Output Power (min.)	56 watts (end of life)
Output Voltage	14 volts DC
Current	2 amps in each of two strings
Overall Efficiency (nominal)	4%
Max. Hot Junction Temp.	1100°F (lunar day)
Max. Cold Junction Temp.	525°F (lunar day)
Fuel Capsule Thermal Output	1500 watts
Max. Fuel Clad Temp.	1390°F
Generator Design Characteristics	
Materials of Construction	Beryllium and IN-102 structure
Overall Generator Dia. (over fins)	15.7 in.
Overall Generator Length	18.1 in.
No. of Fins	8
Fin Radial Length	5.0 in.
Heat Source Characteristics	
No. of Fuel Capsules	1 (2 Compartments)
Capsule O.D.	2.50 in.
Capsule Length	15.6 in.
Total Dose Rate at 1 Meter	69 mrem/hour

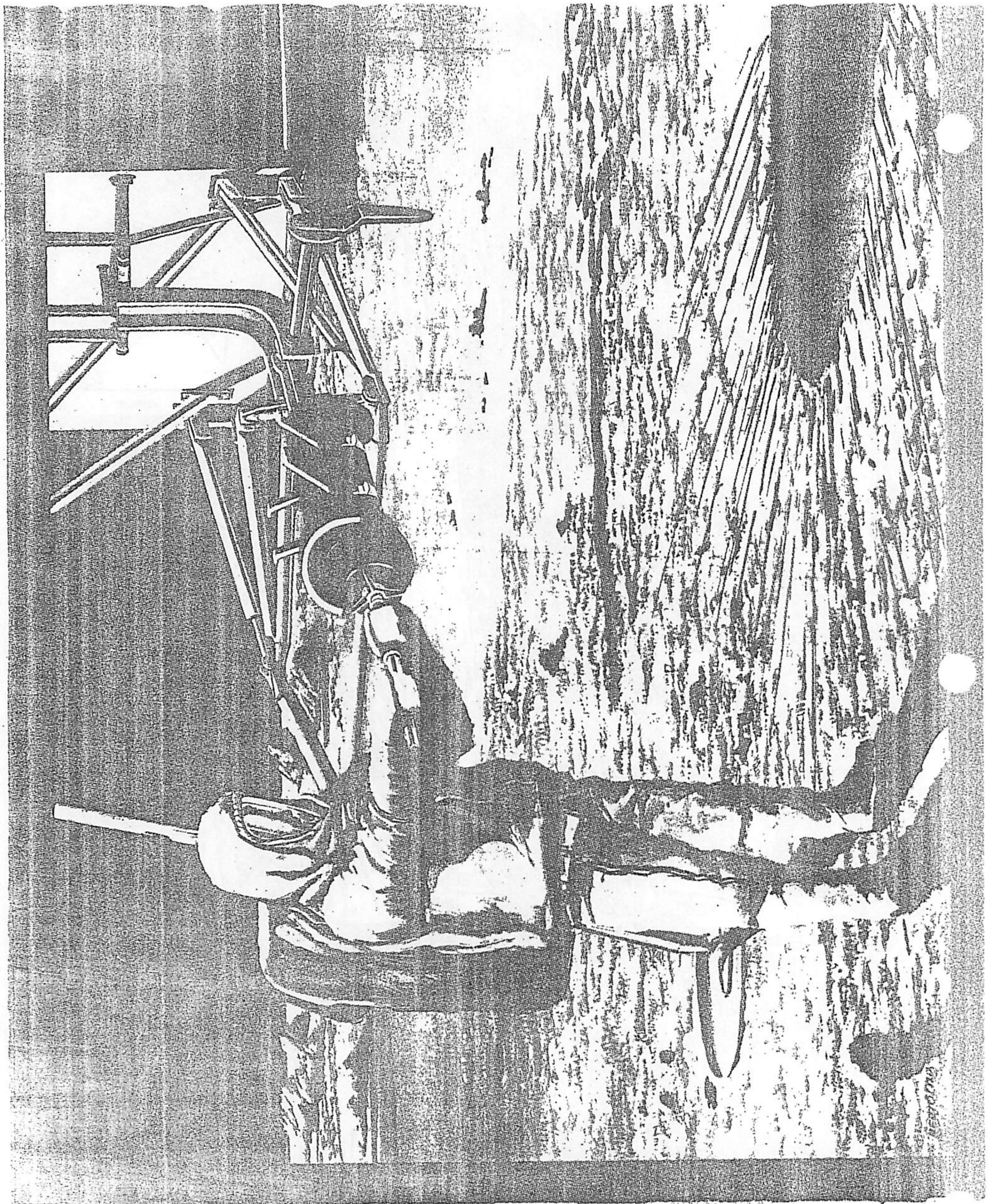


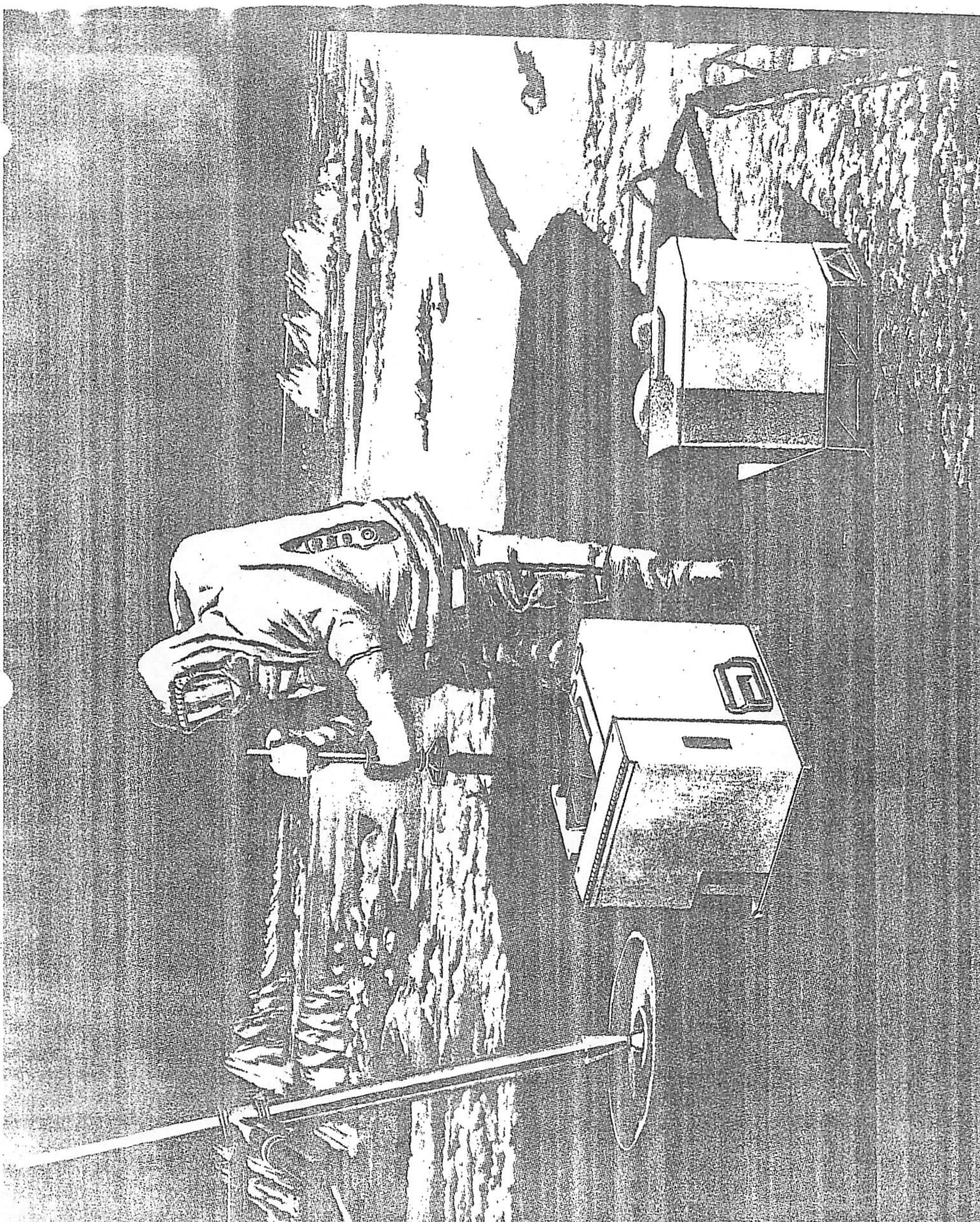

1965
1966
1967
S O N D J F M A M J J A S O N D J F M A M J J A S
• PRELIMINARY DESIGN
• INITIAL SYSTEM & COMPONENT TESTS
MAGNETIC, THERMAL, BERYLLIUM COATING, TRANSITION BRAZE
• ENGINEERING MODELS (5-8)
– DESIGN
– FABRICATION
– TESTING
• SYSTEM & COMPONENT TESTS
ENGINEERING PROOF MODELS, HERMETIC CLOSURES, 10 COUPLE, 100 COUPLE, COMPATIBILITY, INSULATION STABILITY, FRICTION TESTING, FOLLOWER TESTING, FUEL CAPSULE IMPACT, CLAD MATERIALS EVALUATION
• QUALIFICATION MODELS (10-13)
– DESIGN
– FABRICATION
– TESTING
• FLIGHT UNITS
– FABRICATION & ACCEPTANCE TESTING
– DELIVERY

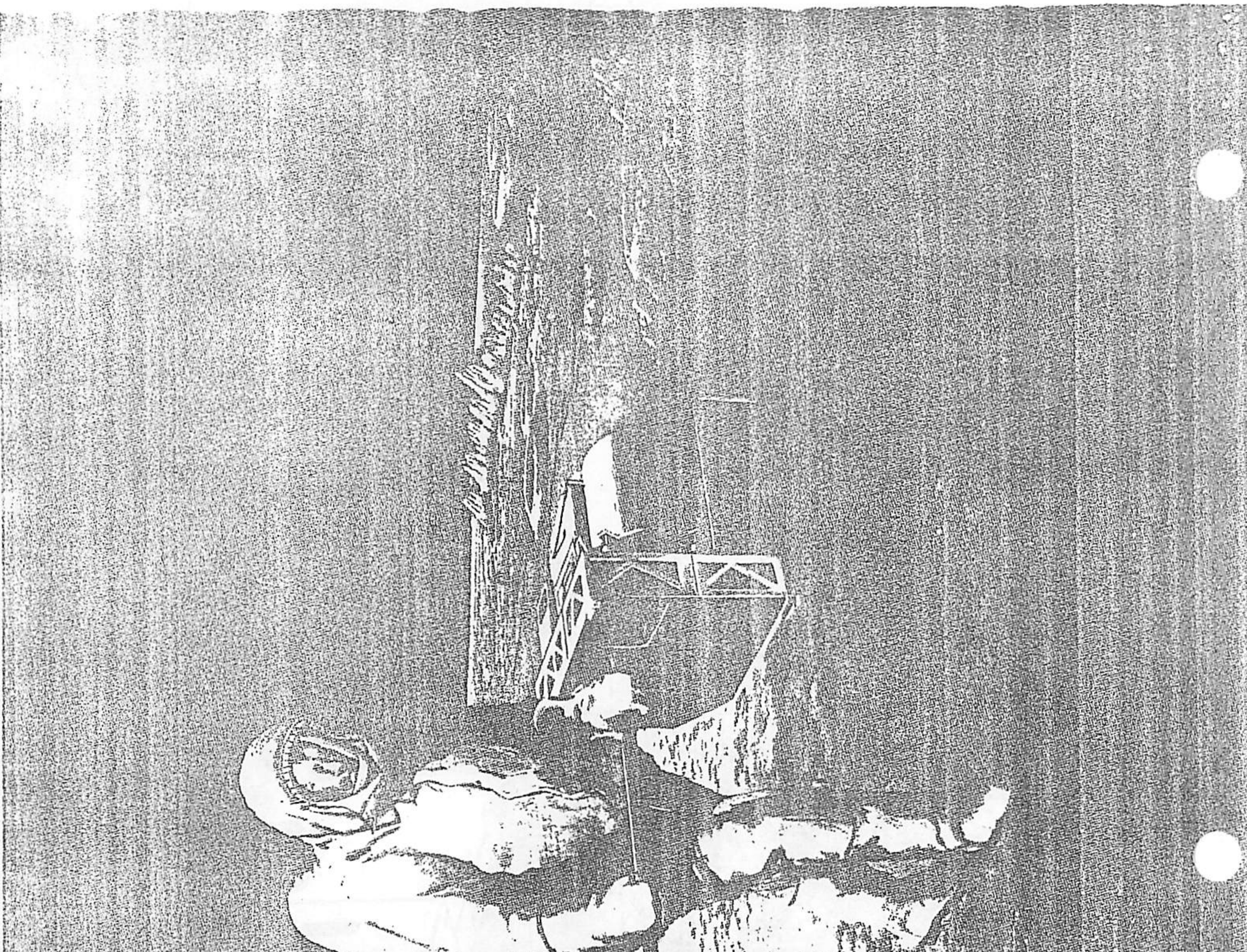


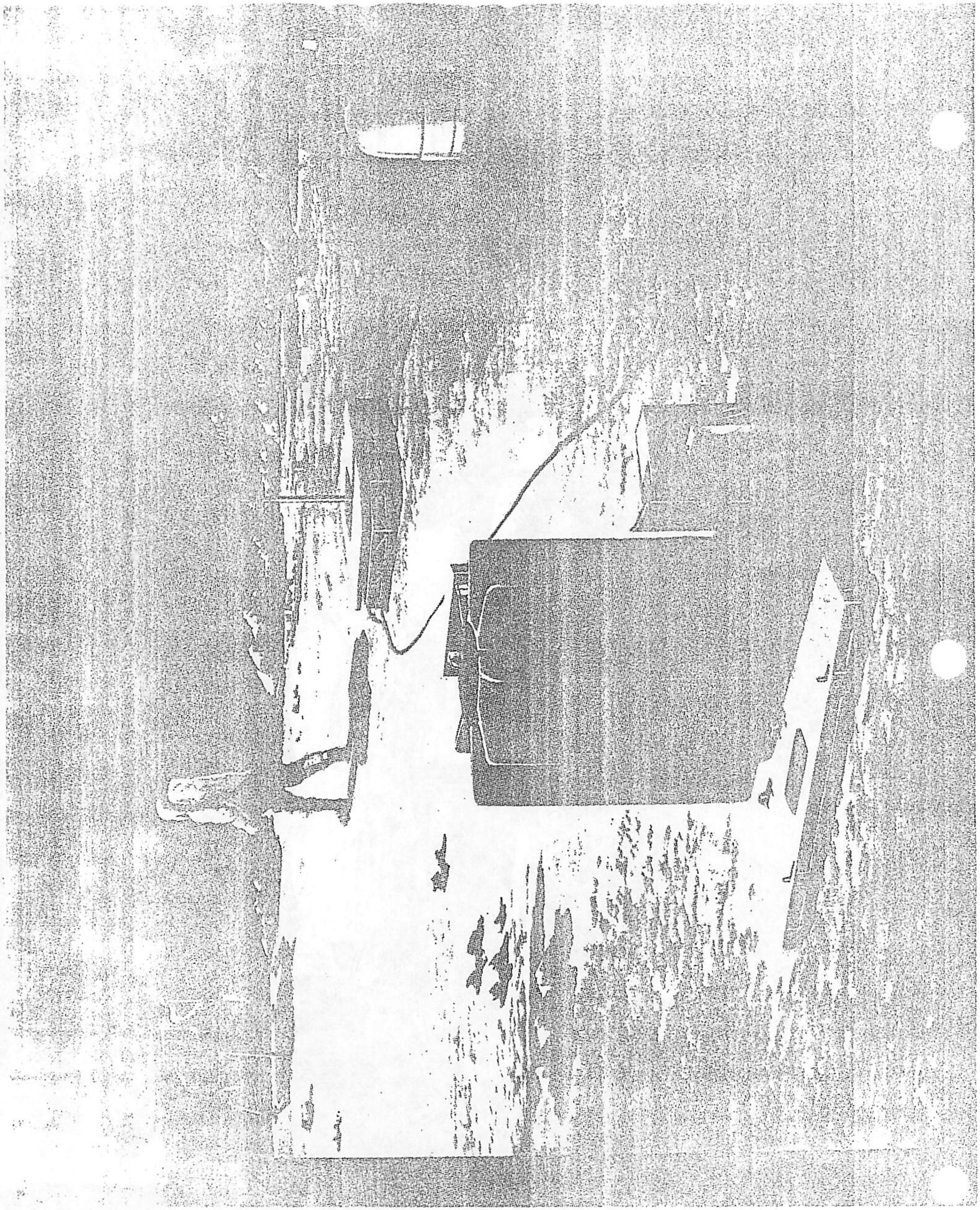


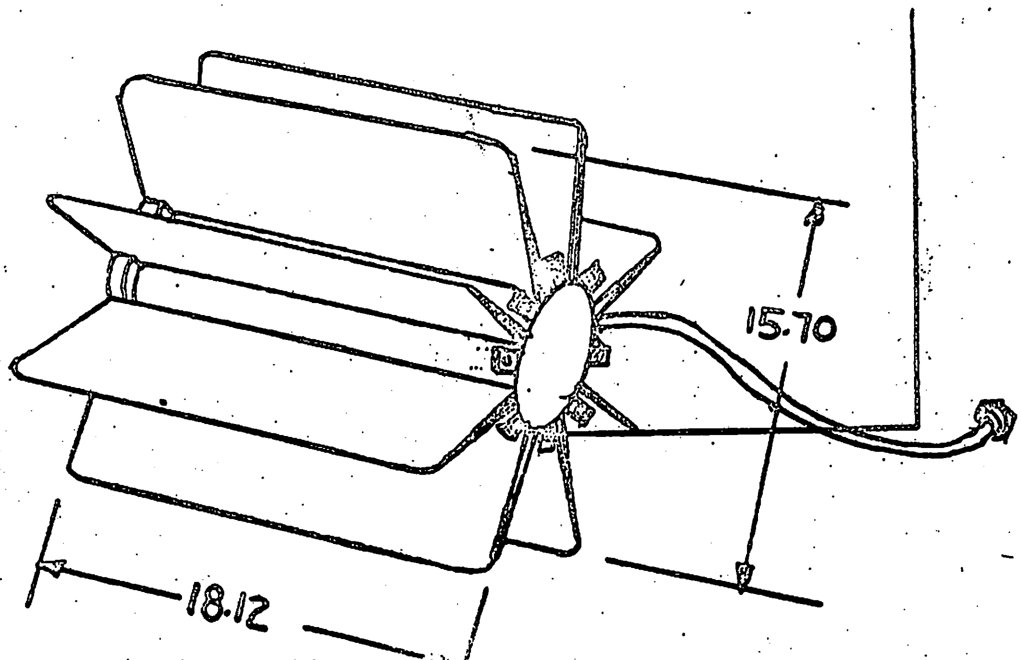




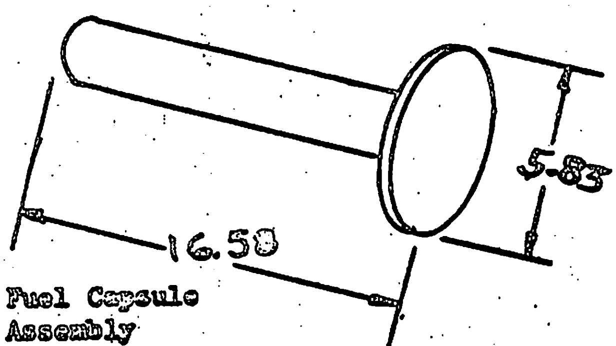




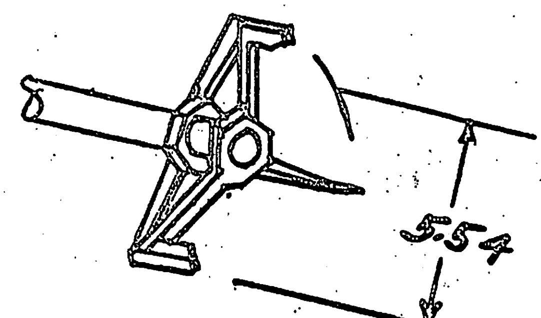




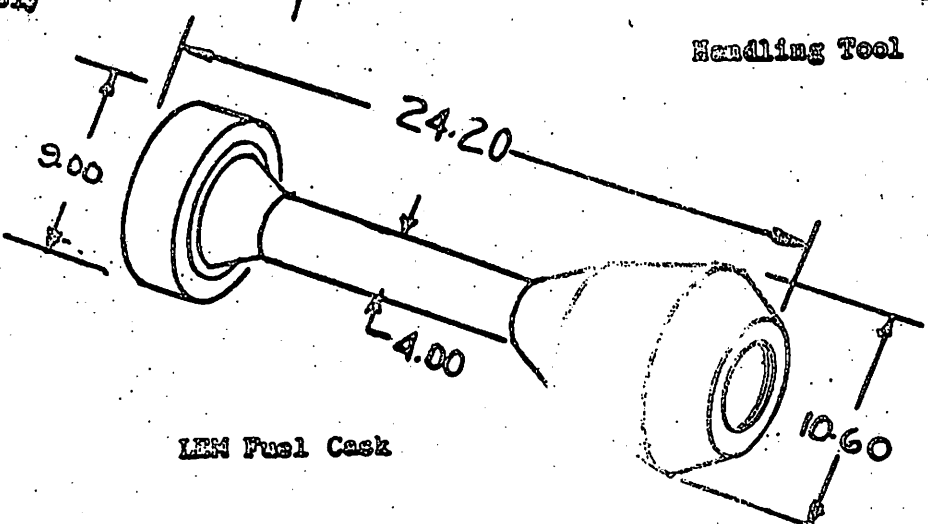
Generator Assembly



Fuel Capsule Assembly



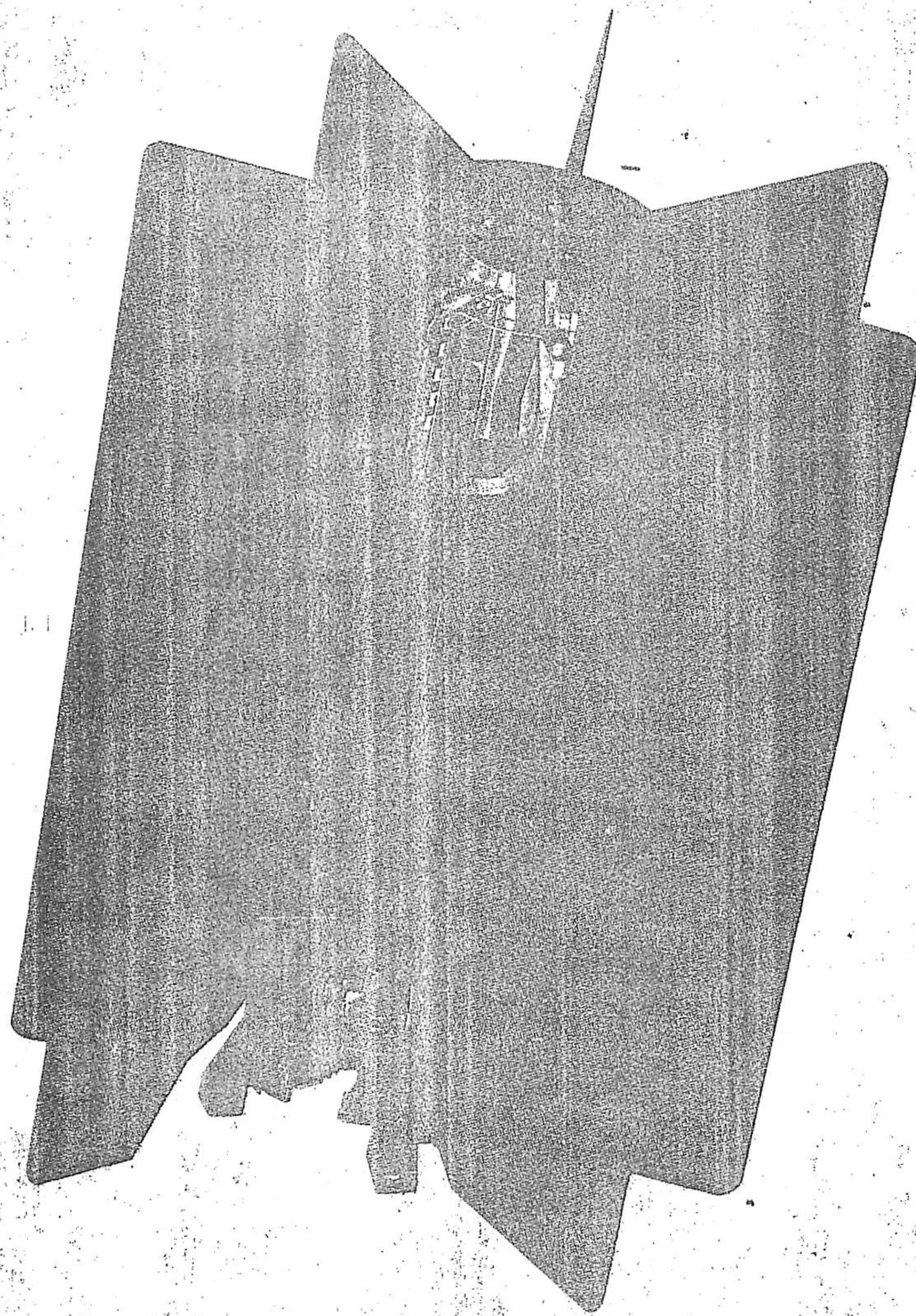
Handling Tool

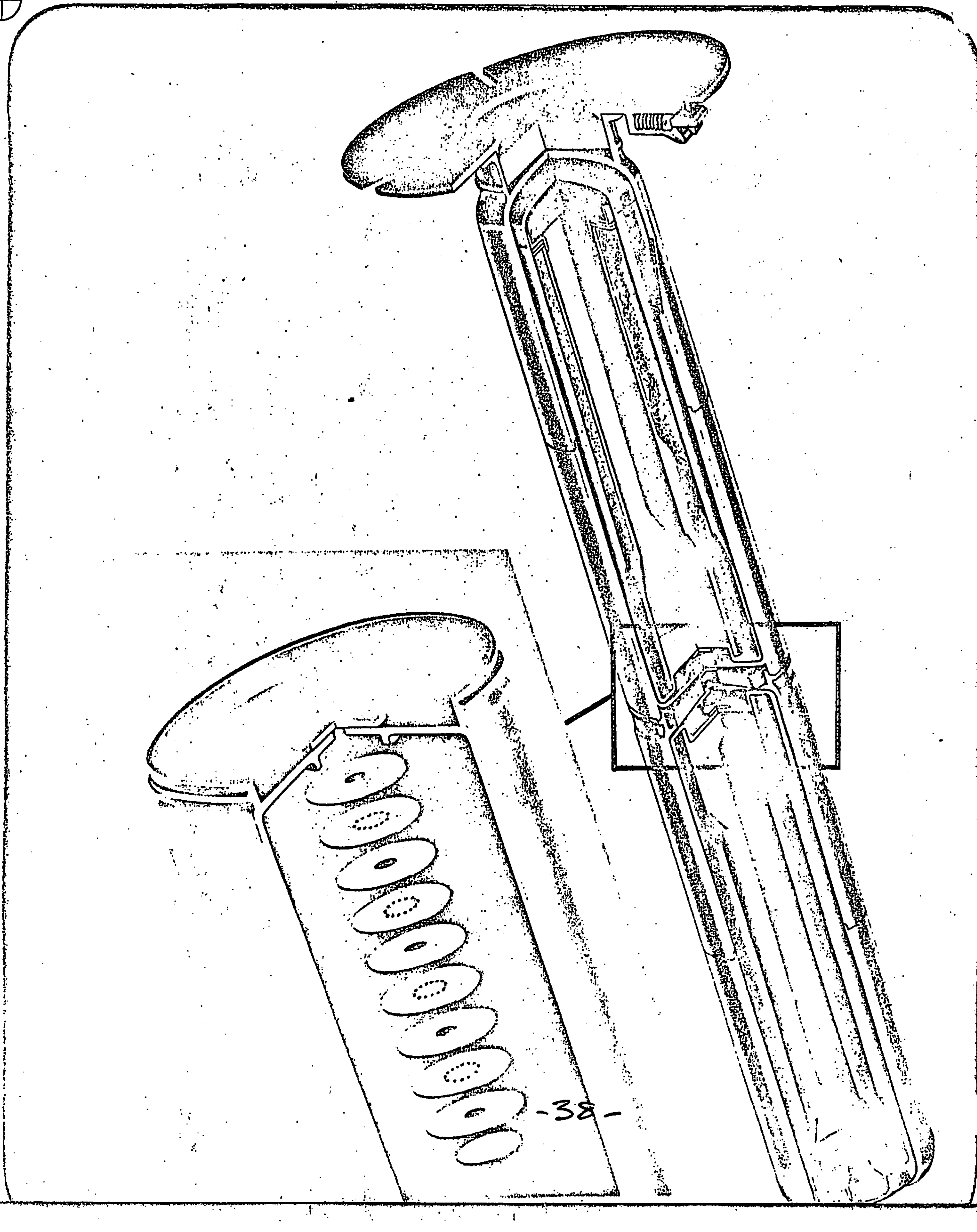


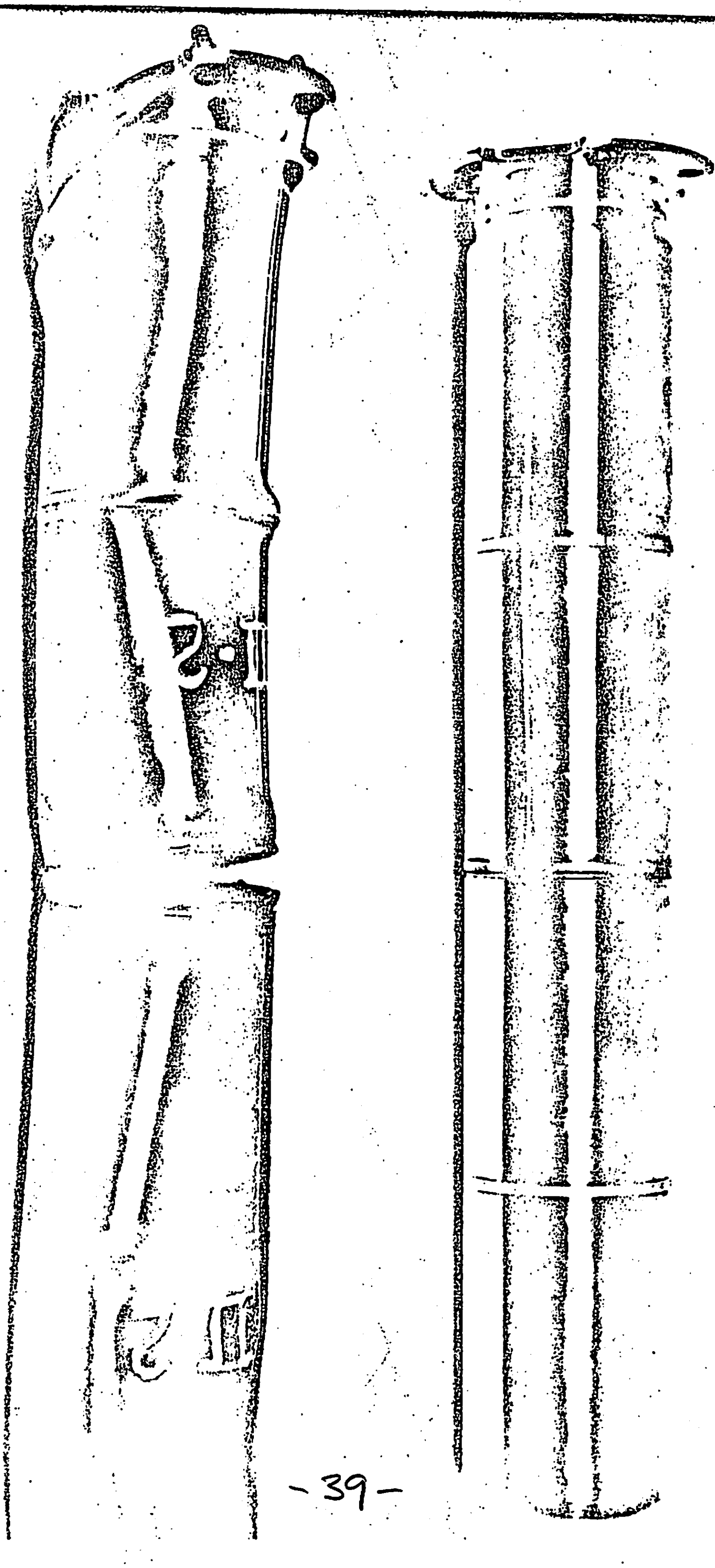
LEM Fuel Cask

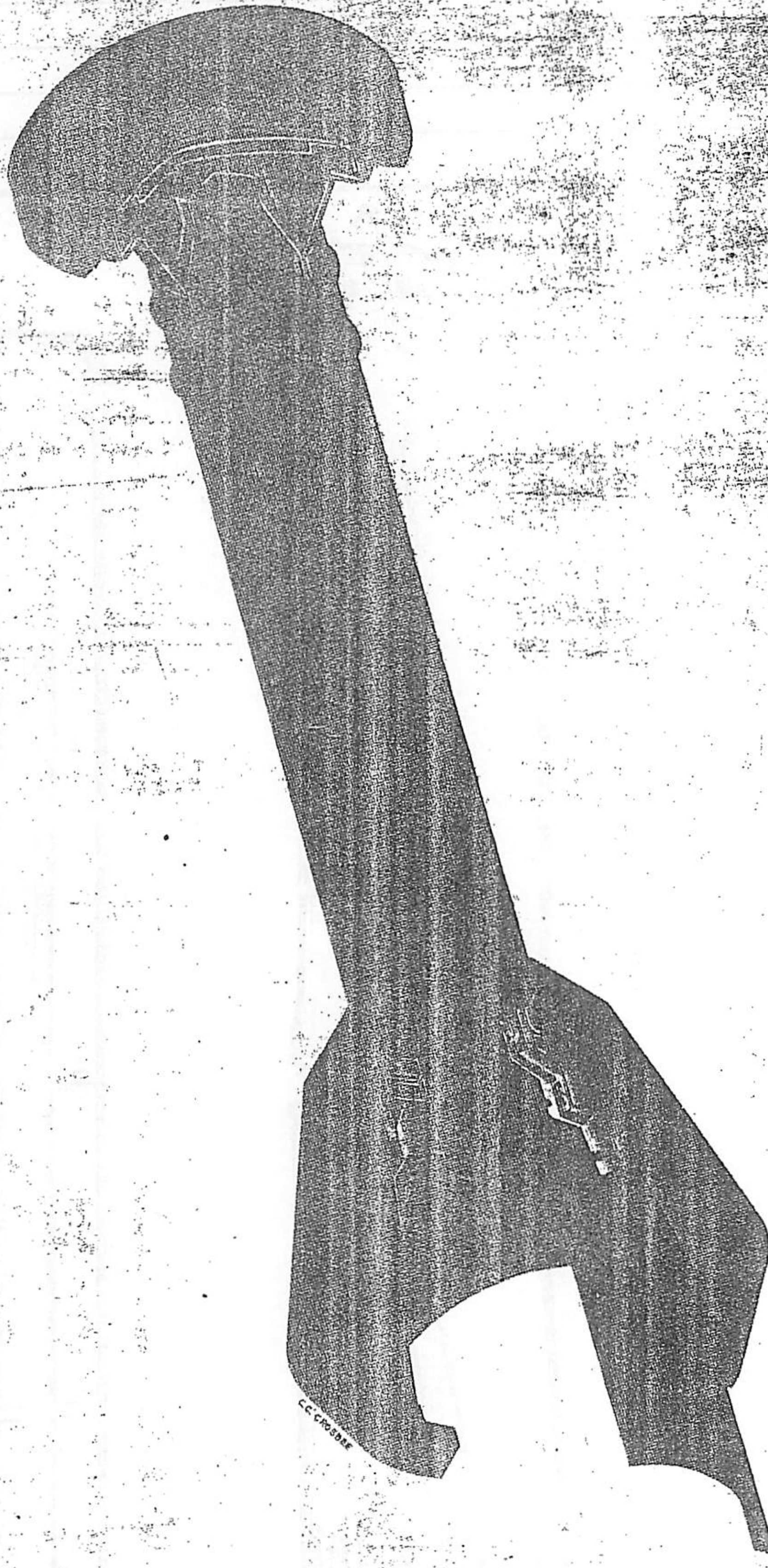
SNAP-27 WEIGHTS

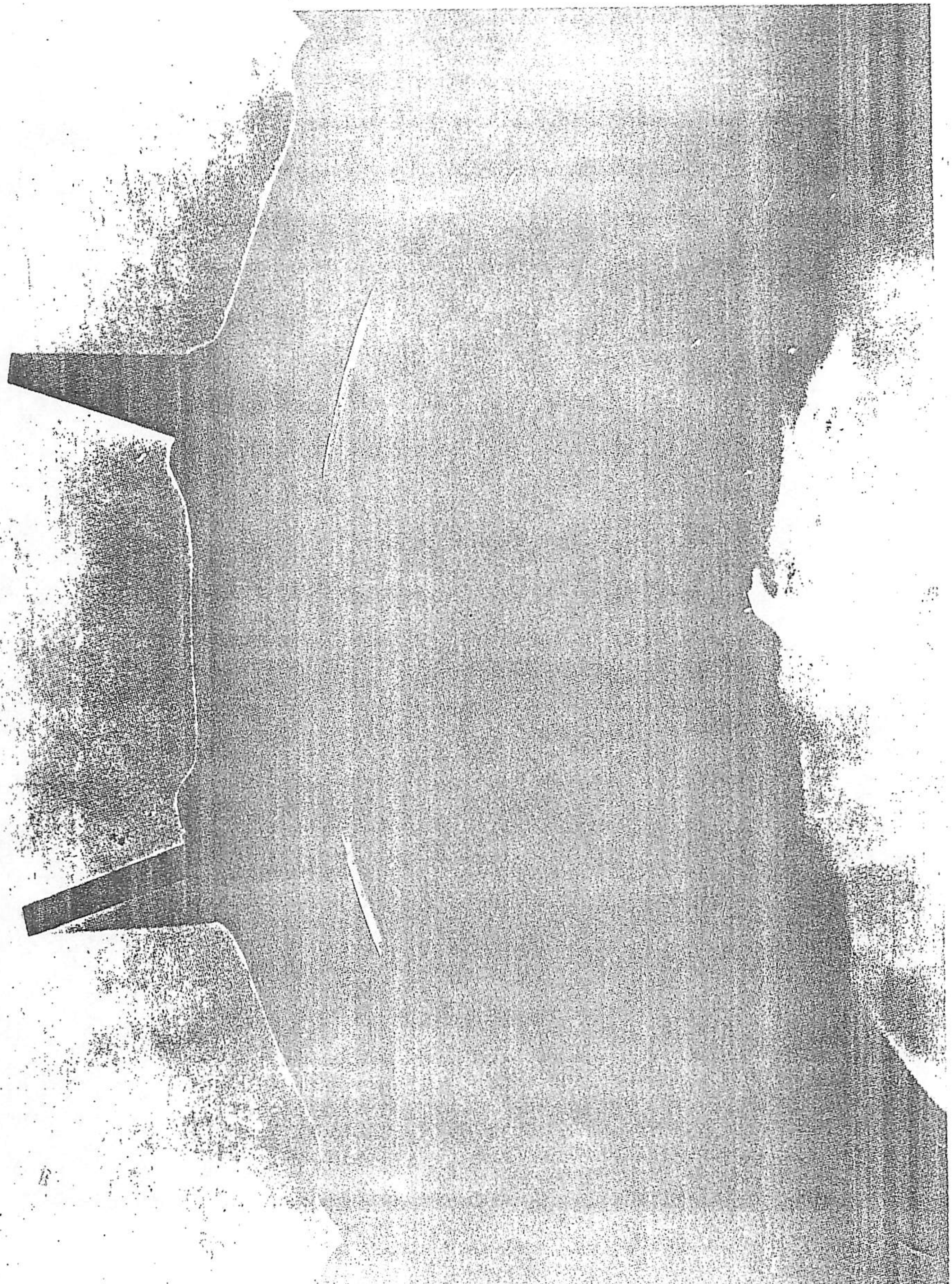
⊙ GENERATOR ASSEMBLY (Less Capsule)	24.6 lbs.
⊙ FUEL CAPSULE ASSEMBLY	13.5
⊙ LEM FUEL CASK ASSEMBLY	7.5
⊙ FLIGHT HANDLING TOOL	0.6
	46.2 lbs.

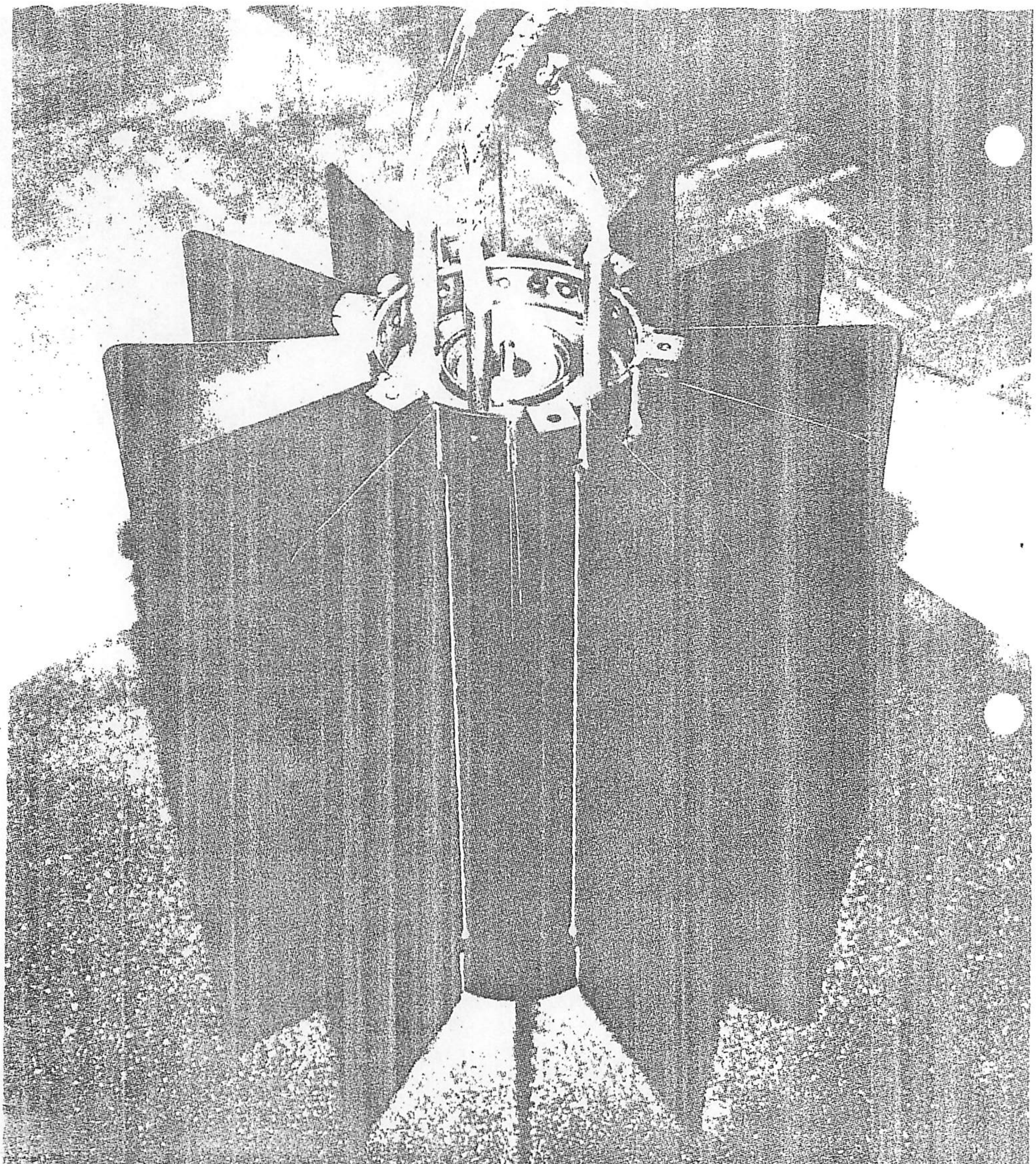




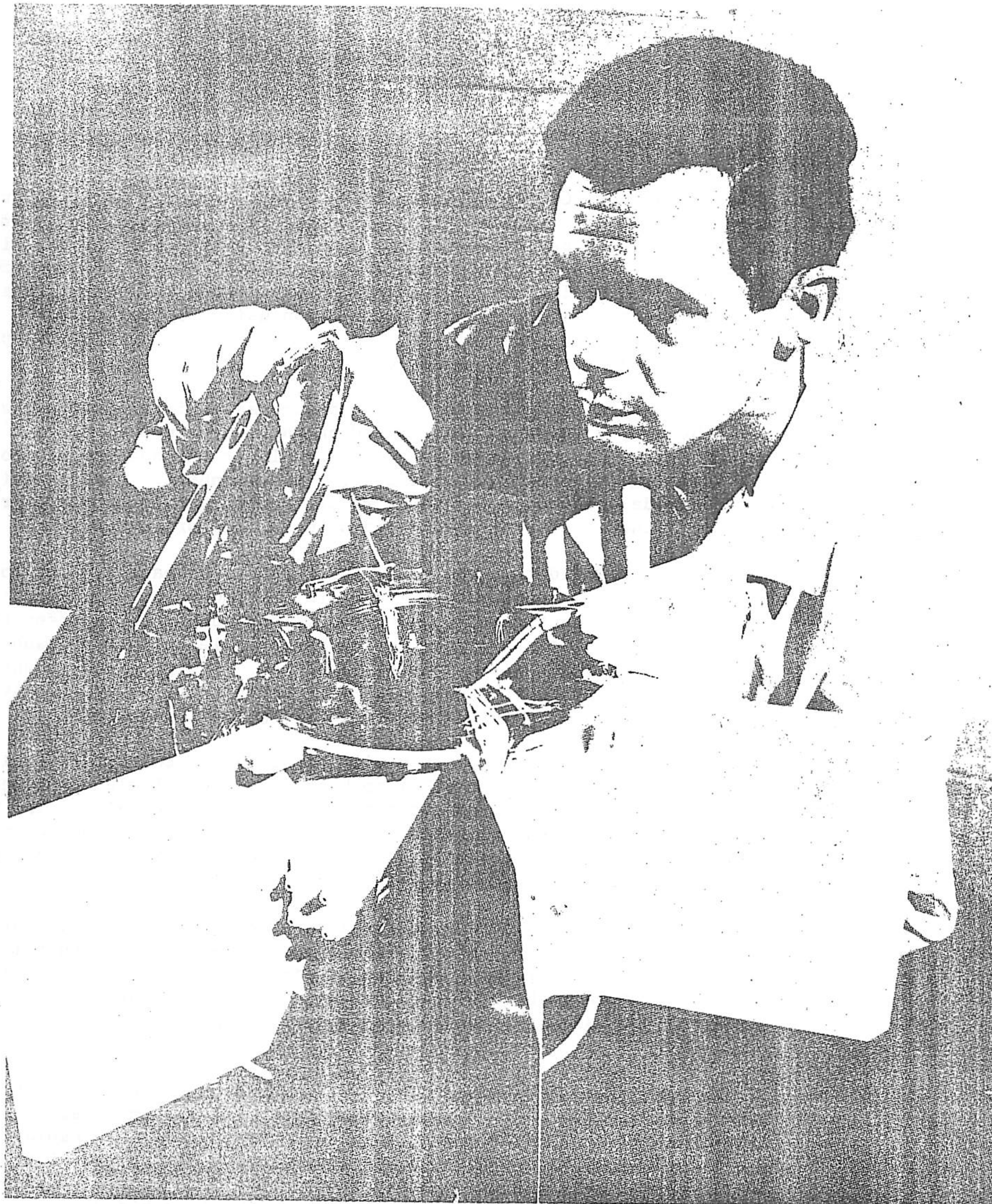








SNAP 11 GENERATOR



250-WATT(e) GENERATOR ENGINEERING PROGRAM STATUS 7/1/66

I. Introduction

The 250-watt(e) Generator Engineering Study currently under contract will, as the first task, investigate the feasibility of constructing an efficient Radioisotope Thermoelectric Generator (RTG) with the capability of reentering the earth's atmosphere in an intact manner. Plutonium-238 Strontium-90 will be considered as the applicable fuel forms; the generator system will be sized for operation using Sr-90 fuel data, yet still should be capable of efficient operation by insertion of appropriate number of Pu-238 fueled capsules.

1. The system being studied will include, as an integrated package, the RTG, a guidance and command package and a mechanism to initiate retrograde velocity to the RTG or the heat source to control its reentry into the earth's atmosphere. Analyses will be made concerning shielding requirements, ground handling prior to launch, and post-abort safe disposal and/or recovery.

2. It is expected that the generator itself will have a power-to-weight ratio greater than 1 watt/lb. It must be recognized that the intact reentry feature will add additional burdens to the weight because of heat or radiation shields required and the addition of a reentry control mechanism. The power-to-weight ratio of the entire power system is expected to be approximately .5 watt/lb. at power levels in the range of 150-400 watts.

The second task of this program will be to evolve conceptual designs of ground handling equipment (GHE) to handle and load fueled capsules into the generator when located about and on the launch complex. The GHE will have sufficient flexibility to the extent that several types of vehicles can be serviced independently so that only one design of the equipment need be produced and located at a particular test range.

This study is currently funded for a Phase Zero effort to run for approximately six months. Lockheed Missile and Space Company, General Electric Missile and Space Division and Martin Company are the contractors. It is planned to select the most attractive features from each contractor's study and to issue a Request for Proposals (RFP's).

Follow-on phases for this program will depend on (1) the attractiveness of the system that evolves during phases Zero, (2) the interest of user agencies in establishing a requirement for such a system, (3) the resolution of some of the outstanding safety problems associated with the use of strontium material in space and (4) the demonstration of the ground handling of large amounts of isotopic material on and around the launch pad.

250 WATT(E) GENERATOR ENGINEERING STUDY

CONTROLLED INTACT REENTRY

ISOTOPE POWER SYSTEM

Design Life	3-5 Years
Radioisotope Fuel	Sr-90, Pu-238

Use

Communications, Navigational and Meterological Satellites; Military Applications; Extraterrestrial Probes

Status

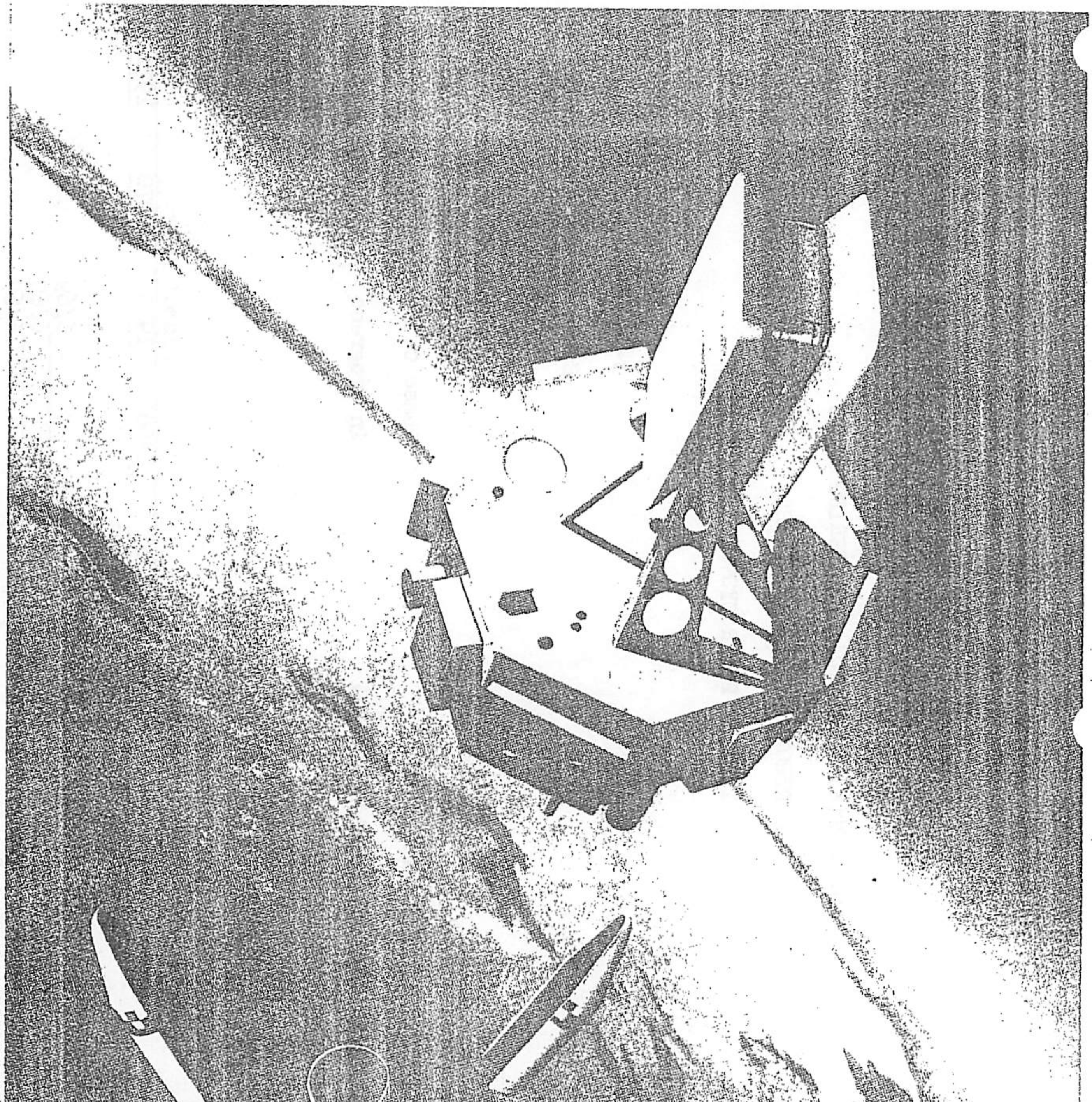
Phase 0, Feasibility of Concept, in Progress

Schedule

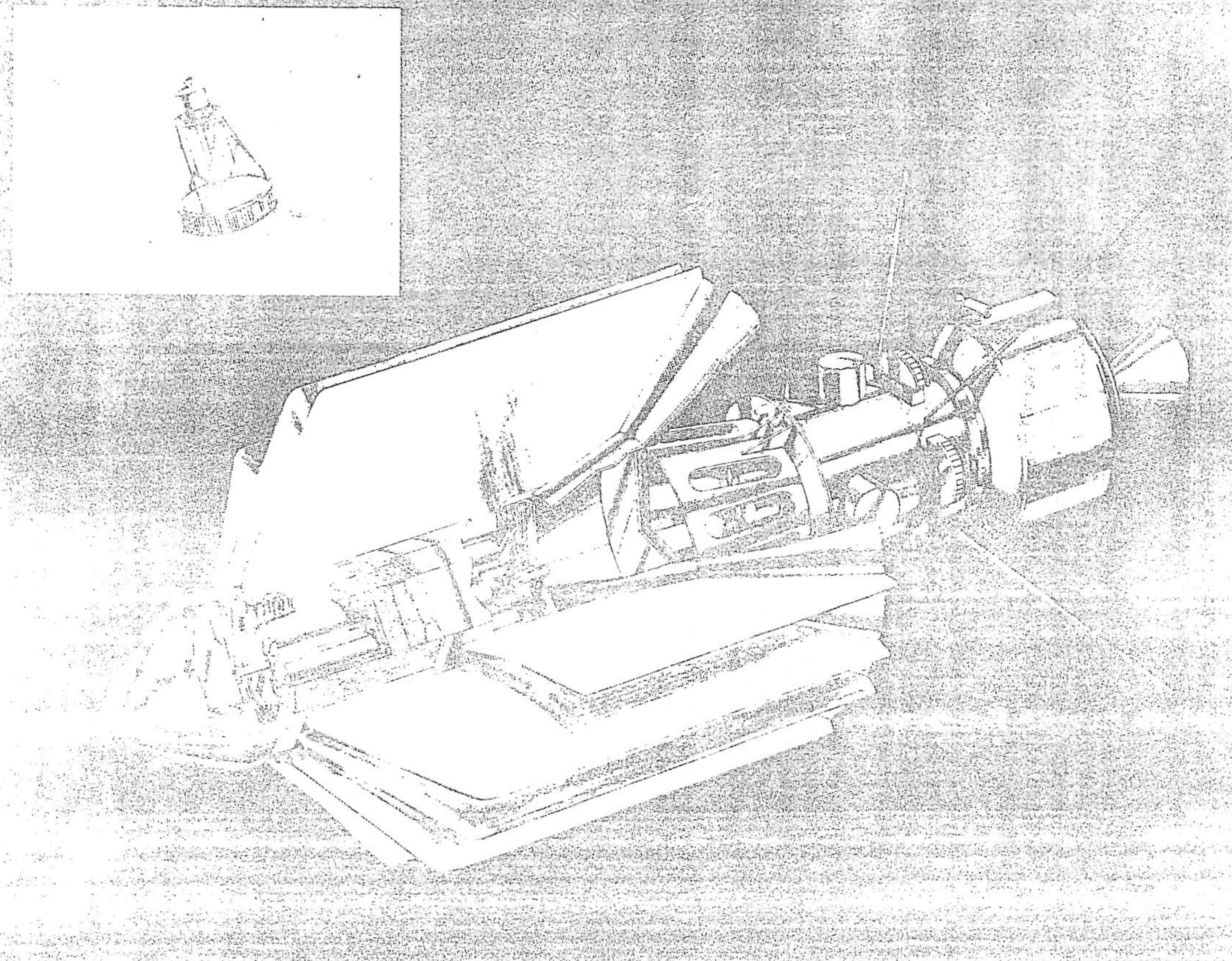
Fiscal Year 1966 – 1972

Feasibility of Concept	1966	1966
RTG Development	3/67	6/69
Controlled Intact Reentry System Development	4/67	6/69
Aerospace Ground Equipment System Development	2/67	6/69
Site Demonstration of RTG, CIR and AGE		6/69
Reliability Test on RTG, CIR and AGE	6/69	6/72
Operational Systems Delivery	10/71	6/72

ADVANCED SR-90 SPACE POWER SUPPLY--STATIC SYSTEM



RTG RE-ENTRY CONFIGURATION



LIGHT WEIGHT THERMOELECTRIC GENERATOR (ISOTEC)
PROGRAM STATUS 7/1/66

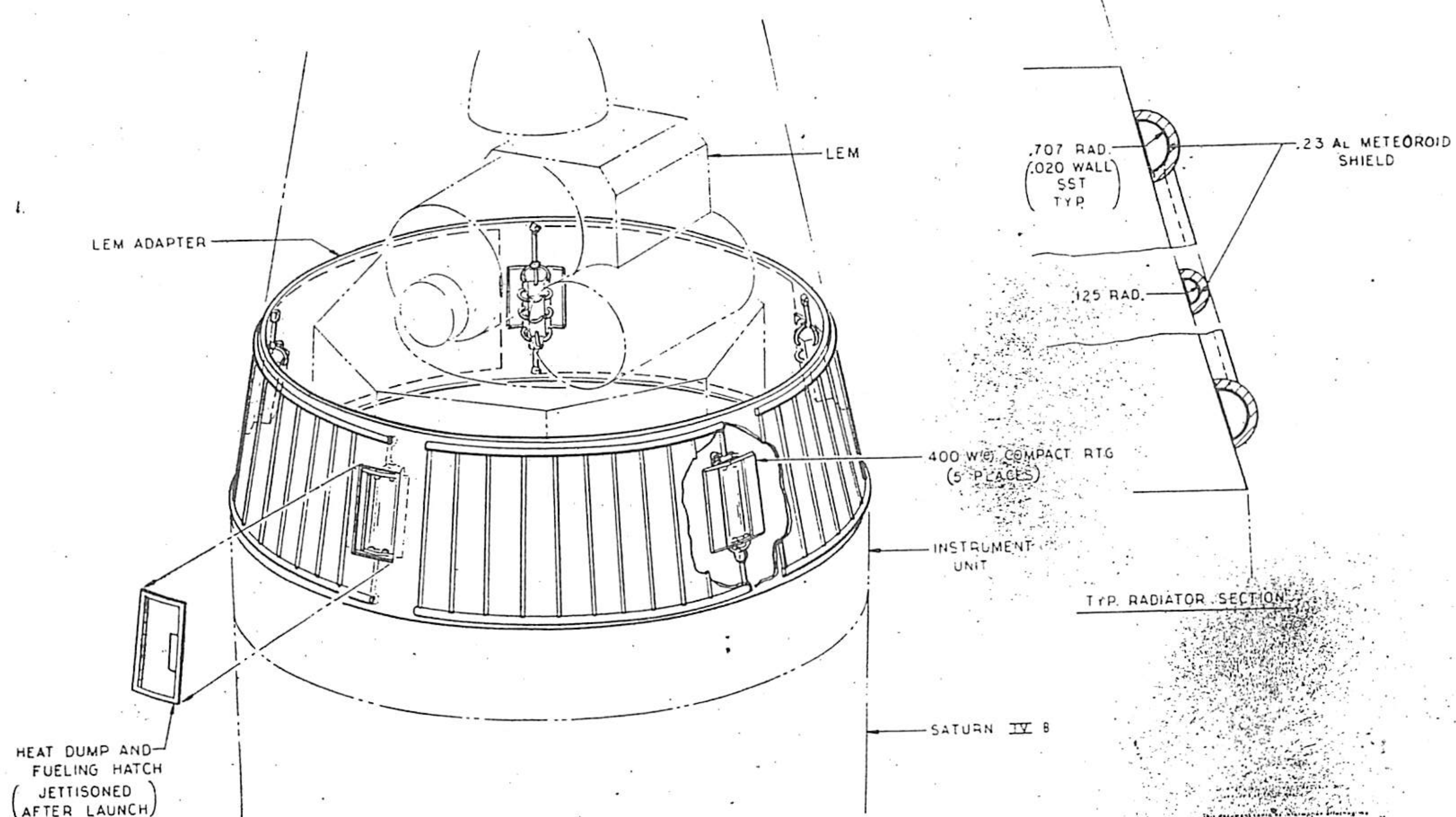
In experiments performed on 2-couple submodules representative of those that will be incorporated in ISOTEC panels experimental operation has been extended beyond 9000 hours. Total degradation in performance has been less than 1% and upon thermal cycling step-wise degradation of performance has not been experienced.

Eighteen couple panels have been operated beyond 500 hours without discernible degradation. Fabrication of sufficient panels to operate an experimental electrically heated generator is complete. Tests on the generator will be completed by December 1966.

A program to extend the power level to hundreds of watts is planned to be initiated very shortly. Among the efforts will be experiments to incorporate heat pipes with the ISOTEC panel design. Cascading of silicon-germanium with the lead telluride ISOTEC panels for ultimate usage with high temperature heat sources will be investigated.

In the 10-30 watt power range it is believed that the ISOTEC concept is ready for immediate development.

CONFIDENTIAL



TWO KILOWATT (e) T/E POWER SYSTEM (CONCEPT 3A)

CONFIDENTIAL

CRD

ISOTOPE POWER SYSTEM

DESIGN STUDIES
LUNAR SURFACE

KILOWATT (e) (Mission Duration - 1 year)

RADIOISOTOPE FUEL	Pu-238
CONVERSION MATERIAL	Pb-Te
WEIGHT (lb) UNSHIELDED	1000
WEIGHT (lb) SHIELDED - 3 REM	1800
POWER OUTPUT (kw)	1.5
RADIATOR AREA (ft^2)	150
OPERATING TEMPERATURE ^oF	
HOT JUNCTION	1000
COLD JUNCTION	400
HEAT INPUT (kw)	26.6
OVERALL EFFICIENCY (%)	6.9

USE

LUNAR SURFACE APPLICATION

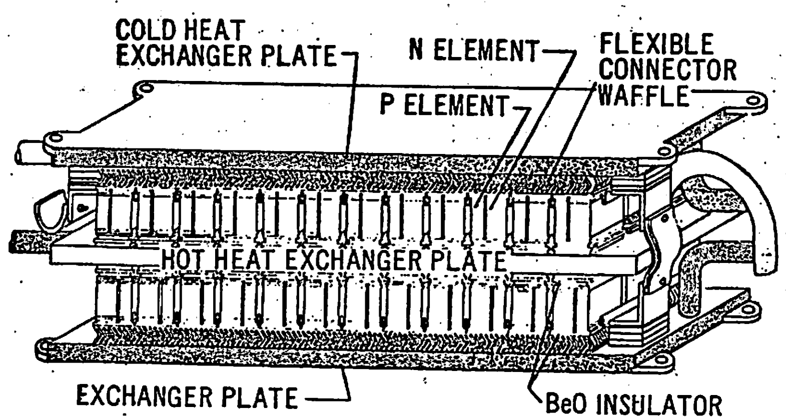
STATUS

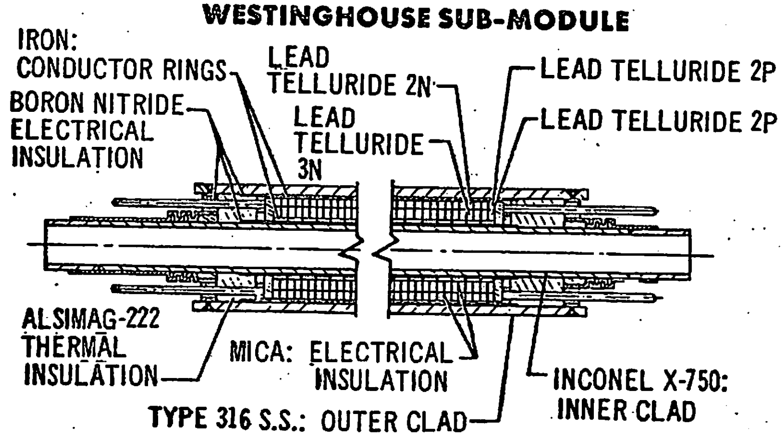
PRELIMINARY PARAMETRIC STUDY COMPLETED

CRL

CONFIDENTIAL DEFENSE INFORMATION (G-3)

CONVERTER SUB-MODULE DESIGN AND PERFORMANCE





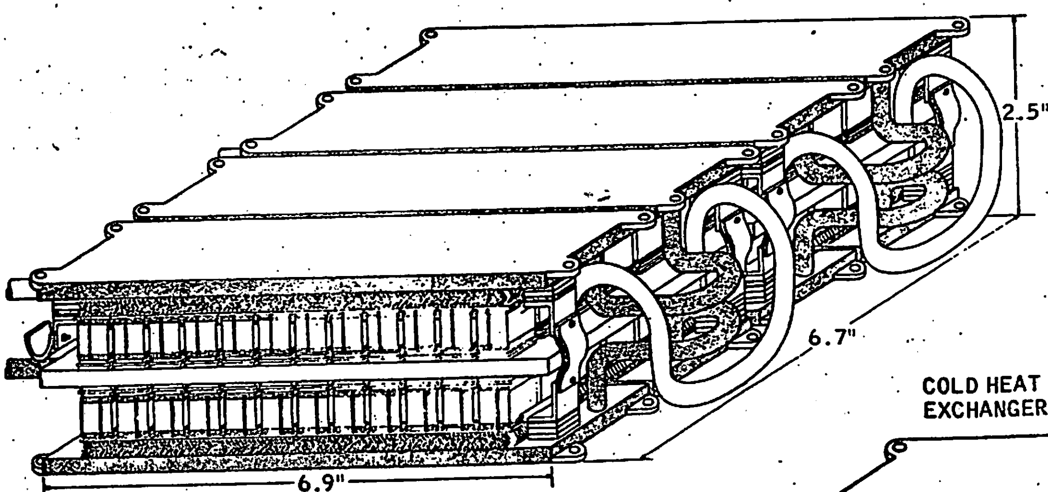
RCA SUBMODULE	PARAMETERS	WESTINGHOUSE SUBMODULE
SiGe	THERMOELECTRIC MATERIAL	Pb-Te
1200	HOT NaK TEMPERATURE (°F)	1200
500	COLD NaK TEMPERATURE (°F)	425
1560	HEAT INPUT (watts)	2641
61.2	ELECTRICAL OUTPUT (watts)	173
2.42	MATCHED LOAD VOLTAGE (volts)	7
3.92	EFFICIENCY (%)	6.50
506 lb/hr at 0.35 psi	HOT NaK FLOW RATE AND PRESSURE DROP	216 lb/hr at 0.195 psi
486 lb/hr at 0.29 psi	COLD NaK FLOW RATE AND PRESSURE DROP	198 lb/hr at 0.185 psi

CONFIDENTIAL DEFENSE INFORMATION (G-3)

5-4-66

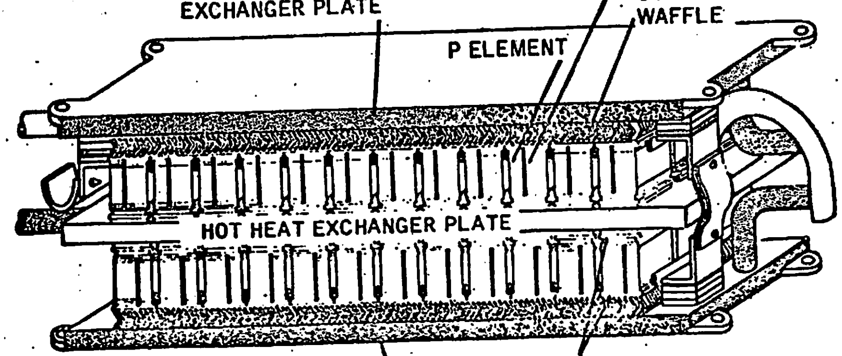
7569-02438

RCA SiGe CONVERTER MODULE



MODULE DESIGN POINT

HOT NaK TEMPERATURE (°F)	1200
COLD NaK TEMPERATURE (°F)	500
HEAT INPUT (watts)	6241
ELECTRICAL OUTPUT (watts)	245
MATCHED LOAD VOLTAGE (volts)	9.66
EFFICIENCY (%)	3.92
HOT NaK FLOWRATE AND PRESSURE DROP	506 lb/hr AT 1.4 psi
COLD NaK FLOWRATE AND PRESSURE DROP	486 lb/hr AT 1.16 psi
MODULE SPECIFIC WEIGHT (lb/kwe)	49



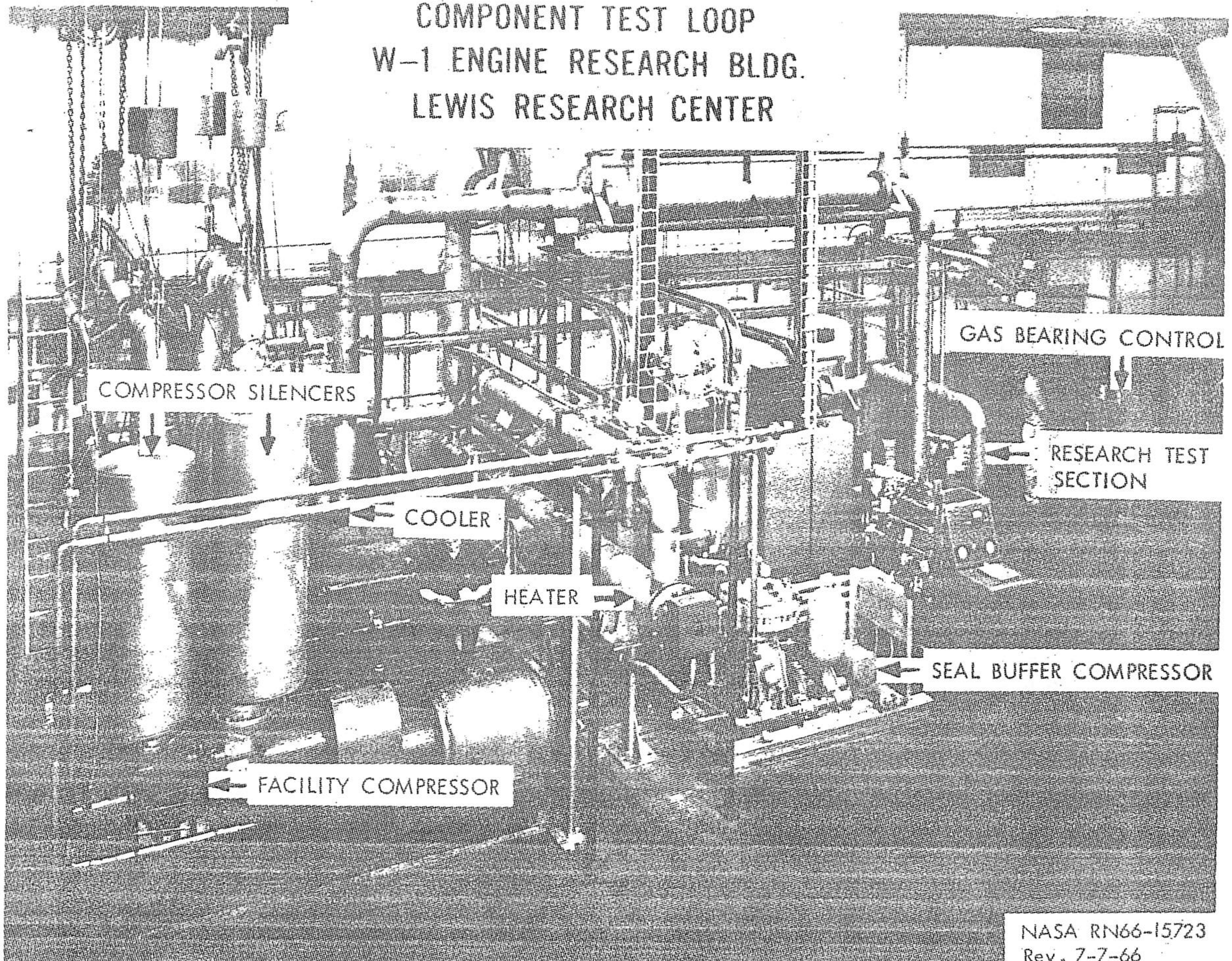
7569-0275

ISOTOPE-BRAYTON DEVELOPMENT

The schematic shows the two-shaft Brayton cycle power system on which the Lewis Research Center is conducting tests. The alternator package runs at 12,000 rpm to accommodate the 400 cycle alternator. The turbo-compressor package runs at 38,500 rpm design speed. Both packages will operate on gas bearings. Nominal power is 8KW. A single loop version at 5.5KWe is now under design.

This chart shows the test rig that the radial flow turbo-compressor was checked out on. The turbine was run up to 950°F on both air and argon. The compressor was loaded separately on ambient air. The gas bearings were operated on air and argon, both hydrostatically and hydrodynamically. Turbine operated at 950°F at design speed of 38,500 rpm. The rig was limited to 1000°F by heater capacity.

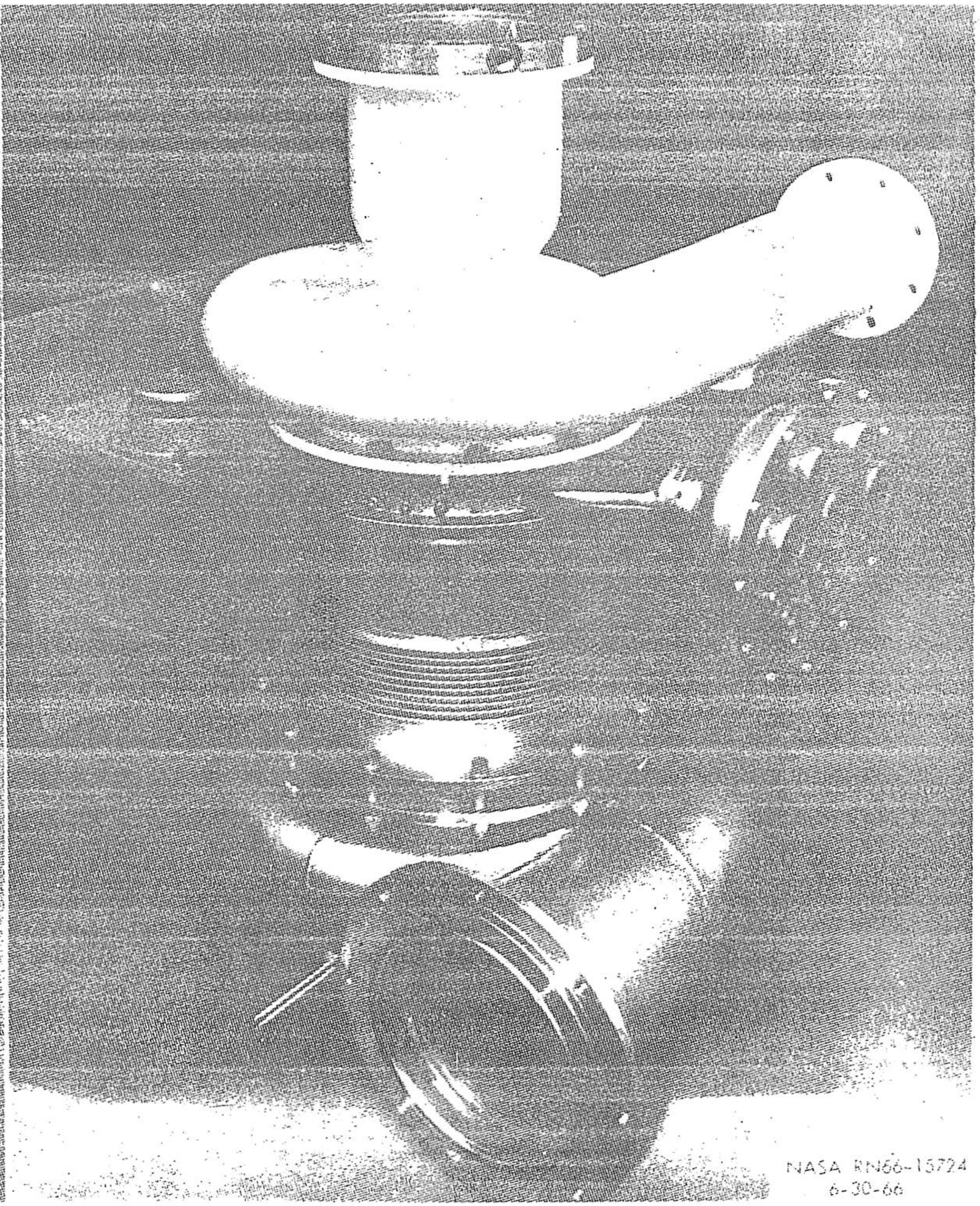
BRAYTON CYCLE EVALUATION
COMPONENT TEST LOOP
W-1 ENGINE RESEARCH BLDG.
LEWIS RESEARCH CENTER



NASA RN66-15723
Rev. 7-7-66

Figure 3

This is a picture of the radial flow turbo-compressor package (with gas bearings) that was tested in the W-5 facility. In this facility the unit will be tested as a self-sustaining engine at design speeds from about 900°F to 1550°F.

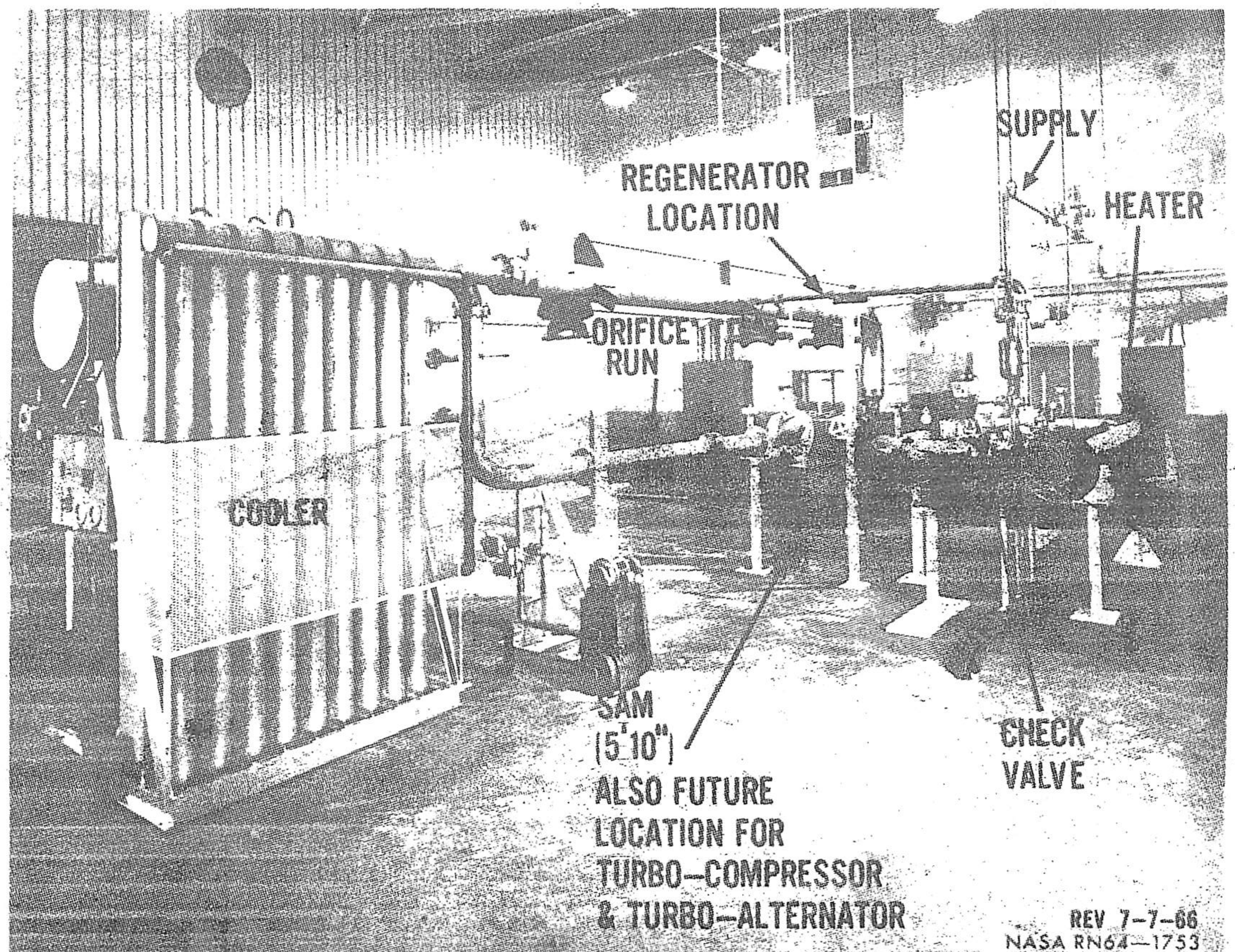


TURBO-COMPRESSOR
GAS BEARING UNIT
8 KW

NASA RN66-15724
6-30-66

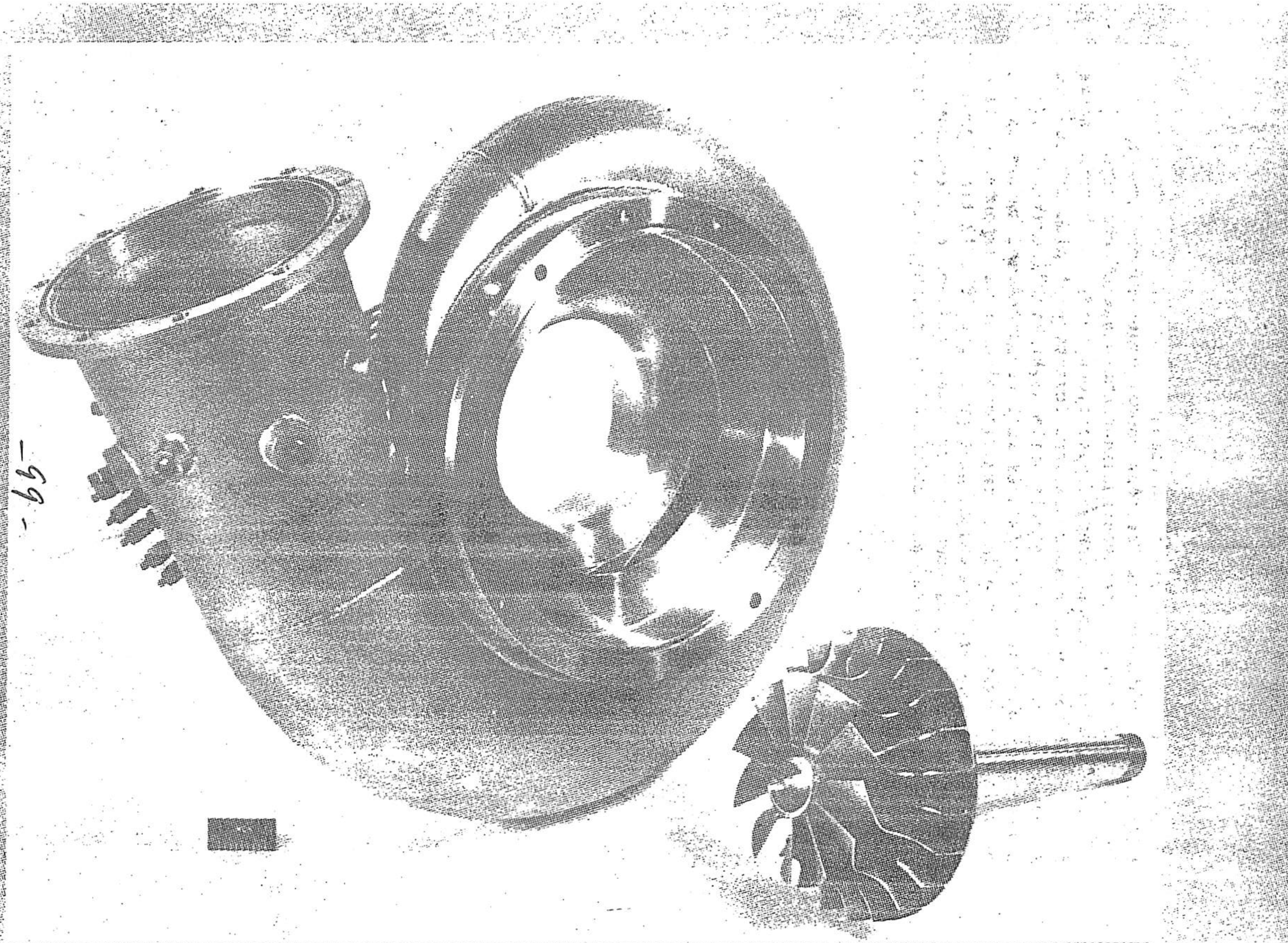
This test rig can test all Brayton system components to 1550°F.

TURBO–COMPRESSOR HOT ARGON TEST LOOP
W–5 ENGINE RESEARCH BUILDING LEWIS RESEARCH CENTER



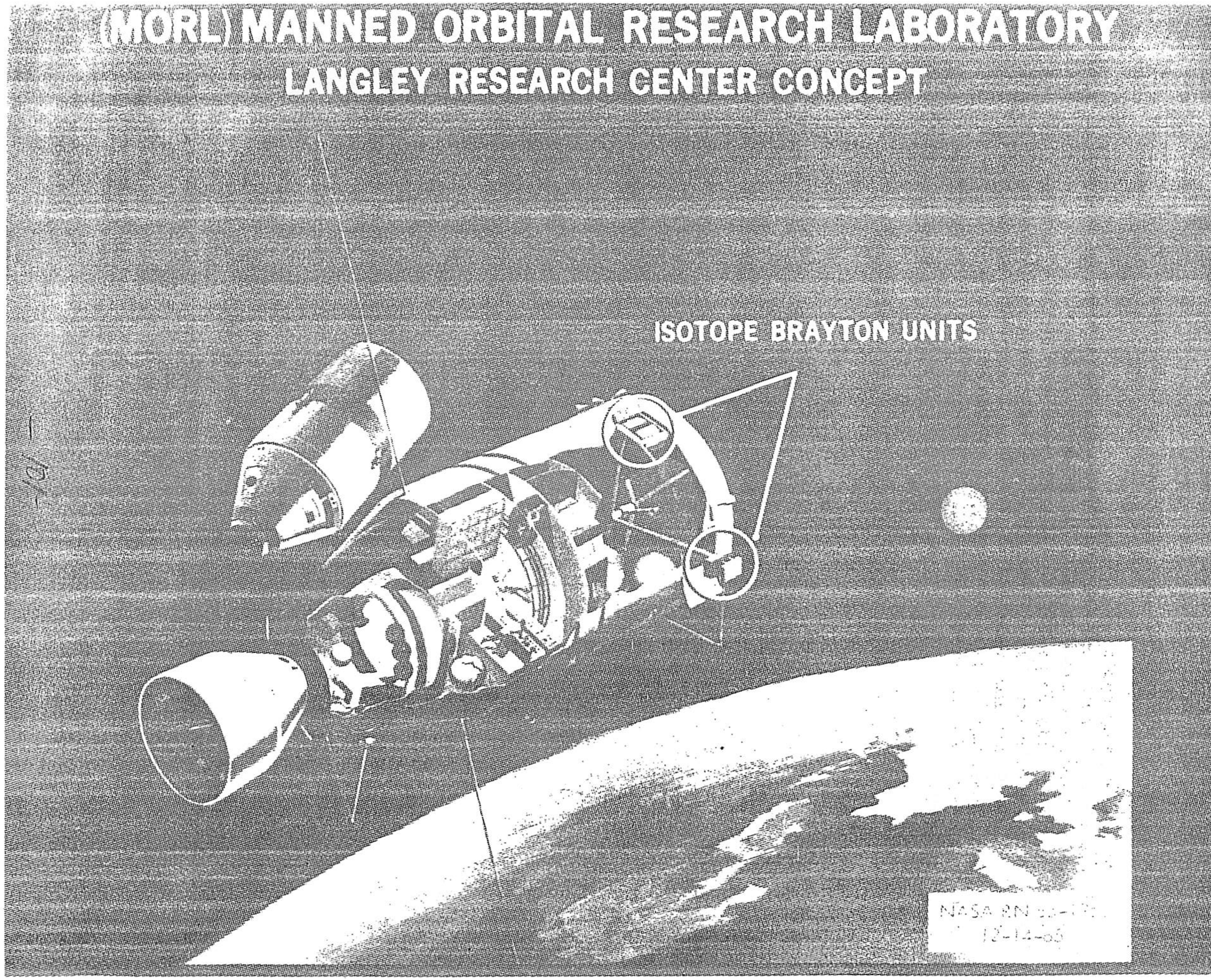
This is a picture of the research turbine that was tested for confirming the aerodynamic design performance. This turbine is a duplicate of the turbine in the gas bearing package shown on the previous chart. The package operates on oil-rolling element bearings for ease of testing.

BRAYTON CYCLE RESEARCH TURBINE 6 INCH DIAMETER



NASA RN 66-285
2-1-66

In the class of advanced systems for the 1970's is this isotope-Brayton unit which is the reference power plant of the Langley-Douglas MORL design. The design takes advantage of the high efficiency of the closed Brayton cycle, and fairly high temperatures in the fuel blocks to reduce the inventory of plutonium 238 fuel to a reasonable level. (It is still high: Even at 27% over-all efficiency, one 11-kwe system will require 41 thermal kilowatts of plutonium, which is a large fraction of what is expected to be produced for the whole national program by the 1971-72 time frame of this station; and of course, extra fuel would be required for spare units, experimental and qualification testing, and fabrication wastage.


(MORL) MANNED ORBITAL RESEARCH LABORATORY
LANGLEY RESEARCH CENTER CONCEPT
ISOTOPE BRAYTON UNITS

But its performance, as summarized here, is impressive.

It should be pointed out that this system uses technology which is not available. We hope the high temperature heat source and the space ready Brayton cycle can be ready in the mid 1970's, but we view this capability as being somewhat more speculative than the improved thermoelectrics and the Mercury vapor Rankine technology which will be described later. These intermediate systems would yield specific weights somewhere between the present RTG at 1-2 watts per pound and the 1640 F Brayton cycle at about 6 watts per pound.

MORL ISOTOPE BRAYTON SYSTEM

POWER - kwe	2 MODULES @ 5.5 kwe
RADIATOR AREA - ft^2	400 PER MODULE (73 ft^2/kwe)
SYSTEM WEIGHT - lbs	900 PER MODULE (163 #/kwe)
ISOTOPE LOADING - kwt	20.4 per MODULE
SYSTEM EFFICIENCY - %	27.0
TURBINE INLET TEMP. - ^{o}F	1640
MAINTENANCE	PCS PACKAGES ARE REPLACEABLE

DISCUSSION OF ISOTOPE AVAILABILITY - NASA MEETING
JULY 12, 1966

In November 1964, the AEC supplied NASA with indications of the cost versus quantity relationships through 1980 for a number of isotopes with the expectation that this would lead to NASA studies and eventual indications of user interest. Subsequently, in March 1965, we received guidance from Mr. Webb which formed the basis for intensive production studies on our part. Information on the steps taken to implement these studies is an important part of my discussion today.

The isotopes to be discussed are as follows:

Fission Products

Pm-147
Sr-90

Reactor Products

Pu-238
Cm-244
Po-210
Co-60
Tm-170

As you are aware, there are hundreds of other isotopes, may of which have very interesting properties. However, there seems to be a perversity in nature which makes it either difficult, costly or virtually impossible to produce the most interesting ones in quantity. The net result is that the isotopes considered for large scale use in space represent compromises between the ideal and the possible.

Fission Products

Promethium-147 and strontium-90 will be produced in significant quantities in a new commercial plant at Hanford which is expected to be in operation in 1969. Much larger quantities are potentially available from power reactors. About 4 years would be required to provide facilities to recover them, and naturally, whether such facilities will be built depends on customer demand.

Note that the potential amounts from power reactors increase rapidly after the early seventies. The important point is that the materials will be there, and it can be assumed that facilities would be built to recover them if need develops and the cost of products from such operations is acceptable to the users.

Reactor Produced Isotopes

The availability of these materials depends on the total power, the neutron flux levels and the space available for irradiation of target material. For immediate applications, the only large scale sources of all reactor products will be the AEC production reactors. For certain reactor products, such as those requiring high flux, these reactors will probably continue to be the only source for the long-term also. The production reactors are large size, they are specifically designed for production purposes, being quite flexible as to space for target material and ability to modify neutron flux. These capabilities are very important, because different products require quite different irradiation conditions.

Power reactors are potentially of great value, but currently their total power output is relatively small, space within them is limited and neutron flux is relatively low. Despite these current disadvantages, it should be noted that the growth of power reactor capacity will be so rapid that they cannot be overlooked as a source of supply for specific isotopes in the long term future.

Increased interest in the heat producing isotopes has provided the AEC with a strong stimulus to develop methods of production that are effective, economical and flexible. At the two reactor sites, this has led to a substantial change in the process development programs. Counting work on both the near-term and long-range programs, more than 50% of the $12 million/year R&D budget is being spent on some aspect of producing isotopes in conjunction with weapon materials.

Production of Pu-238 is a frustrating job because of the immediate precursor element, neptunium-237, is very difficult to make and the overall Pu-238 production time cycle is seven years or more. The AEC has recently approved a program for the production of Pu-238 that will produce about twice as much as would be available from the previous by-product method alone. However, even these increased quantities may still be inadequate to meet the needs of NASA, DOD and other government agencies unless it proves possible to obtain Np-237 from power reactors. At the present time, the AEC is working with the commercial chemical processing plant operator to determine the feasibility of neptunium recovery from spent power reactor fuel.

In a search for possible alternative isotopes which could supplement the Pu-238 supply, special interest has been focused on curium-244. Experimental amounts are currently being produced at the Savannah River plant, and the AEC has recently approved an increase in the production goal of 3 kg to 4.5 kg. for the purpose of accelerating the evaluation program. Approximately 10 kg additional could be made over the next several years by irradiation of target materials remaining after the present program; much larger amounts could be made if large-scale production were authorized.

Polonium-210 has the advantages of very high specific power, low shielding requirements, no shortage of feed stock and ability to be made in large amounts. Disadvantages are short half-life, substantial investment necessary to provide chemical processing facilities and some reactor scheduling problems for amounts of intermediate size. Polonium-210 has been made on a small scale for many years, and currently, vigorous development work is in progress on the technology of large-scale separation and encapsulation.

High specific activity Co-60 has some very strong advantages including no requirement for capital investment, low cost per thermal watt and ability to be made in very large amounts. An especially important advantage is that no chemical processing is required after irradiation. Because of the attractiveness of high activity cobalt-60 as a heat producer in several applications, R&D work has been approved on a considerable scale.

There is no apparent problem in making thulium-170 at substantial rates since there are no foreseeable difficulties of obtaining target material or of providing necessary irradiation. Thulium-170 has the disadvantage of a short half-life, but relatively little investment would be required for its production since no chemical processing would be necessary after irradiation. Thulium-171, with a better half-life, would be a much more desirable isotope, but is virtually impossible to produce. We have so far been unable to find any practical method.

The AEC has a strong belief that the role of isotopes will increase with time and has, therefore, embarked on a planned 5-year process development program that will establish the technology for producing a wide variety of reactor-produced isotopes. In working on this problem, we take into account that AEC has to meet diverse demands for many products, and to assure the most effective utilization of available facilities, has to develop alternate long-range operating plans. The letter sent to AEC by Mr. Webb in March 1965, regarding requirements was a valuable assist in such planning and provided a basis for an extensive study of many alternate means of meeting NASA goals, while at the same time providing for other commitments. As might be expected, we would like to see more of this mutual effort in planning; the receipt of possible isotope usage would enable us to explore the many alternatives that are available to determine our ability to meet all requirements. Such long-term planning would facilitate early response to NASA requests once the requirements become firm. Also, any potential conflicts in our future planning could be recognized and corrective actions sought.

HEAT PRODUCING ISOTOPES

Isotope	Half Life	Heat Output Watts/g	Shielding Needed	Lead Time	Production Difficulty
Pm-147*	2.5 yrs	0.33	minor	short 1/	moderate
Sr-90*	28 yrs.	0.95	heavy	short 1/	moderate
Pu-238	89 yrs.	0.56	minor	7 years	substantial
Po-210	138 days	141	minor	short 1/	moderate
Cm-244	18 yrs	2.8	moderate	4 - 6 yrs	substantial
Tm-170	128 days	~ 2	minor	short	small
Co-60	5.3 yrs	~ 3	heavy	short	small

* Fission product.

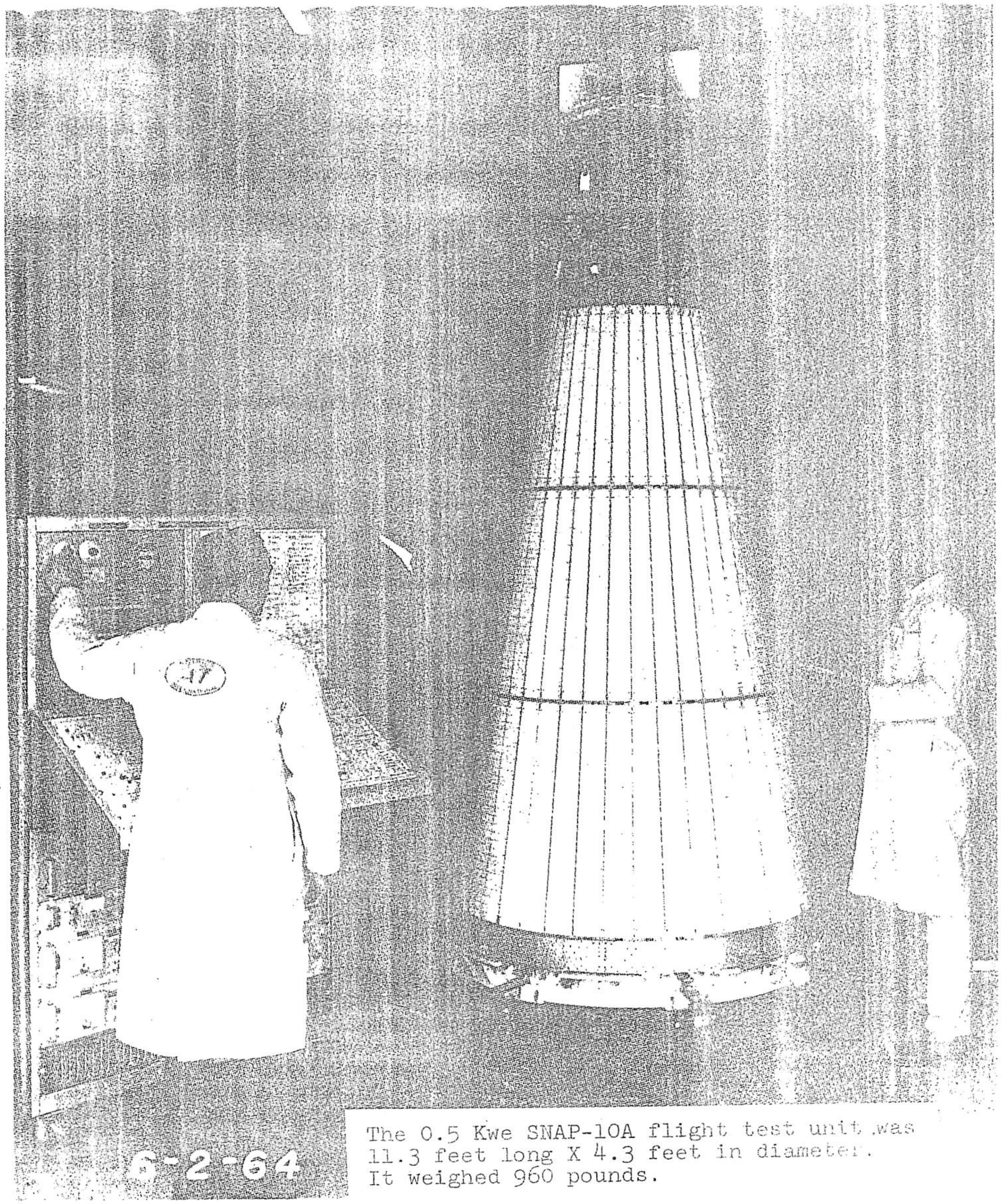
1/ After plant construction

Pm-147

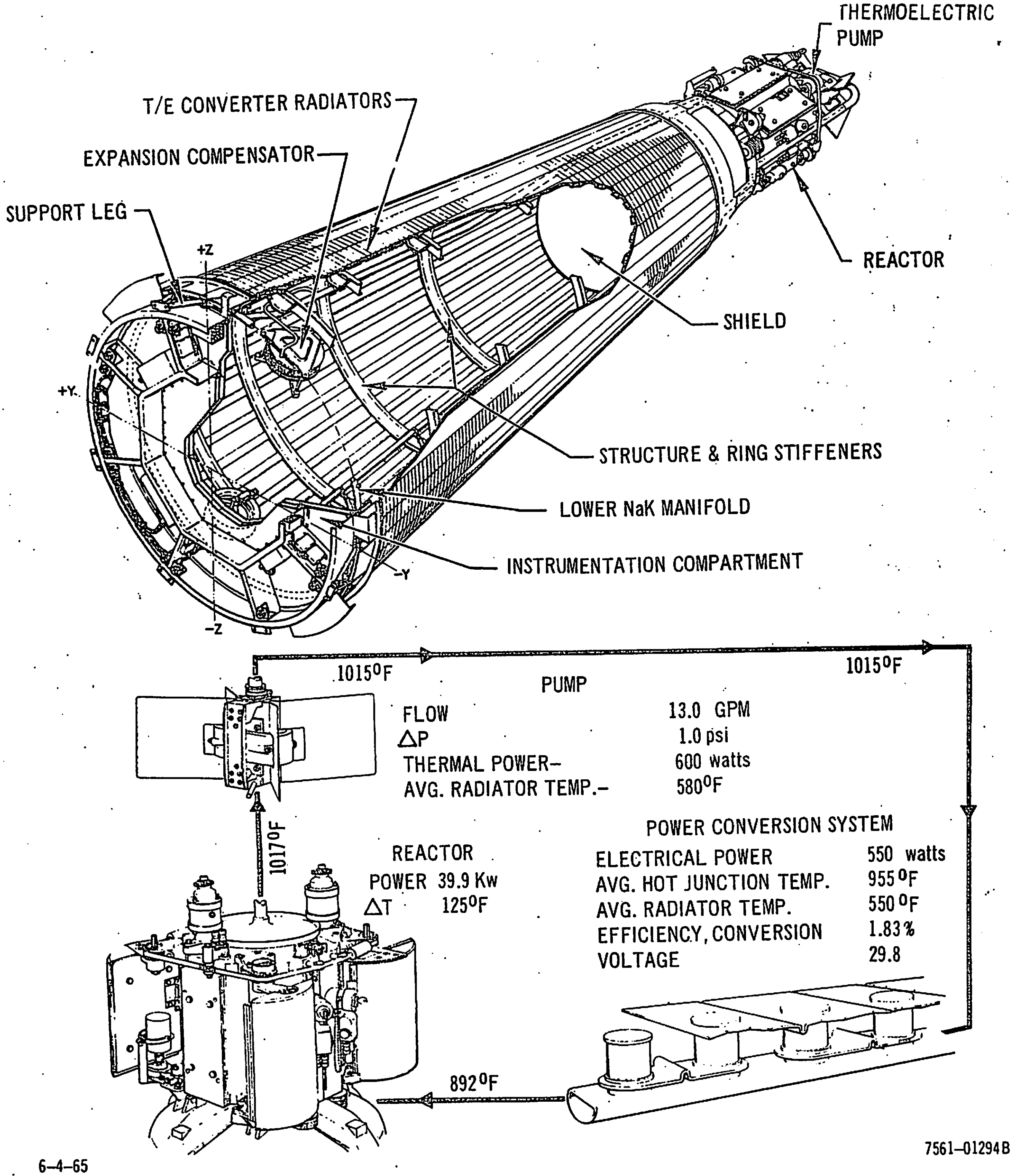
POTENTIAL ANNUAL PRODUCTION - THERMAL Kw

Fiscal Year	Hanford	Power Reactors
1968	1.5	-
1969	3.5	-
1970	7	10
1972	10	8
1974	10	20
1976	10	35
1978	*	55
1980	*	85

* Projection uncertain

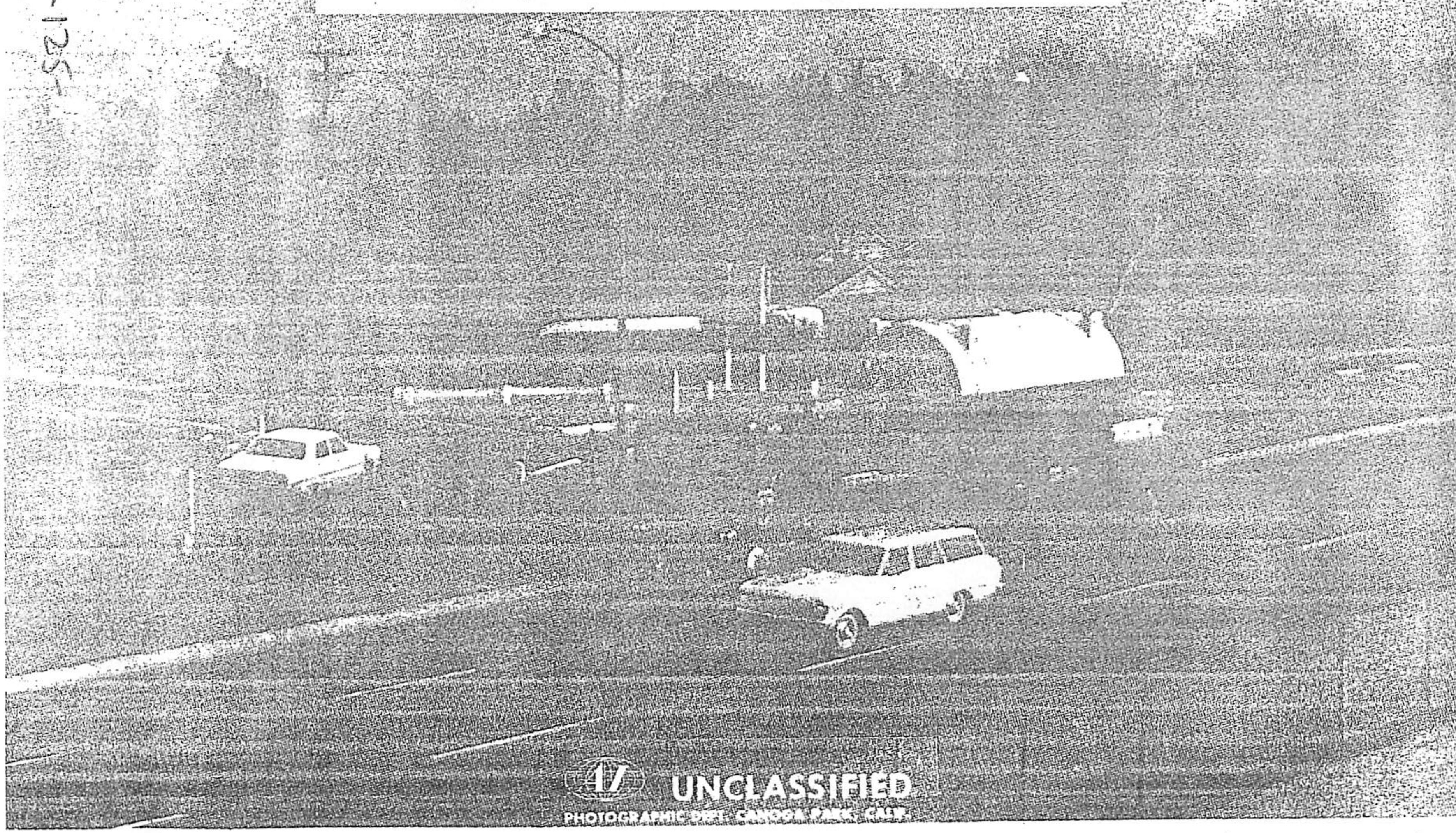


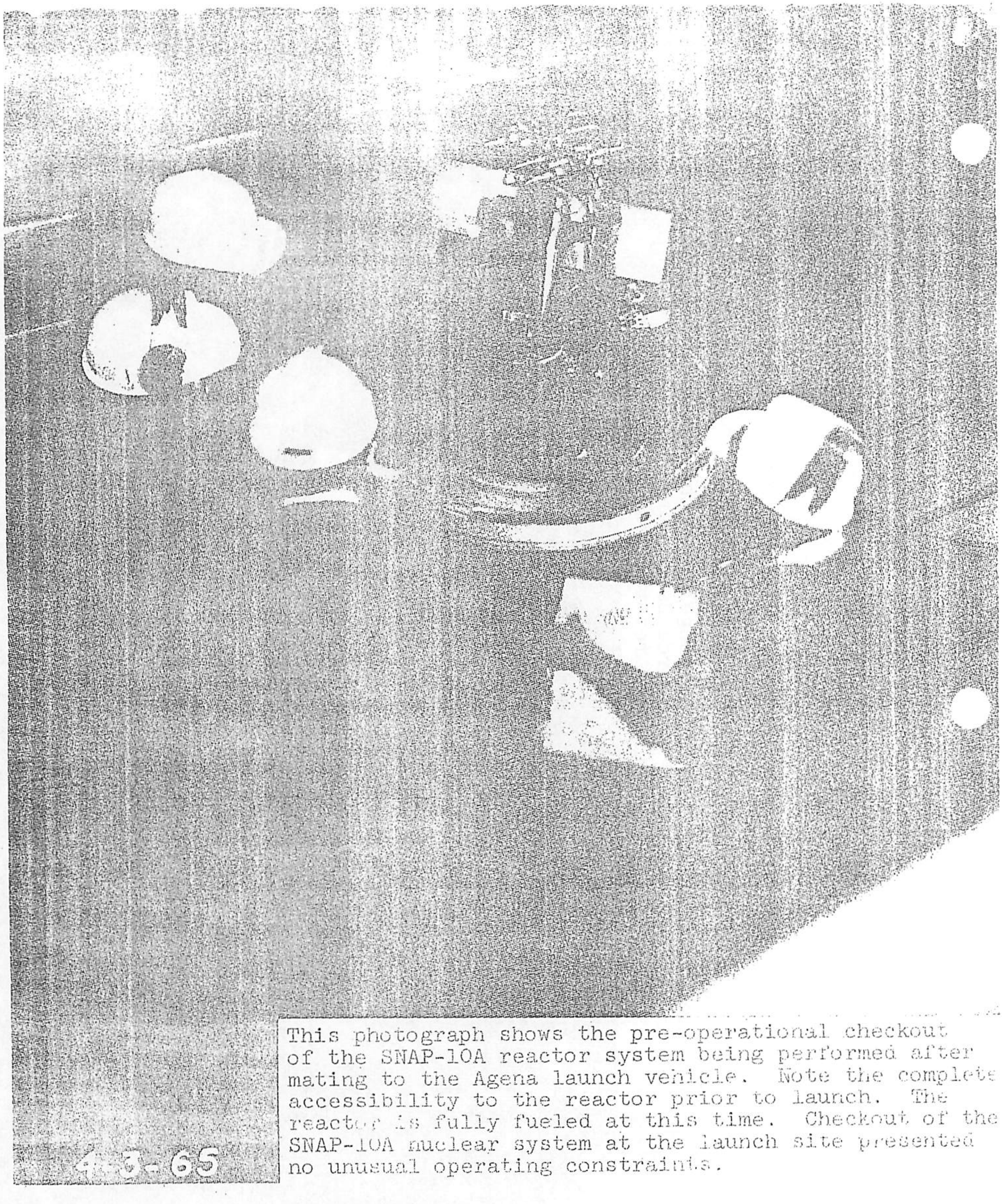
The 0.5 Kwe SNAP-10A flight test unit was 11.3 feet long X 4.3 feet in diameter. It weighed 960 pounds.



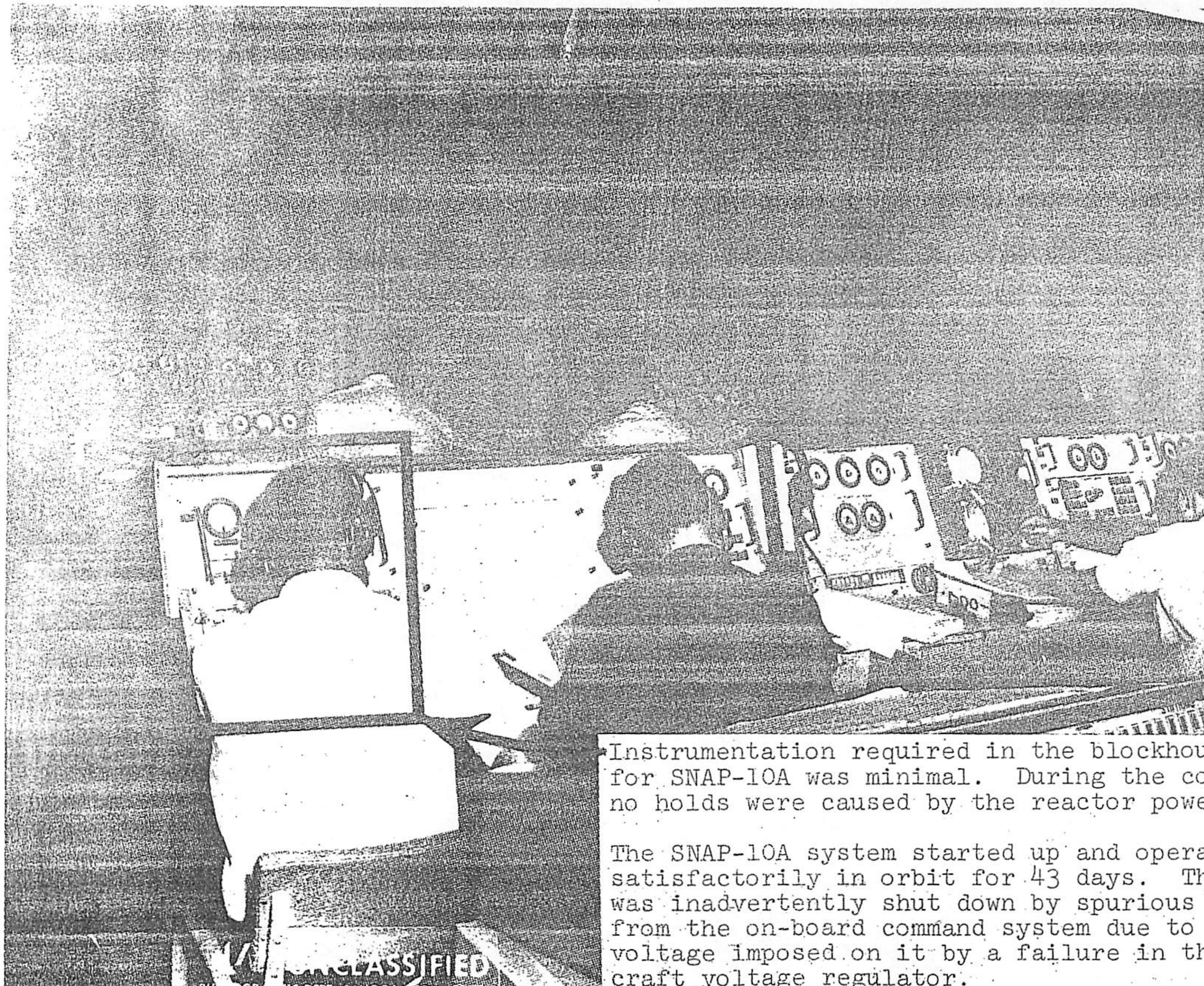
THERMOELECTRIC PUMP
T/E CONVERTER RADIATORS
EXPANSION COMPENSATOR
SUPPORT LEG
+Z
REACTOR
SHIELD
+Y
STRUCTURE & RING STIFFENERS
LOWER NaK MANIFOLD
INSTRUMENTATION COMPARTMENT
-Y
-Z
1015°F
1015°F
PUMP
FLOW 13.0 GPM
ΔP 1.0 psi
THERMAL POWER- 600 watts
AVG. RADIATOR TEMP.- 580°F
1017°F
REACTOR
POWER 39.9 Kw
ΔT 125°F
POWER CONVERSION SYSTEM
ELECTRICAL POWER 550 watts
AVG. HOT JUNCTION TEMP. 955 °F
AVG. RADIATOR TEMP. 550 °F
EFFICIENCY, CONVERSION 1.83 %
VOLTAGE 29.8
892°F
7561-01294B
6-4-65

The SNAP-10A system was fueled and acceptance tested at the factory and then was transported by truck from the factory in Los Angeles to the launch site at Vandenberg Air Force Base along U.S. Highway 101 without unusual restrictions.





This photograph shows the pre-operational checkout of the SNAP-10A reactor system being performed after mating to the Agena launch vehicle. Note the complete accessibility to the reactor prior to launch. The reactor is fully fueled at this time. Checkout of the SNAP-10A nuclear system at the launch site presented no unusual operating constraints.



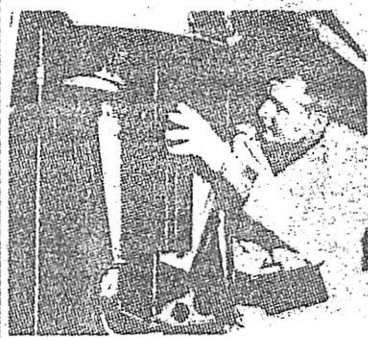
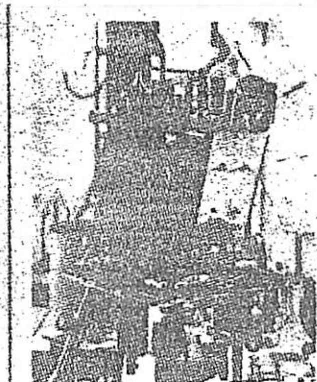
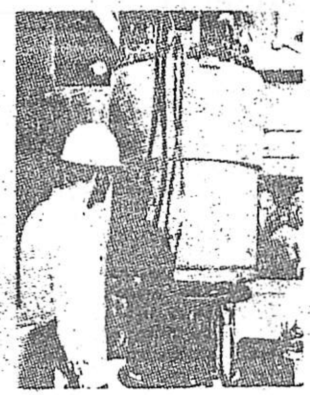
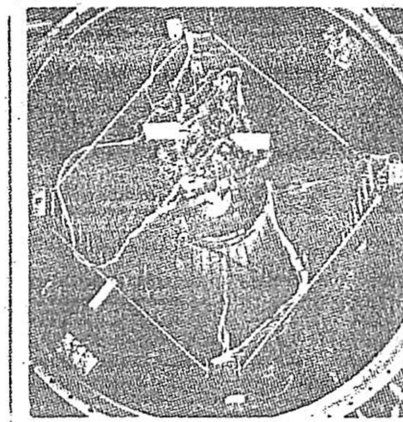
Instrumentation required in the blockhouse for SNAP-10A was minimal. During the countdown, no holds were caused by the reactor power system.

The SNAP-10A system started up and operated satisfactorily in orbit for 43 days. The reactor was inadvertently shut down by spurious commands from the on-board command system due to an over-voltage imposed on it by a failure in the space-craft voltage regulator.

A duplicate of the flight test SNAP-10A system was operated successfully on the ground for 10,000 hours.

The SNAP-8 reactor is similar in concept to the SNAP-10A reactor, but it is scaled up a factor of 15 in thermal power and 300°F in temperature. This photo shows a non-nuclear test model of the SNAP-8 reactor and shield being prepared for environment testing on a shake table.

SNAP REACTOR TEST EXPERIENCE

	SNAP EXPERIMENTAL REACTOR (SER)	SNAP DEVELOPMENTAL REACTOR (SDR)	SNAP 8 EXPERIMENTAL REACTOR (S8ER)	SNAP 10A FLIGHT SYSTEM (FS-3)	SNAP 10A FLIGHT SYSTEM (FS-4)
CRITICAL	SEPTEMBER 1959	APRIL 1961	MAY 1963	JANUARY 1965	APRIL 1965
SHUTDOWN	DECEMBER 1960	DECEMBER 1962	APRIL 1965	MARCH 1966	MAY 1965
THERMAL POWER	50 kwt	65 kwt	600 kwt	38 kwt	43 kwt
THERMAL ENERGY	225,000 kwt-hr	273,000 kwt-hr	5.1 x 10^6 kwt-hr	382,944 kwt-hr	41,000 kwt-hr
ELECTRIC POWER	–	–	–	402 watts	560 watts
ELECTRIC ENERGY	–	–	–	4028 kw-hr	574 kw-hr
TIME AT POWER AND TEMPERATURE	1800 hr AT 1200°F 3500 hr ABOVE 900°F	2800 hr AT 1200°F 7700 hr ABOVE 900°F	1 yr AT 1300°F 400 TO 600 kwt	10,005 hr (417 days)	43 days

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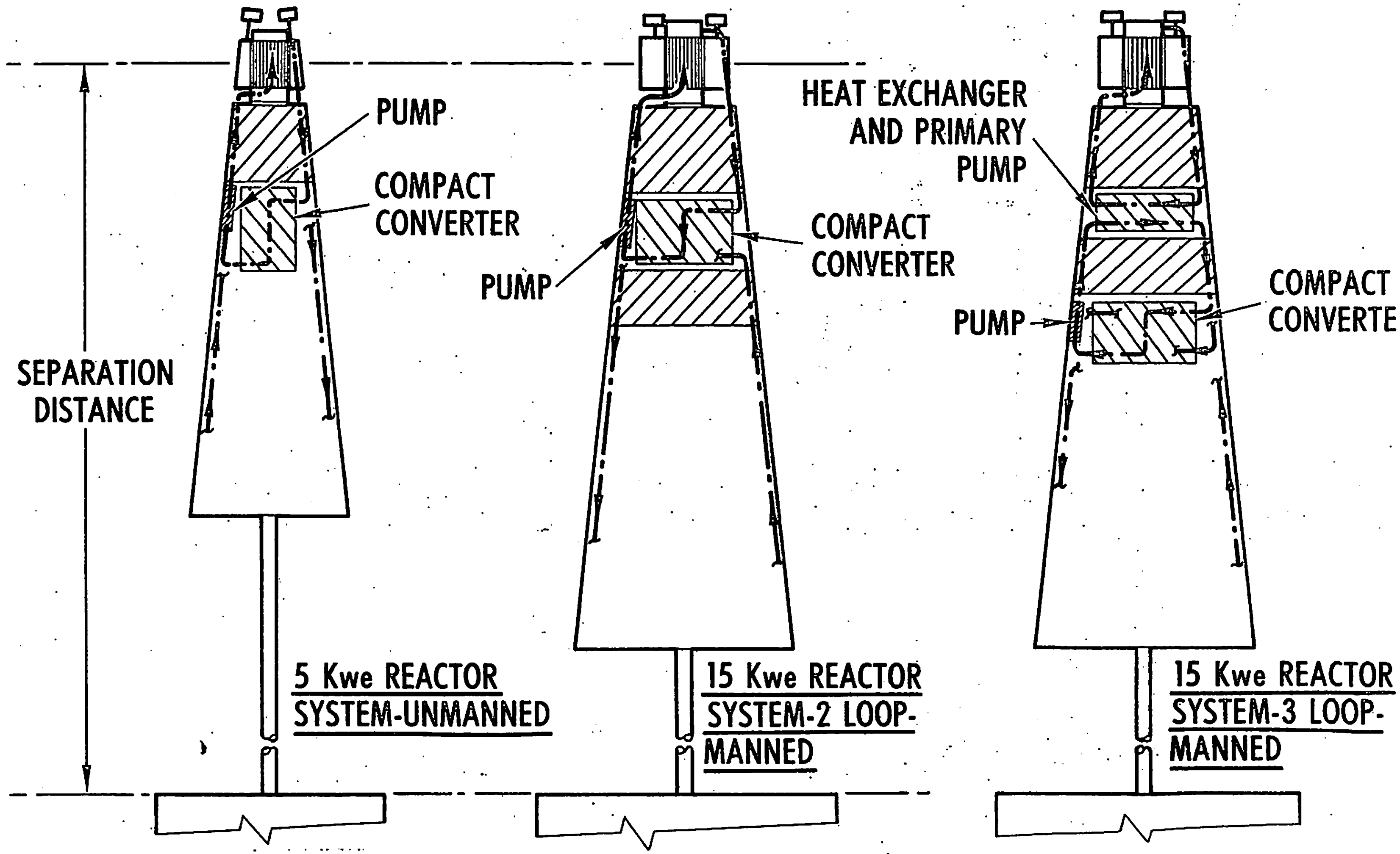
POWER SYSTEM CONFIGURATION APPROACHES



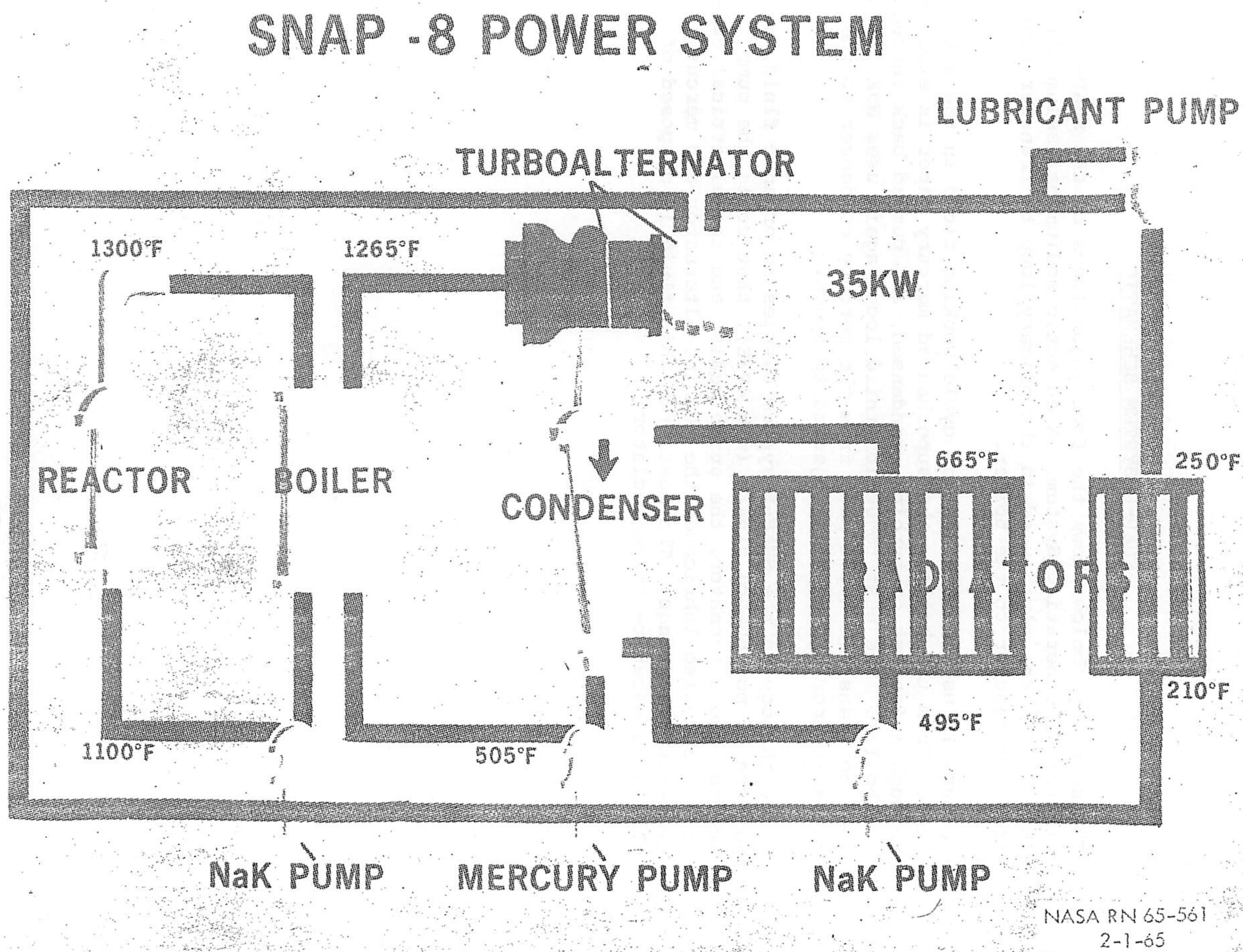
Typical configurations of higher-powered thermoelectric systems are shown in these sketches. Design studies indicate radiator areas in the range of 60 to 80 ft.^{2}/Kwe and weights of 1-2 watts/lb. for shadow-shielded manned systems and 2-3 watts/lb. for unmanned systems.

SNAP-8 SYSTEM SCHEMATIC

The schematic below shows the four major loops of the SNAP-8 Electrical Generating System. A eutectic sodium-potassium alloy (NaK 78) is the fluid used in the primary loop to transfer heat from the reactor to the boiler.

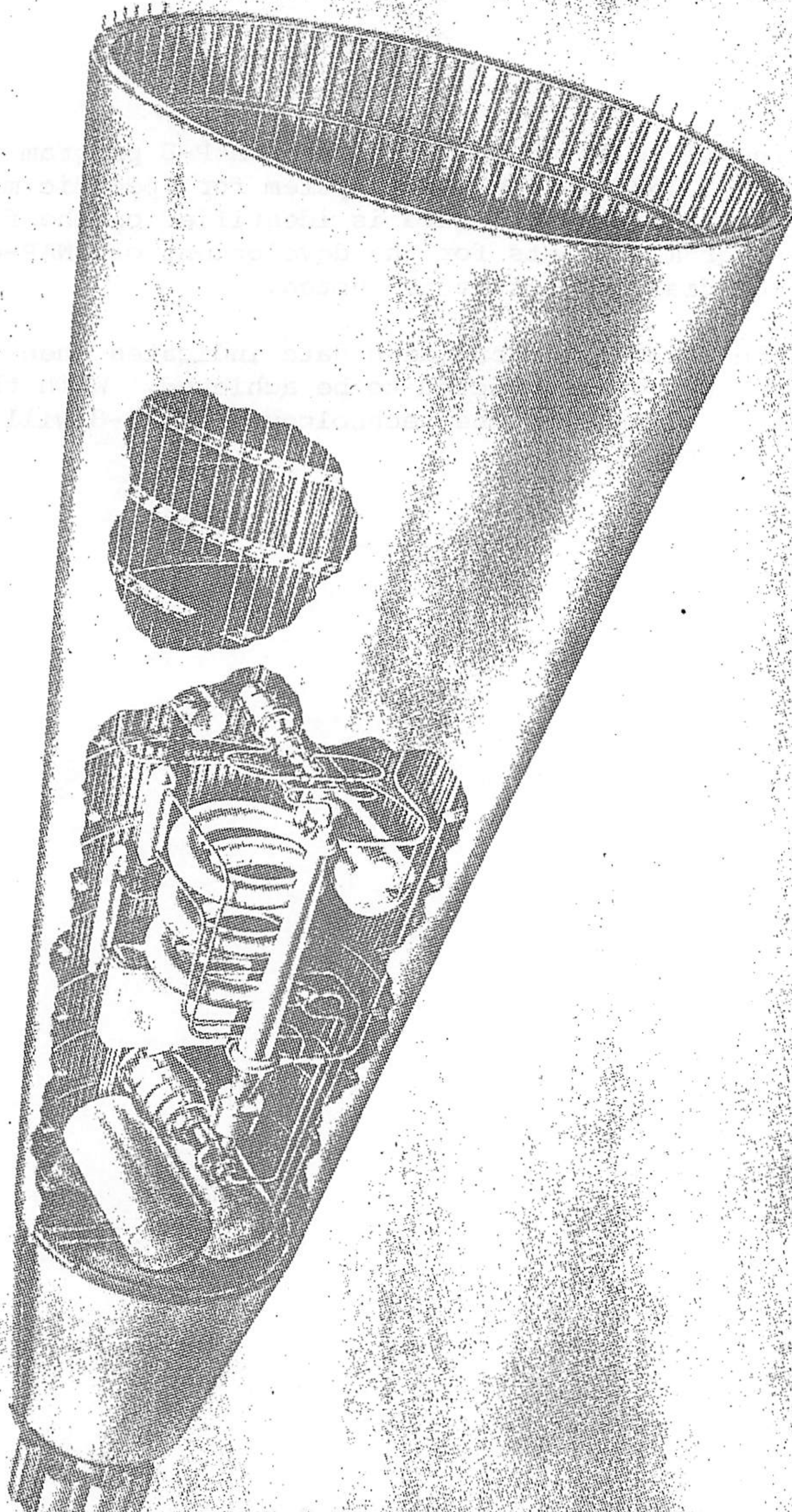
Mercury is used as the Rankine cycle working fluid in the second loop. From the boiler, the superheated mercury vapor is expanded through a four-stage turbine, condensed, and pumped back through the boiler. The heat rejection third loop, again uses NaK; this time to transport waste heat from the mercury condenser to the radiator from which it is rejected to space.

A fourth loop in the SNAP-8 system carries an organic fluid, a mixture of polyphenol ethers (mix 4P3E), that cools the pump motors, the alternator, the space seals, and the electrical controls, and also lubricates the turbine-alternator and mercury pump-motor bearings. The heat from this fluid is disposed of through a separate space radiator.


SNAP -8 POWER SYSTEM
LUBRICANT PUMP
TURBOALTERNATOR
1300°F
1265°F
35KW
REACTOR
BOILER
CONDENSER
665°F
250°F
RADIATORS
210°F
1100°F
505°F
495°F
NaK PUMP
MERCURY PUMP
NaK PUMP
NASA RN 65-561
2-1-65

The ultimate objective of the SNAP-8 program is to develop a long life, 35KWe space power system for specific mission use. An intermediate objective is identified on the figure. The major program elements for the development of SNAP-8 to meet the immediate objective are noted.

The technology readiness date indicates when the intermediate objective is expected to be achieved. With the achievement of this objective, the technology of SNAP-8 will be ready to support mission selection.



SNAP-8

SNAP-8 PERFORMANCE

System performance has been estimated for SNAP-8 (single PCS) in a configuration typical for unmanned missions where thin shields are adequate. Electrical poweris that available at alternator terminals. "Demonstrated" capability is based on actual performance achieved by the first power conversion system using first generation hardware. Projected mission capability is based on component improvements not requiring major additional development.

System performance has also been estimated for SNAP-8 as the main power unit for a manned, medium-sized space station (MORL class). Power output has been conditioned for the specific requirements of the station. Dual PCS's and redundant components have been added to the power system for extended life and reliability. Specific weight (lbs/KWe) is substantially higher than for the unmanned case due to the added weight items noted at the bottom of the figure.

NUCLEAR SAFETY CRITERIA AND PROCEDURES

1. The essential criterion for any mission is that there be no undue hazard to the public or to the operating crews, ranges, and equipment. In each case, an analysis is made of all the possible chains of events which could occur. This analysis produces an estimate of the hazard which would result at the end point of each possible chain of events. Often, the hazard is expressed as a probability function. This analysis is made at time of design; it is updated regularly. It is used as a basis for planning experimental and analytical verifications programs. This analysis affords an opportunity for management to evaluate the potential hazards and make decisions using the information contained. It is essentially identical to the range safety evaluations made for booster safety analysis. In fact, the injury probabilities used for booster launches have been used in the past as a guide in evaluating safety of SNAP devices. By this early evaluation and continual re-evaluation, we assure that no surprise situations delay launch or require large design changes.

2. This "hazards tree" analysis provides a basis for planning our supporting R&D program as well as the experimental programs for verifying assumptions of the analysis of each specific system. AEC is spending $6,000,000 on supporting nuclear safety research not including that spent on specific system projects.

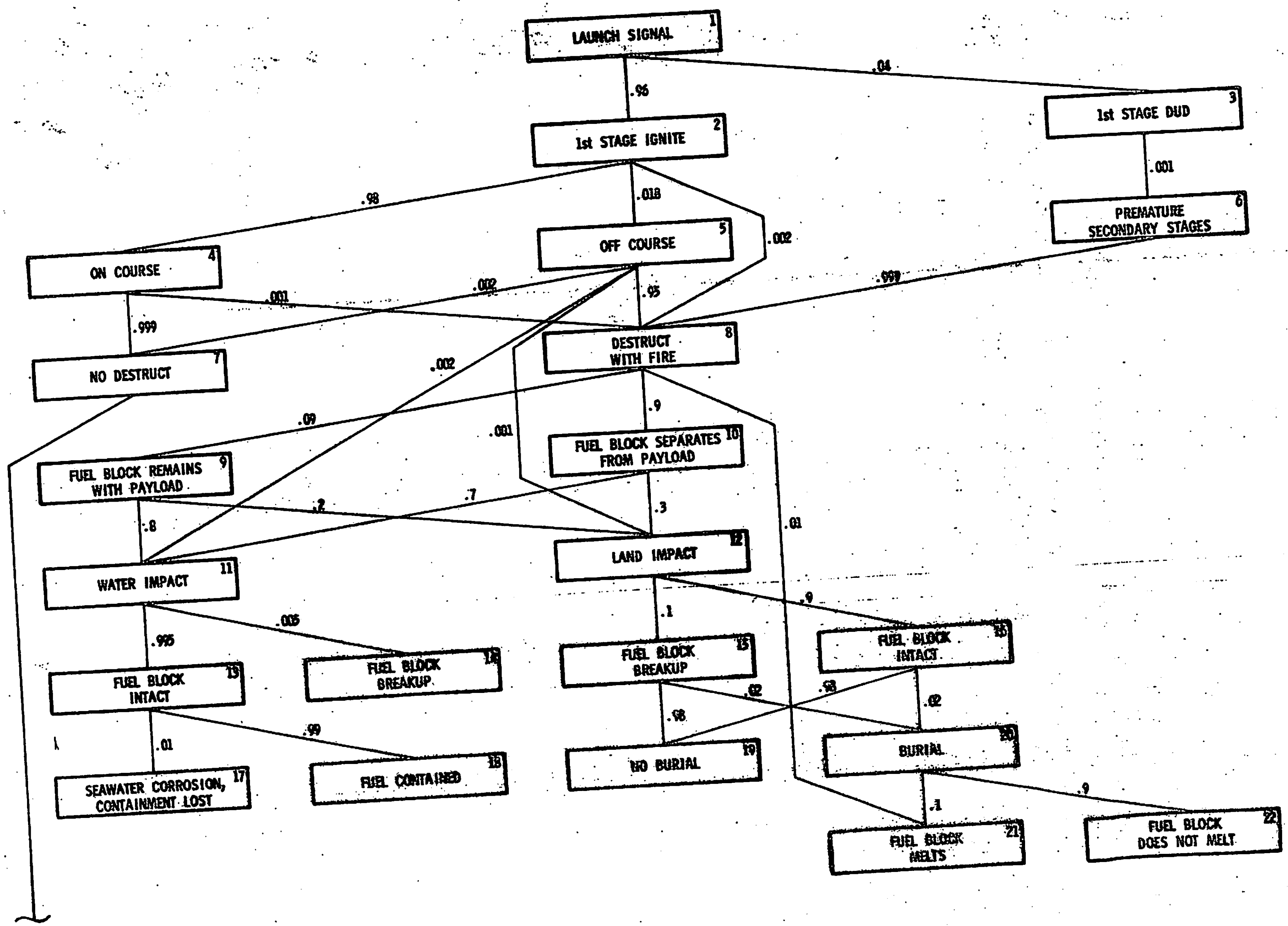
3. We are developing a systematic procedure for each system to follow in gaining approval to launch. We are participating with NASA in a joint safety program on SNAP-19 and SNAP-27.

ISOTOPE GENERATOR HAZARDS FLOW CHART

The detailed flow chart shows the types of events and events network that are examined in a typical flight safety analysis. Each event has a probability associated with it, and system safety criteria are specified to counter each event.

1

SNAP-29 FLIGHT SAFETY ANALYSIS
TITAN III B MISSILE



GENERAL SAFETY PROBLEMS AND COUNTERMEASURES (REACTOR SYSTEMS)

Operational Phase	Accident	Problem	Countermeasure
Launch (Reactor Subcritical)	Launch Abort	Reactor Excursion	Control System Shims to Maintain Reactor Subcritical
Early Part of Ascent (Reactor Subcritical)	Ocean Impact	Reactor Excursion	Intrinsic Subcriticality of Core When Flooded, Reactor Disassembly
Late Part of Ascent (Reactor Subcritical)	Ballistic Trajectory, Short Orbit	Reactor Excursion	Same as above and Reentry Disassembly of Core
Post-Mission Disposal (Reactor run at Power)	---	Random Earth Impact	Long Orbit Lifetime for Decay or Deorbit in Ocean for Refractory Systems Burnup of Zr-H Systems.
		Contamination at Impact	Decay, Impact Control in Ocean
		Direct Radiation	Decay, Impact Control In Ocean

GENERAL SAFETY PROBLEMS AND COUNTERMEASURES (ISOTOPE SYSTEMS)

Operational Phase	Accident	Problem	Countermeasure
Launch	Launch Abort	Site Contamination Direct Radiation Off-Site Impact	Fuel Containment Exclusion, Source Recovery Range Safety System
Early Part of Ascent	Ocean Impact	Off-Range Impact Marine Contamination	Range Safety System Fuel Containment, Insoluble Fuel, Decay
Late Part of Ascent (Near Orbital)	Ballistic Trajectory	Random Earth Impact	Trajectory Shaping, On-Board Safety System to De-orbit
	Short Orbit	Contamination at Impact Direct Radiation	Containment, Insoluble Fuel Distribute Contained Source to Minimize Dose or Track and Recover
Post-Mission Disposal	---	Random Earth Impact	Long-Orbit Lifetime to Decay or Deorbit in Ocean Below 5000 meters for Short Missions
		Contamination at Impact Direct Radiation	Fuel Containment, decay Decay or Impact Control