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⑥ FEASIBILITY STUDY OF
NUCLEAR POWER FOR DISASTER CONTROL
AND CONSTRUCTION BATTALION OPERATIONS,
BEACHHEAD PLANT

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W108/2161/01

Department of the Navy
Bureau of Yards and Docks
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Approved by:

S. Baron

Dr. S. Baron
Vice President, Nuclear Engineering

⑪ 8 July ~~1963~~

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I. INTRODUCTION

This report presents the results of a study of the feasibility of nuclear power plant application to beachhead operations by the military. The study is a portion of Burns and Roe, Inc. obligations under contract NBy-45824, to the Bureau of Yards and Docks, Department of the Navy.

The object of the study is to provide a general plant description, an assessment of the feasibility of the application, a cost estimate, recommendations on power generation methods and technical plant requirements. A prototype plant will be required before production models can be built.

II. CRITERIA

The beachhead plant, power level up to 100 kw electric, will be used to satisfy the need for replacement of the direct-drive diesel engines and the diesel electric generators used at present by the Navy during beachhead operations.

During discussions with Navy personnel, including Construction Battalion and Marine Corps experts (see Appendix A, Fact Sheet on Beachhead Operations), the requirements for power sources with the following properties was developed:

- a. Light weight
- b. Small volume
- c. Transportability and relocatability
- d. High reliability
- e. Relatively low noise level
- f. Rapid startup and shutdown
- g. Plant multiplicity and dispersion
- h. Minimum operating crew
- i. Low cost
- j. Minimum requirements for logistic support such as fuel, cooling water and lubricants
- k. Minimum site preparation
- l. Some degree of resistance to weapons attack and shock

The diesel plants utilized at the present time satisfy each of these criteria to some degree. A drastically reduced fuel requirement is the only advantage of the nuclear power plant which might allow it

to achieve some measure of competition with diesel generation in the near future. The diesel plant has undergone considerable development, and relatively high-speed diesel generators are currently meeting the military requirements.

Military operations by the Navy, at or near beachheads, are frequently dependent upon air support. The Marine Corps has definite weight and volume limits for materials handled during beachhead operations. These mobility considerations stem from the load carrying capacity of helicopters and light military field vehicles. In addition, the rating of the diesel generators utilized in beachhead operations is limited to about 100 kw. In Marine Corps Force Service Regiments or comparable Construction Battalion Units, the requirements for dispersion and operational flexibility necessitate the use of compact power plants. Perimeter-type military operations do not permit the stringing of power wire over long distances and the power loads at any particular location are limited. For these reasons the lowest power level specified in the Scope of Work for the Beachhead Plant was selected for major consideration. This power level was set at 100-kw electric output.

Based on this information on beachhead operations requirements and on the criteria for the Army ML-1 plant, typical criteria for a beachhead nuclear plant capable of replacing diesel units were formulated. These appear as Appendix B of this report.

III. REPORT OF INVESTIGATION AND STUDY

a. Shielding During Operational and Nonoperational Periods

A reactor that is built as a completely shielded unit will not be able to meet the weight limitations for an operational beachhead plant. Supplementary field shielding during operations is necessary. The exact form of this shielding depends on the actual reactor design and beachhead operations, but enough of it should be removable to permit plant relocation within the times specified. The use of an exclusion area to minimize shielding weight is of only marginal value, though it may be useful to control the access of unauthorized personnel to areas of high radiological hazard. Weight penalties are introduced into the system by the provision of long control and power cables needed where a large exclusion radius is used.

It is desirable that adequate shutdown shielding be provided on the reactor to allow its relocation without the need for additional supplementary shielding during the shutdown period. This may be possible if advantage is taken of such things as exclusion distances, relaxed radiation exposure criteria, short exposure times and shadow shielding. The complex interplay of many factors such as time of operation, integral and external shielding, mode of plant operation and relocation, vehicles available for transport, shutdown core and activation radioactivity must be considered in the final design of the optimum plant.

b. Afterheat Removal and Reactor System Cooling Period

The beachhead plant unit should not require a special emergency afterheat removal system. The primary coolant loop can be utilized to remove heat from the unit for a short interval after reactor shutdown, if necessary. The plant should be capable of rejecting enough of its afterheat to its environment under all normal and most abnormal conditions to prevent core meltdown or other extremely hazardous conditions. Plant mobility criteria dictate that the required afterheat cooling period be short.

c. Resistance to Weapons Attack

The hazardous nature of the conditions under which the beachhead plant is expected to operate, and the need for a very high degree of unit reliability require that the plant vulnerability to weapons attack be minimized. It is essential for the plant to be as compact as possible if it is to be located aboveground. The massive radiation shielding required for such a plant will act as armor for the unit. Shield penetrations, the control unit, and the air-cooled condenser will be the especially vulnerable parts of the system. An underground plant is highly invulnerable to attack. If parts of the plant, for example the air cooler, are located aboveground the vulnerability will approach that of the surface plant. The beachhead plant should be so mobile that advantage can be taken of the natural invulnerability of certain sites, i.e., the plant should not be limited by its transportation means to only a few types of sites such as large, flat, open areas.

d. Containment and Hazards

A vapor containment shell, as is usually used for civilian power stations, is not required for the beachhead plant. In evaluating the additional hazard introduced by the reactor plant, consideration must be given to the fact that the environment in which the plant is expected to operate is intrinsically hazardous even without the reactor. The unit itself will of necessity be engineered for high reliability of operation. This fact implies that the probability of accidents or incidents that might take the plant out of operation will be minimized. As a consequence, hazards to personnel will also tend to be minimized. In light of the foregoing, it is obvious that containment of the type specified for civilian units is entirely unnecessary in military plants operating under beachhead conditions.

The reactor will operate as a closed unit. No field refueling or reactor maintenance will be needed. The primary system will be sealed and no radioactive wastes can be expected from this source. Operations should be carried on in such a way as to minimize the production of high levels of induced activities. The release of radioactive materials to the environment is permissible under the supervision of qualified personnel, so long as appropriate military radiological standards are adhered to (see q. Radiation Exposure Criteria). Care should also be taken to minimize nonnuclear hazards such as fire.

The possibility of weapons attack, discussed in the previous section, is an important facet of the beachhead plant hazards analysis. The reactor itself is relatively invulnerable to direct attack because of the very large amount of shielding used around it to protect operating personnel from radiation. The total plant must be so designed that damage to a more vulnerable part such as the air cooler, or a sudden loss of load, or any loss of coolant flow will not result in irreparable damage to the reactor unit. Moreover, to ensure maximum plant reliability, any plant components particularly vulnerable to attack should be easily replaceable in the field, and enough spares should be provided to accomplish this quickly.

The direct destruction of the reactor and primary system is potentially an extremely hazardous situation because of the inventory of fission products contained in the reactor. The hazard of release is lessened somewhat, however, by the fact that the reactor will have operated a relatively short time and its inventory of fission products is not likely to be large. In addition, as stated above, the reactor will be buried or otherwise encompassed by a large amount of shielding material. Only a very heavy attack, which would be catastrophic anyway, could directly damage the reactor under these conditions. It is essential that the power plant be as compact as possible to minimize the sensitive volume of the plant and to avoid identifying the unit as a specific target.

e. Field Maintenance

Requirements for field maintenance of the beachhead plant should be kept to a minimum. Any maintenance work necessary should be of a very simple nature and require little training of the personnel involved. Field maintenance personnel should be capable of servicing several reactors each and replacements for this personnel should be readily available. The plant should not require any maintenance work on the reactor itself or on other highly radioactive components during beachhead operations.

f. Plant Life

The nature of beachhead operations is such that the small mobile power plants will be needed for only a comparatively short duration -- less than three months. Base development electric load growth and the building of a power distribution system will necessitate the installation of a plant of the advanced-base type to produce power after the beachhead is secured. Thus, ideally it might be best to simply abandon the mobile plant after 3 months. The high cost of enriched uranium reactor fuel, however, dictates that the core life be at least 1 year. Several means of resolving this problem can be considered. The mobile plants could be used for power production in peripheral or remote areas even after the advanced base plant has begun operation. It is also possible that beachhead plant units could be recovered and shipped to another beachhead. Cores with remaining life might be removed from the power units and installed in

new units, although this procedure would introduce the problems of field refueling. Another possibility is the use of the beachhead plants or their cores as modular units for building up the advanced base plant.

The ML-1 criteria specify a core life of 10,000 hours. This is considered to be a reasonable criterion for the beachhead plant. It is therefore concluded that any plant developed for beachhead use should be capable of 10,000-hour operation without major overhaul or core refueling.

g. Available Heat Sinks

In order to achieve the maximum mobility and versatility of the beachhead plant system, it was decided that it must be capable of operating without cooling water. Thus the plant should reject its heat to the atmosphere and/or to the earth if it is an underground installation. It should be possible to modify such a plant fairly easily to allow it to reject its heat to water, if an adequate supply of cooling water exists at the plant site. The nature of the operations envisaged for the beachhead plant are such that a completely underground plant, rejecting its heat only to the surrounding earth, would be best. The necessity for air-cooling equipment may introduce problems into plant operation, such as increased vulnerability of the unit and increased noise level due to fan operation. However, since the diesel engines currently used are similarly limited by a need for air, these

factors should not necessarily rule out the air-cooled beachhead nuclear plant.

h. Electric Load Requirements

The beachhead plant should supply the amount of electric power needed in the form required by the loads it supplies. At the present time these loads are supplied by diesel generators of up to 100-kwe size. Based on present needs and the criteria for the ML-1 plant, the plant output characteristics should be as follows:

1. Nominal Output at "Design Conditions" (ambient temperature -65°F to 100°F)
 - (a) Electrical power output in the range of 100-500 kw at 0.8 power factor
 - (b) 2400/4160 volts, 3 phase, 60 cycles per second
2. Output at "Off Design Conditions"
 - (a) Net output of the system in hot weather (100°F) or high-altitude operation shall be limited only by the plant heat rejection capacity.
 - (b) Operation shall be possible up to 500-kw net output except as limited by plant heat rejection capacity.
3. Power Quality
 - (a) Steady state and transient frequency and voltage control characteristics should meet applicable military specifications.
4. Other Requirements
 - (a) Automatic power level control is required.
 - (b) The power plant shall operate satisfactorily in parallel with other units of similar rating.

- (c) The power plant shall be capable of 50-cycle operation with correspondingly reduced power output.
- (d) The power plant shall be treated for the elimination of interference with radio communication in accordance with applicable specifications.
- (e) Requirements for 400-cps power may also be considered.

i. Power Plant Mobility

The beachhead nuclear plant should be at least as mobile as is the diesel generator currently used for this application. It should be possible to move the plant with only the aid of trucks, helicopters and lifting equipment normally found on the beachhead. This limits relocatable plant weight to 2500-5000 lb total, based on vehicles available today. Site preparation time should be minimal -- no more than a few hours -- at both the old and new sites of a relocated plant. The plant should be moved as a unit, if possible. If several packages are required, the attendant disassembly and assembly work needed should be limited to a few hours' duration and to simple operations performed by relatively untrained personnel.

The nature of beachhead operations is such that frequent relocation of the plant may be necessary. This is especially true during the first few weeks of operations when the situation is changing rapidly, all units are quite mobile, and loads are variable in size and location. Fortunately, the reactor unit will be most readily relocatable during the early part of its

operation when such movement is necessary, since the inventory of contained fission products and induced activities will still be relatively small. At later times, the shutdown radioactivity will have increased but the mobility requirements of the plants will be reduced.

j. Plant Operating Crew, Automation and Instrumentation

The plant should be capable of operating with a minimum amount of attendance. A crew of one man per shift is deemed adequate for normal operation of the plant. Highly trained personnel should not be required. Replacement personnel for these operators should be readily available. The services of support personnel -- a mechanic-electrician, a radiological safety technician and an officer or NCO in charge -- can be shared by several beachhead plant units. Maintenance, installation and relocation of this unit in the field will also be carried out by these personnel with the possible assistance of a minimum number of troops.

The plant should be self-regulating and automated to the degree necessary to assure its reliable operation by one man under beachhead conditions. The system should be as simple as possible, reliable and easily maintained. Where manual control can increase the system simplicity and reliability, it should be used. The use of sensitive and highly sophisticated electronic equipment of questionable reliability, requiring specialized maintenance, should be avoided.

Process control and radiation monitoring instrumentation should satisfy military criteria for accuracy, reliability and ruggedness. Instrumentation should be kept to the minimum required for safe and reliable plant operation. Spares should be supplied as required but the emphasis should be on the provision of a system requiring minimum maintenance and parts replacement. Instruments should be easy to read, and the significance of instrumentation readings should be immediately evident to the operators.

k. Speed of Plant Location

In order to be useful for beachhead operations, the plant must be easily and quickly relocatable -- similar to the diesel unit. The 12- and 24-hour location and relocation times are about the maximum that can be allowed during the early phases of beachhead operations. Under these conditions setup and moving times of less than a few hours are much more desirable. At later stages of plant operation, a day or two required for moving can be tolerated. As has been previously noted, the small inventory of fission products in the plant, after short operating times early in its life, will facilitate comparatively easy and swift plant relocation during this period.

l. Plant Transportation by Truck, Rail and Water

The beachhead power plant must be capable of being transported to the beachhead, moved to its first operating site, relocated as required, removed from the area of operations, and transported

to a maintenance, processing or dismantling facility. In general, the primary form of transportation to the site will be by water, due to the nature of the marine beachhead operations and the absence of airfields suitable for cargo craft during the initial phases of operations. Plants will be sent ashore by landing craft and moved to their operating site by light ground vehicle, or in some instances by helicopter lift. It is obvious that the plant must be light, compact and shipped in one module to satisfy the requirements for early beachhead operation. The units shipped should be complete and ready for operation. This will prevent the possibility of otherwise complete plants being inoperable because of a lost or destroyed shipment of associated equipment.

As now envisaged, the beachhead plant will be transported by rail only in the United States or other support area. Trains may be utilized to ship the completed plant from its fabrication point to port of embarkation, or, on its return, to maintenance or other support facilities from the port.

The limiting criteria is the capability of light truck transport (2-1/2 - 5 tons) or helicopter lift (2-1/2 tons). Units which satisfy these requirements will be transportable by sea or rail.

m. Plant Transportation by Aircraft

During the early phases of beachhead operations, it is probably not feasible to transport the plant directly to the site of operations by means of C-130 aircraft. Air transport

can be utilized over part of the trip from the support area if necessary, but the unit would be transferred to a marine vessel for beachhead landing.

Plants that are to be utilized only in operations which take place after airfield facilities are available may be shipped by C-130 or other aircraft. Since at this later time the beachhead power plants do not require the degree of mobility needed during earlier operations, a larger, less compact plant is acceptable. However, such a plant does not satisfy the criteria previously established, and it may be too large for use during early beachhead operations which require a very high degree of plant mobility.

In order to be consistent with other criteria for the beachhead nuclear power plant capable of replacing the diesel unit, one lift or module only can be allowed. The weight of the plant, limited by the requirement for light vehicle transport, will be well within the 20,000-lb limitation of the C-130. In this case it might be possible to select a smaller cargo plane, capable of landing on smaller, less developed air strips, for the transport of the beachhead nuclear plant.

n. Plant Refueling at Site

Field refueling of a beachhead power plant is not required. The design complexities and operating difficulties that would be introduced by the need for refueling are not acceptable for a small, simple, compact plant of this type. The relatively short

duration of beachhead operations makes such refueling unnecessary. Depending on the design of the actual plant, it may be possible, desirable, or necessary to design the plant so that the core can be removed and shipped back to the support area. In this case, however, core removal would be accomplished only after the completion of military operations in the area. Special personnel and equipment would be available at that time to accomplish the core removal.

It is probable that even this minimum defueling operation will be avoided and the plant will be shipped as a unit to the support area where all refueling, maintenance, dismantling and disposal operations that may be required will be carried on for a number of plants.

o. Optimum Plant Arrangement

The optimum plant arrangement for deployment at the site, transportation, ease of field assembly and disassembly depends on the nature of the specific plant design and the actual conditions of beachhead operations. At the present time the design of the nuclear plant that will replace the diesel power units is conceptual only. The engineering problems that will arise in the development of actual units are to a large extent unknown. The nature of beachhead operations as conducted today is known. However, changes in future operations that will greatly affect the use of the power plant are quite possible.

As envisaged, the beachhead nuclear power plant will require substantial external shielding during operation. One way to achieve this is to bury the entire reactor unit. Ideally, only electric power cables would penetrate above the ground; however, the necessity for cooling may introduce ducts or coolers, or even fluid piping aboveground. The relocation of such a buried plant may entail difficulties due to the presence of induced activities. Appropriate operational procedures must be developed if this sort of plant is to satisfy mobility criteria. For this buried plant it is possible to consider a series of very small units consisting of a reactor portion which is abandoned (buried) instead of being moved at each relocation, and a power portion -- not highly radioactive -- which is moved and connected to a new buried reactor. This is a rather expensive arrangement in light of the cost of highly enriched nuclear fuel; however, it would be possible to eventually reuse or reclaim the abandoned reactors.

An aboveground or partially buried system, shielded with dirt and other field material, is also a possibility. Movement of such a unit as a whole would be somewhat simpler than relocation of the buried plant, but operating radiological safety would be more of a problem.

It may be that the actual beachhead plant will be adaptable to many types of field use depending on the terrain and the nature of operations problems encountered in a particular action.

In general, the beachhead power plant design should be as flexible with regard to operations as is possible.

p. Site Preparation and Equipment Requirements

The preparation of the site for a power unit should not require the presence of any equipment or personnel not normally found on the beachhead. The choice of site should not be severely limited by plant arrangement or operating characteristics. Site preparation time should be minimal and should be accomplished by a minimum number of unskilled and semiskilled personnel. Large exclusion areas should not be required.

q. Radiation Exposure Criteria

Radiation exposure criteria are set to assure that the risks to personnel working in a radiation environment will not be substantially greater, due to the presence of radiation, than the risks caused by other hazards to which they are unavoidably subjected. On this basis, maximum permissible doses to personnel in peacetime operations are set at very low levels, commensurate with the small hazards inherent in normal occupations.

The radiation exposure criteria for personnel operating the beachhead power plant should be based on the limits set forth by the AEC in 10 CFR 20. The plant must therefore be fully shielded in areas where personnel access is allowed. The size and weight limitations on plant design severely limit the amount of shielding on the reactor itself. Therefore, provision of supplementary field shielding must be the major means of meeting these radiation criteria.

The relocation or removal of a reactor containing fission products may be the most important cause of radiation exposure during operations. Advantage can be taken of time, distance, manpower and supplementary shielding factors in performing these operations, to meet radiation criteria.

In certain cases, for example under combat conditions, where the environment is intrinsically hazardous and exposures occur over a relatively limited time period, the extra risk introduced by allowing radiation levels in excess of those permitted for civilian continuous exposures by 10 CFR 20 may be acceptable.

r. Vibration

A vapor-cycle power plant employing a well-balanced turbine should produce far less vibration than a comparable internal combustion engine.

Vibration may be reduced still further by supporting all rotating equipment (such as fans and pumps) on well-damped supports. With a reciprocating engine, the vibration is somewhat more intense but still not as severe as that of a comparable diesel engine.

Direct conversion systems which use static devices will be free of vibration in themselves. However, the total system may include pump and fans which introduce some vibration.

s. Noise

No correlation is generally available to accurately determine noise levels emanating from a completely enclosed turbine. The information that is obtainable indicates that this problem is not

particularly serious and the effective control of noise is relatively simple.

The noise problem for vapor-cycle power plants is most serious in air-cooled units because of the fans. The attenuation of noises from other parts of the system can be quite readily achieved, but extreme care must be exercised in minimizing fan and radiator noise to avoid introducing excessive air pressure drop.

Static direct converters are intrinsically quiet devices. However, the systems in which they are used may require the circulation of a coolant fluid. The pumps and fans required will then be a source of noise, just as they are in dynamic systems.

t. Startup

The beachhead nuclear power plant should be capable of being started with a minimum of auxiliary equipment. A plant that can be started up without any external source of power is most desirable. However, the use of a limited amount of supplementary battery power can be tolerated. The system should not require the support of a diesel unit of considerable size.

IV. SUMMARY AND CONCLUSIONS

At the present time, there is no nuclear power system available for use as a beachhead plant. The plant which most closely approaches this mode of operation -- the gas-cooled reactor, closed-cycle power plant ML-1 (AEC Army Reactors Branch designation) -- has weight, volume and other operational characteristics which seriously limit its mobility and its adaptability to the contemplated use in this study. There are, however, a number of nuclear power plant systems under development or in the research stage which show promise of eventually meeting the stringent operational requirements of a beachhead plant. The following are some types of nuclear electric power supplies for 100-kwe output:

<u>Energy Source</u>	<u>Conversion System</u>
Hydride Moderated Liquid Metal Cooled Reactor (SNAP-8 type)	Mercury Turboelectric (SNAP-2,-8 type)
Hydride Moderated Liquid Metal Cooled Reactor (SNAP-8 type)	Advanced Mercury Turboelectric
Advanced Fast Reactor Liquid Metal Cooled	Potassium or Rubidium Turboelectric
Advanced Fast Reactor Liquid Metal Cooled	Advanced Alkali Metal Vapor Cycle (Na, etc.)
Hydride Moderated Liquid Metal Cooled Reactor (SNAP-8 type)	External Thermoelectric Converter
Water Cooled Reactor	In-Core Thermoelectric Converter
Advanced Fast Reactor Liquid Metal Cooled	External Thermionic Converter
Thermionic Fuel Element Advanced Reactor	In-Core Thermionic Converter

Future changes in military operations can greatly influence the feasibility of nuclear power plants for beachhead use. At the present time, power fuel requirements represent only a small portion (10 percent or less) of the overall fuel supplied to a typical beachhead. Under existing conditions, fuel must be supplied for planes, helicopters and ground vehicles. Therefore, a reduction in fuel requirements, due to the replacement of the diesel generators, will not significantly affect the logistic situation and cannot justify the abandonment of the proven diesel units for relatively unproven nuclear power plants. If, however, the overall fuel supply problem is solved by the use of power sources independent of continuous ship-to-shore fossil fuel supply for all propulsion plants, the case for utilizing nuclear power plants will become strong. The availability of independent power supplies for field vehicles such as trucks, tanks, tractors and half tracks, as well as planes and helicopters, will affect the total load to be met by the beachhead plants. Moreover, although at present the battle communication system power load is easily handled by transportable diesel generators, the ever-increasing field communications requirements will ultimately lead to larger load duties, a situation which will tend to favor the use of nuclear power sources.

Electric power generators utilizing the decay of unstable radioactive isotopes to produce heat and energy have been developed and are presently available. The high cost and low output characteristics of these plants eliminate them from consideration for beachhead plant use.

The load-carrying capability of helicopters and light ground vehicles during beachhead operations is limited to 2500 lb. Based on the previously established maximum power plant load capability of 100 kwe, the desired weight characteristics of the plant are fixed.

There is a major technical drawback associated with nuclear plants that greatly influences their ability to satisfy the requirements of the beachhead plant. It is the inevitable presence of penetrating nuclear radiation both during and after reactor operation. The massive nature of the shielding needed to attenuate this radiation precludes the operation or relocation of a fully shielded plant as a truly mobile unit.

Operating nuclear power units utilizing conventional shielding materials like water, steel and concrete have weights which greatly exceed 10,000 lb. A compact reactor itself is quite small and is easily moved to its operating site before initial operation. After it has been operated, however, dismantling or removal of the reactor and parts of the associated plant becomes extremely complex and requires specialized equipment and personnel, due to the residual radiation hazard. The magnitude of these problems can be minimized by appropriate engineering design.

The optimization of reactor plant shielding can involve the design of partial and shaped shields; the provision of exclusion areas; the utilization of field expedient shielding materials such as earth and water; the appropriate choice of materials for core, structure and coolant; the use of special shield materials such as

depleted uranium, tungsten and lead impregnated with organic material; the minimizing of reactor core volume; the elimination of shield penetrations; and the judicious relaxation of shielding and radiological exposure criteria. Significant weight reductions resulting from such shield optimization can enhance the mobility, relocatability or expendability of the unit to such an extent that its logistic advantages outweigh the disadvantages incurred by the presence of nuclear radiation. It is likely, however, that the shield will always comprise a major portion of the total plant weight. The following analysis shows the specific weights of component parts of a typical nuclear, rotating conversion machinery power system in the power range of 100 kilowatts to 1 megawatt:

<u>Component</u>	<u>Specific wt(lb/kw)</u>
Reactor and associated equipment including shielding	3.2
Conversion equipment	
Rotating machinery	2.5
Radiator	2.0
Heat exchanger	0.3
Miscellaneous - controls, piping fluid inventory	0.5
<hr/>	
Total	8.5 lb/kw

To minimize reactor and shielding weight, an extremely compact core with a volume of 1 to 2 cubic feet or less is required. The use of highly enriched fuel is therefore necessary and the resulting nuclear characteristics indicate that, in general, a fast or intermediate core neutron spectrum is expected. Advanced types of nuclear

fuel will be required for these highly compact cores. Fuel element materials now under development which show promise for such applications include uranium oxides, carbides, nitrides and mixed metal hydrides. The effect of fuel form and fuel element shape, the necessity for cladding, and the advantages of using moderator materials in homogeneous and heterogeneous compact cores are being studied. It is impossible to state, at the present time, the exact form of nuclear fuel that will ultimately prove best for this application.

The necessity for using a very compact reactor core and small total system suggests the use of liquid metal as a heat transfer agent. Depending on the conversion system and cycle chosen, one- or two-phase reactor cooling is possible. An indirect power cycle can be utilized, although a direct cycle may lead to a somewhat more compact system. Nuclear as well as chemical and physical properties of the coolant must be considered. Problems associated with the activation of circulating liquid-metal coolant during operation and after shutdown may dictate the use of specially selected, separated nonactivating isotopes to minimize shielding weight requirements. Coolants and cycle fluids currently being investigated include: sodium, potassium, lithium, cesium, mercury, rubidium, sulfur, water and various isotopes of these elements.

Although conventional control rods are not feasible in such a reactor, because of the large volumes and shielding weights associated with their use, mechanically moving control elements (e.g., reflector drums) may be used. Plant attendance during beachhead operations is feasible, but the operation of the plant should not require highly trained personnel.

The object of the current Systems for Nuclear Auxiliary Power (SNAP) program is to develop power plants with weights of 10 lb per kwe, including reactor, power conversion unit, radiator and any necessary shielding. Usually the shielding considered for these space power plants consists of just enough material to protect sensitive equipment in proximity to the reactor from radiation damage. If it is desired to provide shielding on the beachhead plant reactor, adequate to allow personnel access to the plant after reactor shutdown, the total weight of the unit will be substantially greater than that of a similar unit used unshielded for space power.

The power conversion system must match the reactor in compactness and light weight. At the present time, three systems show promise for this application. All are under development but none is ready yet for efficient practical application.

The following tabulation shows the estimated minimum weights in pounds per kilowatt for unshielded nuclear electric power supplies:

<u>Conversion Device</u>	<u>Generator Capacity Kilowatts</u>		
	<u>10</u>	<u>100</u>	<u>1000</u>
Turbo- generator	55	20	10
Thermo- electric	60	27	20
Thermionic	40	17	7

The turbomechanical system--a single-shaft combined rotating unit operating on a liquid-metal power cycle--is closest to realization. Several of the space reactor projects such as SNAP-2, 8 and 50 utilize

this type of conversion system. This system will generate a-c power at 400 cps or more. Thermoelectric conversion has been used with radio-isotopic and other heat sources, but has not as yet been efficiently applied in a reactor power unit, although development work on such systems is being conducted on SNAP-10A. The thermoelectric system is intrinsically a producer of d-c power, and conversion to a-c power requires additional equipment. Thermionic conversion will require more development effort than the other two systems. However, it appears very promising for eventual operation in conjunction with a reactor heat source, using the thermionic conversion units either in the core itself or heated by the primary reactor coolant outside the core. The device is a d-c source at present but may eventually be adapted to a-c power production. The feasibility and preliminary design of several configurations of thermionic reactor systems are being studied and the conversion devices themselves are under development. It is impossible to state today which of these three conversion systems will eventually prove best for the beachhead plant.

For beachhead plant application, the thermoelectric and thermionic power conversion methods are desirable because of the absence of rotating equipment. Thermionic conversion gives indication of having higher potential efficiencies than the thermoelectric conversion methods. The materials problems requiring solution for both the thermoelectric and thermionic conversion schemes are formidable and will require significant advances before realistic systems can be designed.

The heat rejection requirements of the beachhead plant are somewhat less stringent than those of a comparable space power unit. The latter can reject heat only by radiation in the space vacuum. For beachhead operations, heat rejection will, in general, be to air or ground to enhance the plant mobility and render the plant independent of a water supply.

Depending on the nature of the plant available, and field operation requirements, several operating configurations can be envisaged for the compact nuclear plant. In order to match the mobility of the diesel, the nuclear plant can be truck- or trailer-mounted. Since the reactor will be moved after it has operated, the mobile unit must be provided with adequate shutdown shielding. During operation, however, site materials like water and earth can be utilized to enhance the shielding of the plant to an appropriate degree. These materials will be removed when relocation is required and similar barriers set up at the new site. Other plant arrangements can sacrifice mobility for safety and simplicity. The partially shielded reactor system can be placed at the bottom of a 12-15-foot-deep hole and covered with earth. The electrical leads, controls and the heat rejection system, if necessary, will be aboveground. Setup and operation will be simple, but eventual removal of this plant will require special operations because of the presence of induced activities. The complexity of these operations depends on the amount of shielding provided on the unit.

The accurate determination of the earliest availability of appropriate nuclear reactor systems is not feasible at this time. The SNAP-8 reactor

for use in space is scheduled for demonstration operation at 30 kwe, for 10,000 hours on the ground in 1965. This reactor system will not be directly suitable for the beachhead plant requirements.

The ORNL Intermediate Reactor, somewhat more advanced and more compact than SNAP-8, is scheduled to be run as a reactor experiment in 1965. The total system will probably not be tested until 1967, and the design of a prototype for the beachhead plant cannot be started until at least 1968. The availability of SNAP-50 is probably at least three years later than this, as is the development of the out-of-core thermionic systems. The development of a useful in-core thermionic reactor is estimated to be at least 10-15 years in the future (to 1978 for prototype).

The space reactors will become lighter and more compact as they develop, from SNAP-2 to SNAP-8, to the ORNL Intermediate Reactor, to SNAP-50 and the thermionic reactor. At what point these power plants will become small enough for beachhead plant applications is not immediately apparent.

The selection of a power plant for beachhead operations depends on the following:

- a. Advancement of the technology of nuclear fuels, reactor control, efficient power conversion systems, materials, etc.
- b. The development of a nuclear reactor system, including reactor and power conversion system, capable of 100-kwe output, which weighs 25-50 pounds per kwe or less, including appropriate shielding and which does not require highly trained personnel or setup time in the field.

c. The development of a military requirement for such a plant in beachhead operations which will justify the replacement of diesels.

It is concluded that:

a. There is no nuclear power plant that will be available in time to meet the original schedule stipulated for this plant in the Scope of Work statement.

b. There is a good chance that such a plant will become available within the next 20 years, due to the space nuclear power development effort.

APPENDIX A.

FACT SHEET ON BEACHHEAD OPERATIONS

1. The power plant will reject heat to the surroundings by means of an air-cooled heat exchanger.
2. Beachhead, landing and tactical operations will be completed within three days. Base development will be initiated after the first day of beachhead operation. Base development will be a continuous effort for 30 to 60 days.
3. Requirements for power plants during tactical operations include dispersion, quietness of operation, portability (trailer or truck-mounted) and ship-to-shore delivery systems for fossil fuel.
4. Power plant size will be limited to 100-kwe output. Power plant weight will not exceed 2500 pounds.
5. Tactical airfields, maintenance facilities and communications systems represent the major power demand for the beachhead power plant. All equipment utilized in these facilities requires a high degree of portability.
6. Total electric power requirements for services during beachhead operations are of the order of 1000 kw.
7. During combat operations, power supplies are required to have a high degree of reliability. Reliance on a single power source is difficult to justify and a multiplicity of units is indicated.
8. Transportability during beachhead operations is limited by the lifting capacity of helicopters. Helicopters for beachhead operations are capable of carrying 2500 pounds. Five-ton trucks represent the upper limit of ground-vehicle load-carrying capability.
9. Electric power is also required for hospitals, lighting, refrigeration and water purification during initial beachhead operation.
10. Approximately 10 days after beachhead establishment, base development and electric power demand will justify the utilization of a larger power plant such as the advanced base plant. The stringing of transmission wire for load distribution will be feasible at that time.

APPENDIX B

TYPICAL CRITERIA

NUCLEAR POWER - BEACHHEAD PLANT

Title -- Beachhead Plant, 100 KWE

Proposed Use -- There are requirements within the Navy for portable nuclear power plants with electric power outputs of approximately 100 kw. Examples of military operations requiring a power source without dependence on continuous fuel or cooling water supply are: support of beachhead operations and tactical missile systems, air head operations including field hospitals, remote locations.

Environmental Characteristics -- The plant shall have the inherent capability for acceptable performance under the Basic Operating Conditions, Extreme Cold Weather Operating Conditions, and Extreme Hot Weather Operating Conditions and shall be capable of safe storage and transportation under such conditions. A cold-weather kit may be utilized for operation under extreme cold-weather conditions.

Power Characteristics

1. Nominal Output at Design Conditions (Ambient temperature -65° F to 100° F)
 - a. Electrical power output in the range of 100 kw, at 0.8 power factor.
 - b. 2400/4160 volts, 3 phase, 60 cycles per second.
2. Output at Off Design Conditions
 - a. Net output of the system in hot weather (over 100° F) or high-altitude operation shall be limited only by the plant heat rejection capacity.
 - b. Operation shall be possible up to 100-kw net output except as limited by plant heat rejection capacity.
3. Power Quality -- Steady state and transient frequency and voltage control characteristics shall be in accordance with applicable Military Specifications.
4. Other Requirements
 - a. Automatic power level control

b. The power plant shall operate satisfactorily in parallel with other units of similar rating.

c. The power plant shall be capable of 50-cycle operation with corresponding reduced power output.

d. The power plant shall be treated for the elimination of interference with radio communication in accordance with applicable Military Specifications.

Field Operating Characteristics

1. The plant unit shall be assembled, all preparations completed for relocation to an operating site, and shall be capable of being installed and delivering rated power within 12 hours after arrival at the beachhead. The reduction of this time to two hours or less is desirable.
2. The plant shall be capable of relocation to a new site after reactor shutdown, following operation for extended periods at full power. Relocation shall be accomplished within 24 hours after reactor shutdown. The reduction of this time to four hours or less is desirable.
3. Under normal operating conditions, no personnel shall be subject to radiation exposure levels in excess of those set forth in 10 CFR 20.
4. Equipment may be operated in any plane within 5° from level.
5. Control of the power plant will be from a separate shelter which will be transportable as a unit. During plant operation, it shall be connected to the power plant by quick-disconnect electrical cables of sufficient length to permit flexibility in the relative locations of the power plant and control shelter.
6. Plant design shall include necessary safety features for protection of personnel in the immediate area from results of reasonably conceivable mechanical, electrical or nuclear malfunctions.
7. During plant operations, radiation shielding integral to the plant may be supplemented by field expedient materials (earth, gravel, wood, water) sufficient to reduce radiation levels outside the combined shield to within dosage levels prescribed.

8. The operating crew for three-shift operation of the field unit shall consist of no more than one operator per shift. In addition, the services of support personnel -- a mechanic-electrician, a radiological safety technician and an officer or NCO-in-charge -- will be available to each of several beachhead plant units. The crew will carry out operation, maintenance and installation of the plant. Highly trained personnel will not be required.

Transportability Characteristics

1. The plant shall be capable of being transported by a light military ground vehicle overland and cross-country, by water, and by aircraft including helicopter -- in accordance with Military Transportability Criteria. The maximum weight of the unit is 2500-5000 lb depending on the types of vehicles available on the beachhead.
2. The plant shall meet the shock and vibration criteria satisfied by other beachhead equipment; special handling procedures or equipment shall not be required.
3. When in transit, the plant shall have the capability of shallow fording in fresh or salt water. Caution must be taken to avoid a nuclear accident due to core flooding.

Logistic Characteristics

1. The plant shall be capable of operating for 10,000 hours with no more than periodic field maintenance service.
2. The plant shall be capable of operation without a continuous water supply. Any initial quantity of water required for shielding will be kept to a minimum and of a quality which can be supplied by standard military purification equipment.
3. Radioactive waste from the plant will be kept to a minimum. Disposal of radioactive waste will be in accordance with field procedures approved by the Chief Chemical Officer.
4. The plant will be completely self-sufficient to permit steady-state operation without the need for auxiliary equipment, except for process fluid storage and makeup. For startup and shutdown operations, the plant

should require minimum auxiliary equipment. Special equipment for expedient shield construction shall not be required.

5. The power plant shall be equipped with all special-purpose tools required for normal maintenance and startup, but not for refueling. The prototype plant shall be equipped with a Maintenance Package. The field plant shall be equipped with complete operating manuals defining detailed operating, maintenance, installation and relocation procedures.

APPENDIX C
COMPARISON OF SYSTEMS - TABULAR AND GRAPHIC

1. REACTOR SYSTEMS

a. Turbogenerator Systems

<u>System</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>	<u>lb/kwe</u>	<u>Weight - Shielded to 10⁷ R (Shield for Electronic Equipment)</u>	<u>lb/kwe</u>
SNAP-2 NaK Cooled, Hg Cycle (Atomics International)	3 kwe	Reactor - Reflector Assembly	258	86	-
		Primary NaK System	50	17	-
		Mercury Piping System	10	3	-
		Boiler	75	25	-
		CRU Module	45	15	-
		Radiator-Condenser	220	73	-
		Startup System	70	23	-
		Mercury Inventory	75	25	-
		PCS Control	45	15	-
		Electrical Wiring and Insulation	37	12	-
		Structure	55	18	-
		Destruct Charge	10	3	-
		Diagnostic Instru- mentation	50	17	-
		Contingency	50	17	-
		APU	1050	350	-
		Shield	-	-	300
		APU with Shield	-	-	1350
					100
					450

<u>System</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>		<u>lb/kwe</u>	<u>Weight - Shielded to 10⁷ R (Shield for Electronic Equipment)</u>	<u>lb/kwe</u>
SNAP-2 (NaK Cooled, Hg Cycle Expected Performance)	50 kw 3 kwe	Reactor	200-260	67-87	-	-
		Boiler	100	33	-	-
		CRU	50	17	-	-
		Radiator & Liquid	150	7	-	-
		Controls	20	7	-	-
		Structure	50	17	-	-
		Piping	30	10	-	-
		Shield	-	-	300 (190-250)	100
		System	600-750	200-250	900-1050	300-350
	165 kw 11 kwe	Reactor	320	29	-	-
		NaK System	130	12	-	-
		Shield	-	-	(3800)	345
		CRU (3)	1080	98	-	-
		Radiator- Condenser	660	60	-	-
		Total System	2210	220	(9310)	846

<u>System</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>		<u>lb/kwe</u>	<u>Weight - Shielded to 10⁷ R (Shield for Electronic Equipment)</u>	<u>lb/kwe</u>
SNAP-2 Probable Advances	Adv Components 4.5 kwe		900	200	-	-
	Increased NaK Temperature 5 kwe		900	180	-	-
	Twin Adv Turboalternators - 1 Reactor 9 kwe		1260	140	-	-
	Tandem Reactor 18 kwe Each with Adv Turboalternators		1980	110	-	-
	Binary SNAP-2 11 kwe		880	80	-	-
	SNAP-2 Double Power with Twin Binary 21 kwe		1470	70	-	-
SNAP-8 (similar to SNAP-2)	300 kwt 30 kwe (35-70 kwe)	System	1400-1500	40-50-60	1670-1820	56-61
		Reactor	400	13	-	-
		Shield	-	-	270-320	8-11
		System with Shadow Shield for Man (2.6R)			1900-2300	63-77
Advanced Hg 1300° F Reactor, Hg Boiler @1170° F Shadow Shield	100 kwe	System	6000	60	-	-
	600 kwt	Hg Condens-	600°F 3300	30	-	-
		ing Temp. 800°F	2800	25	-	-
		1000°F	2400	22	-	-

<u>System</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>		<u>lb/kwe</u>	<u>Weight - Shielded to 10⁷ R (Shield for Electronic Equipment)</u>	<u>lb/kwe</u>
ORNL Intermediate Reactor (Potassium Cooled Boiling)	1 mwt, 100-300 kwe (125 kwe)			25	-	-
Lunar Plant ANL (Hg Cooled & Cycle, Unshielded)	1 mwe	Pressure Vessel	2024-2340	2.0-2.3	-	-
		Core	1510	1.5	-	-
		Reflector Controls	1260-1404	1.3-1.4	-	-
		Turbine & Pump	700	0.7	-	-
		Generator	900-1000	0.9-1.0	-	-
		Compressor	53-60	0.05-0.06	-	-
		Subtotal	6447-7014	6.4 7.0	-	-
		Base & Scaffold	293	0.3	-	-
		Radiator	5302-11363	5.3-11.4	-	-
		Mercury	2893	2.9	-	-
		Total Plant	14935-21563	14.9-21.6	-	-
SNAP-50/SPUR (Li Cooled K Cycle)	8-10 mwt 300 kwe - 1 mwe 400 kwe	System Goal	-	10-20	-	-
		Reactor	680	1.7	-	-
		Radiator	1320	3.3	-	-
		Turbogenerator	1480	3.7	-	-

<u>System</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>		<u>lb/kwe</u>	<u>Weight - Shielded to 10⁷ R (Shield for Electronic Equipment)</u>	<u>lb/kwe</u>
Li Cooled K Cycle	400 kwe	Shield	-	-	800	2.0
		System	3480	8.7	-	-
SPUR (similar to SNAP-50)	150-300 kwe	System	1185-2370	7.9	1500-3000	10
ACN Program K-Turbogenerator	300-2000 kwe 500 kwe	Reactor	1000	2	-	-
		Auxiliaries	500	1	-	-
		Shield	-	-	1000	2
		Turbomachinery	1250	2.5	-	-
		Radiator	<u>2250</u>	<u>4.5</u>	<u>-</u>	<u>-</u>
		Total	5000	10	6000	12

b. Thermoelectric Systems

<u>System</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>		<u>lb/kwe</u>	<u>Weight - Unshielded to 10⁷ R (Shield for Electronic Equipment)</u>	<u>lb/kwe</u>
SNAP-10A (SNAP-2 Reactor plus Thermo- electric Conversion)	25 kwt 0.5 kwe	Reactor	250-325	500-650	-	-
		Converter	250	500	-	-
		Shield	-	-	190	380
		System	525-750	1050-1500	715	1430
	12 kwt 0.3 kwe	Reactor	250	833	-	-
		Converter- Radiator	100	333	-	-
		Shield	-	-	250	833
		Misc.; Pump Structure	<u>100</u>	<u>333</u>	<u>-</u>	<u>-</u>
		Total System	450	1500	700	2333
	35 kwt 1.0 kwe	Reactor	250	250	-	-
		Converter Radiator	300	300	-	-
		Shield	-	-	300	300
		Misc.	<u>150</u>	<u>150</u>	<u>-</u>	<u>-</u>
			700	700	1000	1000

c. Thermionic Systems

<u>Out-of-Core</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>		<u>lb/kwe</u>
General Atomics	1-50 kwe	260		200-100
	10 kwe			20-30
	300 kwe	Accessory Equipment		1.7
		Reactor		2.0
		Shield		2.3
		Radiation		<u>4.0</u>
		Total System		10.0
STAR-R (GE) (Radiation Cooled)	13 kwe	System	1300-1450	100
	20 kwe	System	1300-1400	60-65
	80 kwe	System	1300-1400	15-20

<u>In-Core</u>	<u>Output</u>	<u>Weight - Unshielded lb</u>	<u>lb/kwe</u>	<u>Shielded for Electronics</u>
STAR-C Liquid Metal (GE) Cooled	500 kwt - 1000 kwe	Reactor System without Shield and Power Condition- ing -- 5000	5	-
		Conversion 300-7000	3-7	-
		Total System 8000-11000	8-11	-
General Atomics Lithium-Cooled	300 kwe	System 1260	4.2	-
Atoms International	200 kwe	Reactor & TI 460-340	2.3-1.7	-
	600 kwe	Reactor & TI 720-480	1.2-0.8	-
	1600 kwe	Reactor & TI 1280-800	0.8-0.5	-
		System Goal	4-5-8	12.5-8.5 lb kwe

	<u>Output</u>	<u>Weight - Shielded for Transistors</u>	<u>lb/kwe</u>
AGN In-Core TI (Shielded for Transistors)	100 kwe	Auxiliary 100	1
		Shield 1700	17
		Radiator 900	9
		Reactor 1400	14
		Total 4100	41
C-8	500 kwe	Auxiliary 500	1
		Shield 3500	7
		Radiator 4000	8
		Reactor 5500	11
		Total 13500	27

2. CONVERSION SYSTEMS

	<u>Present Size Range</u>	<u>Specific Weight</u>	<u>Efficiency</u>
a. <u>Turbogenerator Systems</u>			
<u>Rotating</u>	1000 kw	2.2-4.4 lb/kw	12%
Mercury Turbine (Rankine Cycle)			
Potassium Turbine (Rankine Cycle)			
Gas Turbine (Brayton Cycle)			
<u>Reciprocating</u>	100 kw	9 lb/kw	20%
Diesel Engine (Diesel Cycle)			
Automobile Engine (Otto Cycle)			
Stirling Engine (Stirling Cycle)			
Fluid Engine (Brayton Cycle)			
b. <u>Thermoelectric</u>			
PbTe Converters	3w to 5w	132-44 lb/kw (Predicted for Future)	5-15% (Predicted for Future)
c. <u>Thermionic</u>	up to 300 kw	33-2.2 lb/kw (Predicted for Future) 66 lb/kw (in 2 years) 6-18 lb/kwe (in 5 years)	10-30% (Predicted for Future)

APPENDIX D
COMPARISON OF SYSTEMS - DISCUSSION

1. Reactor Systems

a. Reactor Types

In the range of 100 to 1000 kw the specific mass of the fission heat source is controlled by criticality requirements. Above this rating, the size of the reactor is determined predominantly by heat transfer considerations.

In choosing a reactor type, low specific mass, small shielding cross section, high-temperature capabilities, and compatibility with the remainder of the system are important. The most promising reactors are the liquid-metal-cooled and the radiation-cooled types. Most contemporary studies have concentrated on the liquid-metal-cooled concept but the radiation-cooled reactor also shows a great deal of promise for very advanced applications in the 50-kwe range.

Because of the nature of the Brayton cycle, gas-cooled reactors yield lower effective radiator temperatures; therefore, gas cycle radiators must be larger than those required in liquid-metal vapor cycles, at comparable states of the art, even though higher reactor outlet temperatures are possible with inert gas coolants. The heat source specific masses of these two reactor types are quite competitive above 1 mwt. Below this power level, because of their higher void fraction, the gas-cooled reactors are physically larger and heavier.

The liquid-metal-cooled reactor predominates in the field of space power today because it yields lightweight power plants and is relatively well developed.

b. Reactor Heat Sources

In considering the use of nuclear reactors as heat sources for small mobile power supplies, the reduction of shielding weight to a reasonable value presents one of the major problems. In the mobile nuclear reactor power plants considered thus far, where weight is a limiting factor (such as the Aircraft Nuclear Reactor, the SNAP series of compact nuclear power reactors, and the ANL Lunar Power Reactor), the approach has been to provide very limited shadow-type shielding or, where possible to eliminate shielding entirely. An example of this latter approach is

COMPARISON OF NUCLEAR HEAT SOURCES kg/kwt vs kwt

SPECIFIC WEIGHTS OF REACTOR HEAT SOURCES AND ISOTOPE HEATERS
(NO SHIELDING)

the proposed LPR, with the reactor set into a hole in the moon's surface. The weight penalty imposed by shielding in even the smallest-size reactors may be illustrated by the fact that the SNAP-2 (50-kwt) reactor weight would be increased by a factor of 100 or so if continuous personnel access were required.

The SNAP-2 and 8 reactors, using a homogeneous mixture of zirconium hydride and uranium for fuel elements, are criticality limited below about 1000 kwt; their weight is almost independent of power output and their specific weight shows a sharp reduction with output, up to this power level. Above 1000 kwt the reactors are heat transfer limited; their size increases as output increases and the reduction in specific weight with increased output is much more gradual. Since they are criticality limited, it is not possible to reduce the total weights of the smaller reactors appreciably, although some reduction in size might be possible with the use of small fast reactors rather than the SNAP-epithermal type. The fast reactor systems do offer substantial weight savings over the thermal reactors in larger output power plants since their power densities are higher.

c. Boiling and Nonboiling Liquid-Metal-Cooled Reactors in Rankine Cycle Power Plants

In a study by Aerojet General Nucleonics, reported at the 1960 ARS Space Power System Conference, the relative merits of boiling and nonboiling nuclear reactors for use in Rankine cycle space power plants over the power range of from 1 to 20 mwe were evaluated. Lithium-cooled fast reactors were found to be most suitable for the nonboiling system, and potassium-cooled boiling fast reactors most suitable for the boiling system. Potassium is the Rankine cycle working fluid in each system. A turbine inlet temperature of 2000° F and shielding weights commensurate with manned applications are assumed. It was found that unless risks in reactor safety are accepted in the boiling reactor system, the nonboiling reactor system is more suitable in terms of size and weight. The weight savings resulting from such risks in the boiling system are marginal and may actually be nonexistent. The nonboiling reactor is not subject to the uncertainties caused by boiling in the core, and its development would therefore be cheaper and less involved. Experimental data on boiling

heat transfer and pressure drop are required to completely determine the relative desirability of the boiling reactor system.

d. Reactors for Direct Conversion Systems

Solid homogeneous reactors are of particular interest for untended systems because of their applicability to an extremely simple power unit, containing essentially no moving parts, in which the reactor core is surrounded by thermoelectric elements or other static energy converter. The heat generated in the core is transferred by conduction to the generator where part of it is converted into electricity and the rest is rejected from the outer surface of the converter.

Such a system requires the use of thermoelectric materials which can operate in an intense nuclear radiation environment. Not all thermoelectric materials meet this criterion but materials applicable to this design are under development.

Since enclosure of the entire generator by a shield would introduce prohibitive temperature drops between the thermoelectric cold junctions and the radiator surface, only a shadow shield can be used. This requirement is not a disadvantage in space systems. However, for terrestrial application shielding is a serious problem.

Reactors considered in the study (WADD-TR-60-22) were:

1. U metal, fully enriched, sphere, unreflected
2. UO_2 ceramic, fully enriched, sphere, unreflected.
3. U-Be mixtures, fully enriched, sphere, unreflected
4. UO_2 -BeO mixtures, fully enriched, spheres, unreflected

Solid homogeneous reactors were chosen because it was felt that liquid homogeneous reactors are not desirable for space applications due to problems of high pressure, molecular dissociation, corrosion and leakage.

The gaseous-moderated reactor, on the other hand, was ruled out because of the difficulty of obtaining a compact reactor, with minimum shield weight. Reactors using a metal hydride moderator were not selected because release of hydrogen would be expected to limit the maximum core operating temperature to 1300-1500° F. For a reactor cooled

only at the surface, this would not allow sufficiently high surface temperatures with any reasonable amount of heat generation. Of other possible moderators, D_2O and carbon require large reactor sizes and were therefore eliminated.

The fast uranium metal reactor is the smallest in size, but its melting point temperature is relatively low. The fast UO_2 reactor, has a very high melting point and uses a proven reactor fuel. Thermal reactors having relatively high temperature moderators, U-Be, and in order to obtain even higher temperatures, UO_2 -BeO, were also selected.

Size and Weight of Minimum Critical Mass, Unreflected, Spherical Reactors

	Reactor Diameter Inches	U-235 Loading lb	Total Weight lb
U metal, fully enriched	6.9	106	113
UO_2 ceramic, fully enriched	11.3	229	279
U-Be mixture, fully enriched	36.4	4.6	1550
UO_2 -BeO mixture, fully enriched	47.3	4.2	5570

Although the thermal reactors require a lower fuel loading than the fast reactors, they pay heavily for this in size and weight. This effect will be compounded in an actual application, since shielding size and weight also increase as reactor core diameter increases. Moreover, the low fuel loading is advantageous, since reactor life is affected by the percent of fuel burnup. Life considerations might, for example, make it desirable to operate the thermal reactors at ratios of moderator to fuel which result in higher fuel loadings.

Results of the analysis indicated that the maximum specific thermal output, in watts per lb is given by the U and U-Be reactors, both of which have about 220 watt/lb (4.5 lb/kw) at a reactor surface temperature of 1100° F. The output of these reactors drops to zero, however, at surface temperatures of 2000 to 2300° F. The UO_2 and UO_2 -BeO reactors have specific thermal outputs of 50 watts per lb or less (20 more lb/kw thermal) at surface temperatures of 1100° F, but can

The U-Be and UO_2 -BeO reactors have the greatest thermal output at surface temperatures of 1100°F (about 335 kw and 310 kw respectively), while the output of the U metal and UO_2 reactors is only about 25 and 10 kw respectively at this temperature.

The above calculations were made for a cosine power distribution in the reactor. If the power distribution were flat, the reactor power output would be approximately doubled. Since the amount of uranium in the U-Be and the UO_2 -BeO reactors is almost insignificant in comparison to the moderator, such a flat distribution should be quite possible in these reactors by preferentially loading the uranium. Consequently, the maximum output for the U-Be reactor at a surface temperature of 1100°F would be about 440 watts per lb (2.3 lb/kw) or about 670 kw total.

Of the reactors investigated, the U-Be reactor was found to give the best performance at surface temperatures (about 1100°F) suitable for thermoelectric generators. However, it is apparent that the requirement of removing heat solely by conduction through the reactor core severely limits the heat output obtainable from the reactor.

Reactors using forced circulation to accomplish cooling are of interest because they are lighter in weight per unit power output than solid homogeneous reactors and they are more flexible, allowing removal of the generator to a position beyond the shielding. Generators incapable of operating in a region of high radiation can be used with such reactors.

Water cooling was not considered because of the high pressures and heavy equipment needed to contain the coolant at required temperatures. Organic-cooled reactors were eliminated because of chemical stability problems.

Gases allow operation at higher temperatures than do liquid metals because of fewer material problems. However, these temperatures are not sufficiently high at present to compensate for the relatively poor heat transfer and pressure drop characteristics of gases compared with those of liquid metals. In addition, there has been experience in

the use of sodium and potassium liquid metals, and a considerable amount of information on their properties is available.

Another advantage of liquid metals is the possibility of utilizing an electromagnetic pump with no moving parts in the system, thus maintaining the basic advantages of a static system. A liquid-metal coolant was therefore chosen.

In a study of reactors using forced circulation, a particular design concept was selected and performance calculations were carried out for various fuels and power levels. Specific weight values obtained, including shielding, range from 0.11 to 1.23 lb/kw of thermal output.

These weight values are predominantly functions of power level and output power per unit weight of U 235 fuel. In particular, power output per unit weight of U 235 fuel provides a direct measure of fuel burnup for any given time. Since the percent of fuel burnup is one of the important factors affecting fuel life, it is apparent that weight comparisons based on widely different values of output per unit fuel weight must be made with caution, since they may represent reactors having widely different life capabilities.

One percent fuel burnup is generally permissible, corresponding to about 36.5 kw/kg of U 235 for one year.

Forced circulation nuclear reactors which can operate for a year or more should be capable of achieving weights appreciably less than 1 lb/kw thermal (probably 0.3 lb/kw or even lower) although not necessarily at low power levels (these weights neglect shielding). Forced circulation reactors are thus significantly lighter in weight than solid homogeneous reactors.

2. Energy Conversion Systems

a. Mechanical Energy Conversion

Today direct conversion is superior to electromechanical conversion at low power levels (below 1 kwe), where turbine efficiencies drop rapidly with miniaturization. Conventional heat engines (turbines and piston engines mated with rotating electromechanical generators) are still important at higher power levels, by virtue of their good

efficiency and advanced state of development.

Three cycles are considered -- Rankine, Brayton and Stirling. Among the heat engines with moving parts, vapor (Rankine cycle) turbo-machinery is, at present the major contender in large nuclear space power plants. The inert gas turbine, operating on the Brayton cycle, can generally attain higher cycle temperatures, but the pumping power requirements are also very high and heat rejection must take place at effective temperatures that are low compared with those in the vapor (Rankine) cycles. Stirling cycle gas piston engines have also been proposed. Reciprocating engines operating on the Stirling cycle are competitive in weight with turbines at low pressure levels. Up to a few kilowatts, the Stirling cycle engine is appreciably lighter than a rotating machine even with a low radiator temperature.

The main difficulties with the Stirling engine for longtime operation include lubrication, preventing oil from clogging the regenerator or contaminating the helium working fluid, and vibration.

On an overall specific mass basis, the Rankine cycle is superior to the Brayton cycle. Above 100 kwe the Rankine cycle has a smaller radiator mass; below 100 kwe, the liquid-metal-cooled reactor is smaller. However, if gas-cooled reactors could produce 4000° F outlet gas temperature instead of the 2200° F predicted for 10,000 hour of operation, the Brayton cycle would become quite suitable.

The rotating equipment in electromechanical conversion systems can be integrated into relatively small packages which contribute little (less than 25 percent) to the overall specific mass, i.e., between 1 and 2 kg/kw. This contribution to the total specific mass of the power plant is usually dwarfed by the reactor at low power levels and the radiator at higher power levels.

b. Advantages and Disadvantages of Stirling Cycle

<u>Advantages</u>	<u>Disadvantages</u>
For small sizes, engine is lighter than turbine.	Lubrication problems.
Reliable -- no valves.	Oil clogging of regenerator and contaminating helium working fluid.
Can use inert working fluid, e.g. helium	Inherent low rotating speeds and intermittent operation result in higher weight per unit output for larger units.
Noncorrosive	The multiplicity of rubbing surfaces and rubbing seals in combination with lubricant deterioration and leakage is detrimental.
Good nuclear properties (e.g. low density and small neutron absorption cross section of helium).	It is generally not possible to add heat from the energy source fast enough to give low weight.
Low density -- low weight of inventory.	
Good heat transfer properties.	
High efficiency (close to Carnot cycle).	
Gas cycle -- temperatures not limited by properties of working fluid.	

c. Advantages of Rankine and Brayton Cycles

<u>Brayton Cycle</u>	<u>Rankine Cycle</u>
Higher cycle temperature.	Lower pumping power requirements.
Can be used with an inert gas such as helium, as working fluid.	Higher heat rejection temperature; smaller radiator required.
Noncorrosive.	Higher efficiency.
Good nuclear properties (e.g. low density and small neutron cross section of helium).	Better heat transfer properties of liquid metal.
Low density -- low weight of inventory.	Cycle temperatures limited by properties of working fluid - condensing temperature, vapor pressure, etc.
Good heat transfer properties.	
Sealing problems less critical.	
Adaptable to open cycles (air) for terrestrial uses.	

d. Comparison of Stirling Cycle with other Cycles

The ideal Stirling engine runs more efficiently than any steam or steam-mercury cycle operating within the same temperature limits. The

table below gives results of studies made by the General Motors Research Laboratories of a Stirling cycle compared with other cycles. The study concluded:

(1) For given temperature limits the Stirling engine is the only practical engine cycle to attain maximum thermodynamic efficiency.

(2) At the same minimum cycle pressure, specific output of a Stirling engine is 85 percent of the same displacement Brayton engine, and one third of the same displacement four-stroke Otto engine.

(3) At the same maximum cycle pressure, specific output of a Stirling engine is more than double that of the same size four-stroke Otto engine.

Inherently the Stirling engine has high efficiency, but it must handle large masses of working fluid to attain competitive specific outputs.

	CYCLE COMPARISON (GMC)			
	Otto	Brayton	Carnot	Stirling
Efficiency, %	59.5	59.5	59.5	59.5
Minimum cycle temperature, R	540	540	540	540
Maximum cycle temperature, R	3418	2100	1330	1330
Corresponding Carnot-cycle efficiency, %	84.1	74.3	59.5	59.5
Network output, Btu per lb	211	110	37.7	37.7
Max cylinder volume, cu in.	47	47	47	47
Working-fluid mass, lb	0.0020	0.0008	0.0020	0.0020
Net work per cycle, Btu	0.422	0.088	0.075	0.075
Hp at 3000 cyc per min	29.9	6.2	5.3	5.3
Min cylinder volume, cu in.	4.94	6.10	2.48	23.5
Volume ratio	9.5	7.8	19.1	2.0
Piston displacement, cu in.	42.06	40.90	44.52	23.50
Hp per cu in.	0.71	0.15	0.12	0.23
Min cylinder press., psia	14.7	14.7	14.7	14.7
Max cylinder press., psia	884.0	72.4	688.0	72.4
Pressure ratio	60.0	4.94	46.7	4.94

3. Complete Power Systems

a. Energy Conversion Systems

At the present time, development work on compact energy conversion systems has been largely directed toward space applications. Many of the criteria to be met by these systems are identical to those of the beachhead plant: light weight, compactness and reliability. In some respects the space units are subject to even more stringent requirements such as launch pad safety and operation at very high temperature. Thermoelectric converters are inferior to thermionic converters in both efficiency and specific mass at the operating temperatures of over 1000° F required in the units being developed for use in outer space. Although the initial space power plants employing direct conversion will be the low-temperature thermoelectric types, the more advanced models are expected to be thermionic. The outlook for terrestrial plants is similar.

Thermionic converters are expected to be incorporated into a power plant with a specific mass contribution of 30 kg/kw in two years and 3-8 kg/kw within five years. Thermoelectric elements are twice as heavy, and are limited to applications below 1000° F. The replacement of currently used rotating conversion equipment (1-2 kg/kw) thus incurs a specific mass disadvantage except at very low power (less than 1 kwe)-- the range for static conversion systems. This disadvantage may be acceptable for a reliable, long-lived system.

Heat-to-electricity conversion efficiencies range from less than 10 percent for contemporary thermoelectric devices to 40 percent for a steam plant. The conversion efficiency, however, is not a fundamental figure of merit for compact power units. For example, the conversion efficiency of the larger Brayton and Rankine power plants (over 100 kwe) is usually fixed at between 12 and 20 percent to achieve the minimum power plant specific mass. The minimum occurs because of the opposing effects of high heat rejection temperatures and equipment weight.

Present-day direct conversion techniques have yet to attain the efficiencies required to allow specific mass optimization. Thermoelectric converters operate at about 5 percent efficiency today, with a potential increase to 15 percent or more; thermionic converters now operate at 10

percent with 30 percent predicted. An increase in efficiency beyond 20 percent is not as important for space or military power plants as it is for conventional electric power generation.

b. Thermionic and Dynamic Space Power Systems - Fuel Inventory
Kg U 235 Versus Mwe

c. In-Core Thermoelectric Converters and In-Core Thermionic Diodes

Compared with currently available thermoelectric generators which utilize PbTe, a neutron poison, the thermionic diode has the advantage of introducing no additional neutron absorbing material in the core. This is balanced somewhat by an increase in fast neutron leakage from the larger core needed in the thermionic system.

For relatively large water-cooled reactors, studies indicate that the presence of the vacuum or plasma layer required for thermionic devices definitely affects the neutron economy of this reactor less than does the presence of the relatively high cross-section material PbTe (the thermoelectric material) if cathode and anode materials with neutron absorption not greater than that of aluminum are used.

In a small epithermal reactor, such as might be used in compact power applications, the difference in effect between a thermoelectric material and a vacuum layer will not be as great. If neutron leakage is the strongest effect, as it might be if the thermoelectric material of low cross section is developed, the thermoelectric system could conceivably require less additional U 235 fuel enrichment than does the thermionic system.

4. Compact Reactor Core and Shielding

a. General Shield Criteria

The reactor shield must be designed to satisfy criteria which may be in conflict. These include:

Radiation Attenuation -- The shield must attenuate radiation to levels which do not represent an unacceptable hazard to personnel at operating areas and in uncontrolled areas. In addition, it must attenuate radiation to safe levels for sensitive equipment in the plant itself.

Mobility -- The shield must be capable of being transported to and from the site safely, meeting size, weight, time and radioactivity limitations. Supplementary materials supplied on site must be relatively abundant, effective and require a minimum preparation time. The shield must satisfy these criteria under several operating conditions. Under preoperational conditions the shield is required only to be mobile. It must be brought to the site, or collected at the site, and prepared, erected and tested. During operation at power, the shield is required only for radiation attenuation. It must attenuate core, coolant and structure radiation to levels below permissible limits set for operators in the

control area and for personnel outside the exclusion area. During operational shutdown and relocation operations radiation attenuation is required since the core and other components are still radioactive. Personnel who may be required for relocation operations must be amply protected by the shield in combination with time, distance and exposure criteria factors. The integral shutdown shield must meet both attenuation and mobility criteria. Induced radioactivity is still present, even in parts of the shield itself, and must be considered.

	<u>Requirements</u>	
	<u>Radiation Attenuation</u>	<u>Mobility</u>
Preoperational	None	Total
Power operation	Total	None
Operational shutdown & relocation	Partial	Partial

b. Shielding and Reactor Core Size

Since the size and weight of the shield depends strongly on its inner radius, it is important for the core of the reactor to be as small as possible. Several factors enter into the determination of minimum core size. A basic consideration is critical mass. Unless a critical mass of fissionable material is provided, the reactor will never produce power. The size of the critical mass depends on the nature of the materials used for the core.

If a critical mass of fuel is provided, the core will sustain a chain reaction. It is necessary, however, to load the core initially with an amount of fuel greater than the critical mass. The excess fuel is burned up over the lifetime of the plant, producing power. The initial core, containing excess fuel, is potentially supercritical, and control must be provided to assure that all excess reactivity present is adequately controlled.

At 10 mwt a lightly enriched reactor burns about 5 kg U 235 per year. For control to compensate for this uranium depletion in a small core, the reactivity worth of uranium in the core must be small. This condition is satisfied in epithermal and fast spectrum reactors but not

in compact, completely thermalized reactors which have a small critical mass. For the moderated reactors, this reactivity problem may limit the minimum core size.

Another limitation on the size of a reactor is the need to remove heat from the core. Channels must be provided for coolant passage. These channels introduce void spaces into the core and increase the volume of the total core. The penalty in size due to heat transfer considerations varies with the core power density, and thermal properties of fuel, moderator and coolant. In many cases these considerations are limiting in determining minimum core size.

Small cores have a high power density, requiring a large amount of heat transfer surface area in the reactor core to minimize heat fluxes and thermal stresses. Thus, if a moderator is used in a very small reactor, it is best mixed homogeneously with the U 235 so that all surface area in the core is useful for heat transfer.

Radiation-induced damage in fuel material can also be a limiting factor. At sufficiently high burnup, swelling, deformation, gas release and other metallurgical effects may endanger the safety of the core. It is thus necessary in some cases to dilute the fuel material which, in effect, increases the core volume.

The general range of reactor specific weights and power levels is indicated herein. The specific mass of the reactors decreases linearly with output up to about 5 mwt, at which point the unit becomes heat transfer rather than criticality limited and the specific weight remains essentially constant as output increases further. The reactors considered for the beachhead plant will probably have thermal outputs in the range of 1-2 mwt. This places them in the upper part of the criticality limited range.

For each fission, 13.5 mev of gamma radiation and 2.5 neutrons are produced. In a fast reactor, about 7 mev of the gamma radiation and 1 neutron escape from the core. At 3 mwt reactor operation about 10^{17} fission/second occur. This gives a dose of about 10^8 R/hr at the core surface.

For a 3 mwt fast reactor, the minimum operating shield weight is 40,000 pounds to give 1.0 R/hr at the face of the shield.

CRITICAL MASS OF BARE 100% DENSE HOMOGENEOUS SPHERES OF U^{235} MODERATED BY WATER AND BY Be AS A FUNCTION OF RADIUS AND MODERATOR TO U ATOM RATIO

c. Thermionic Reactor Size and Criticality Relationships

The active core of a thermionic power reactor is directly proportional to the electrical power output of the reactor.

$$P = Jv \left(\frac{S}{V} \right)_e V_{fe} V_c = Jv \left(\frac{S}{V} \right)_e V_{fe} V_c$$

J = current density at emitter surface

v = output voltage across the thermionic converter

$\left(\frac{S}{V} \right)_e$ = surface-to-volume ratio of the emitter

V_{fe} = volume fraction of the emitters in the reactor core

V_c = core volume

For any given thermionic cell performance and fuel element core configuration, the product $Jv \left(\frac{S}{V} \right)_e V_{fe}$ is constant and the reactor electrical power level P is a function only of the size of the core.

This electrical-power reactor size relationship is different from that of conventional nuclear power systems in which the reactor serves solely as the source of thermal power and electrical conversion takes place outside the reactor. Minimum core sizes are determined by consideration of maximum allowable fuel element surface heat fluxes, determined by the heat removal capability of the coolant and material temperature limitations. Heat fluxes of 300,000 to 400,000 Btu/hr-ft² are readily achievable with present technology.

To compare this present heat removal capability with that required in a thermionic power reactor, consider a typical cylindrical thermionic fuel element. The emitter is located at the center of the fuel element and extends axially along its length. Surrounding the emitters are the collectors, encapsulated within the fuel element cladding. Heat is generated in the emitter, transferred to the collector, and conducted to the coolant. The cladding heat fluxes are thus determined by the heat fluxes at the emitter surface. The cylindrical geometry of the fuel element results in a cladding surface larger than the emitter surface, causing actual cladding heat fluxes to be reduced from those at the emitter surface or an electrical power flux at the emitter surface (Jv) of about 10 watts/cm²; assuming a conversion efficiency of 15 percent, the heat transferred to the collec-

tor is 57 watts/cm² (180,000 Btu/hr-ft²). Since fluxes of this magnitude are well within present heat transfer capability, core sizes for thermionic reactors generally are concluded to be relatively independent of heat transfer considerations and are determined by the emitter surface areas required to produce the desired electrical power output.

The approximate minimum size of a thermionic reactor as a function of power can be determined by means of the equation previously stated.

The volume of the core is expressed as:

$$V_c = \frac{P}{J_v \left(\frac{S}{V}\right)_e V_{f_e}}$$

where typical values are:

$$J_v = 10 \text{ watts/cm}^2$$

$$\left(\frac{S}{V}\right)_e = 3 \text{ cm}^{-1}$$

$$V_{f_e} = 0.50$$

The resulting core volumes and spherical core diameters are:

Electrical power, P (megawatts)	0.1	1	10
Core volume, V_c (cm ³)	6.7×10^3	6.7×10^4	6.7×10^5
Core diameter (cm)	23	50	110

The very small core size at the 0.1 mw power level and the relatively small size at 1 mw are probably achievable only with a fast reactor, because of the high temperature anticipated (emitter temperatures of about 1500 to 2000° C) and the lack of a hydrogen-bearing moderator capable of operation at these temperatures. At the 10-mw power level, the core size is relatively large and a beryllium oxide moderator might be considered to reduce the fuel requirements.

It would probably be quite difficult to achieve criticality in reactor of the size calculated for the 0.1-mw power level. Average fuel densities would have to be high, implying large emitter volume fractions. To achieve these values, the required fuel element configuration would be a large-diameter, single emitter element extending the length of the core, instead of a series-connected array of emitters. While permitting higher fuel densities, this results in lower values of $\frac{S}{V}$, a probable overall

decrease in reactor power density, and an increase in size.

At power levels of 1 mw and above, criticality is no longer a major problem. High fuel densities are not required and the design of the fuel element can be oriented toward higher power densities. The longer fuel elements and the large emitter surface-to-volume ratios needed at these power levels indicate the desirability of using short series-connected thermionic cells within the fuel elements.

d. Field Expedient Shielding and Site Factors

Certain materials such as earth, gravel, rock, wood and water may be useful for shielding nuclear power plants. Since these materials are abundant at many plant sites, they provide a resource of shield material which does not have to be transported to the site.

However, if a plant is to be relocatable within a short time, it must be provided with shielding during transport, since activated portions of the plant as well as the shutdown core will continue to emit radiation. This means that some shielding must be an integral part of the plant package, which implies that at least part of the shielding during operation is not field expedient shielding.

In effect this means that the shutdown shield, which allows at least partial access to the plant within a short time after shutdown, must be an integral part of the plant. Field expedient shielding is limited to use as a supplementary shield during operation to provide adequate protection for personnel.

Water -- Water can be utilized for operational shielding since the effective attenuation of neutron radiation requires a hydrogenous material. Most stationary terrestrial applications envisage the use of water as part of the operating shield. For the beachhead plant some water will, in general, be available, but large quantities cannot be guaranteed and the plant must be capable of operating with limited quantities. At many sites, however, water is available and can be utilized for shielding. It is possible that the local fresh water or seawater will have to be treated in some way to improve its effectiveness as a shield material.

Snow or ice can also be utilized as shield materials in areas

Where they are prevalent. Care must be taken, however, to avoid unplanned melting.

Earth -- On many sites, earth is an ideal supplementary material. It is a fairly good material for shielding purposes and can be moved by equipment available in beachhead areas. Earth shielding can be utilized over and around an underground plant or piled around the perimeter of an aboveground plant. Special barriers can be erected at specific locations when particular work has to be done or when local shielding is required in a particular direction.

When shield walls of earth are used around a reactor without the provision of any top shielding, care must be taken to assure that the radiation scattered over the top of the barrier does not seriously increase the dose to personnel.

The effect of weather on the material should also be considered.

✓ Gravel, Sand, Rocks -- These materials can be utilized much as is earth. Since they have little or no water content, however, they are less effective as neutron shields though they may be somewhat more effective for gammas. Another use for these materials is as aggregate in concrete mixed on site.

Wood -- In some areas wood, as a hydrogenous material, can be used for shielding, possibly in combination with other materials.

Materials Evaluation -- In evaluating these materials for a specific site the following should be considered:

- (1) Shield effectiveness
 - (a) for gammas: material density
 - (b) for neutrons: hydrogen content
- (2) Availability of material and need for special equipment
- (3) Ease of handling material
- (4) Hazards associated with use of material e.g., leakage of water, fire hazard in wood
- (5) Time required to set up and dismantle the field shield

Terrain and Site Location -- The site of a plant greatly influences the operating field external shield design. Plants can be located in such areas as:

- (1) Underground - infinitely shielded in all directions and wholly or partially on top.
- (2) In a cave - i.e., infinitely shielded in three directions and wholly or partially on top.
- (3) Against a cliff - infinitely shielded on two sides.
- (4) Aboveground - on essentially flat terrain, essentially unshielded.

Environment -- The environment affects the plant in such ways as: choice of materials, choice of heat rejection system, necessity for external supplementary plant structure, etc. Environment variables include:

- (1) Temperature
- (2) Humidity
- (3) Precipitation
- (4) Probability of disastrous storms, earthquakes, landslides and other disturbances.

APPENDIX E
COMPACT POWER SYSTEMS - DATA SHEETS

1. Reactor Systems

a Mobile and Compact Reactors and Reactors of Advanced Technology

Army Reactors Program - Power for Remote Installations

<u>Designation</u>	<u>Location</u>	<u>Principal Nuclear Contractor</u>	<u>Type</u>	<u>Power</u>		<u>Start- up</u>	<u>Dis- mantled</u>
				<u>Plant Net KWE</u>	<u>Reactor KWT</u>		
SL-1	NRTS, Idaho	ANL	Boiling Water	200		1958	1961
SM-1	Fort Belvoir, Virginia	Alco	Pressurized Water	1855		1957	
PM-2A	Camp Century, Greenland	Alco	Pressurized Water	1500		1960	
GCRE-1	NRTS, Idaho	AGN	Gas Cooled, Light Water Moderated	No elec	2200	1960	
ML-1	NRTS, Idaho	AGN	Gas Cooled, Light Water Moderated	330	3300	1961	
PM-1	Sundance, Wyoming	Martin	Pressurized Water	1000	9370	1962	
PM-3A	McMurdo Sound, Antarctica	Martin	Pressurized Water	1500		1962	
SM-1A	Fort Greely, Alaska	Alco	Pressurized Water	1640		1962	
(PL-3) PM-3B	Byrd Station, Antarctica	Alco-A.C.	Pressurized Water	1000	9000		
PL-2		C.E.	Boiling Water	1000	7480		
MH-1A	Liberty Ship Walter H. Perry	Martin	Pressurized Water	10,000		1965	
MCR(MM-1)		Allison NDA	Liquid Metal Cooled	Up to 3 MWe			
Energy Depot System		G.M.					

Space Nuclear Auxiliary Power Program - Auxiliary Power for Space (SNAP)

Designation	Location	Principal Nuclear Contractor	Type	Power		Start-up	Dis-mantled
				Plant Net KWE	Reactor KWT		
SER	Santa Susana, California	AI	NaK Cooled		50	1959	1961
S2DS	Santa Susana, California	AI	NaK Cooled	3	50	1961	
S8ER	Santa Susana, California	AI	NaK Cooled		600	1962	
S8GPS	Santa Susana, California	AI	NaK Cooled	30-60	600	1963	
S8FPS	Santa Susana, California	AI	NaK Cooled	30-60	600	1964	
S2FS-I	Santa Susana, California	AI	NaK Cooled	3	50	1963	
S10FS-I	Santa Susana, California	AI	NaK Cooled	0.5	30	1962	
LCRE	NRTS, Idaho	P&W	Liquid Metal Cooled No elec	10,000		1964	
ORNL Intermediate Reactor Prototype		ORNL	K Cooled	100-300	1000	1965	
S50DR-1	NRTS, Idaho	P&W	Li Cooled	300-1000	8000-10,000	1965	

Aircraft Propulsion Reactor Program

ARE	Oak Ridge, Tennessee	ORNL	Molten Salt			1954	1954
HTRE-1	NRTS, Idaho	ANPD	Air Cooled	20,000		1956	1957
HTRE-2	NRTS, Idaho	ANPD	Air Cooled	14,000		1957	
HTRE-3	NRTS, Idaho	ANPD	Air Cooled	32,000		1958	

Missile Propulsion Reactor Program (Pluto)

Tory-IIA-1	NTS, Nevada	UCLRL	Air Cooled	160,000		1960	1961
Tory-IIC	NTS, Nevada	UCLRL	Air Cooled				

Rocket Propulsion Experiments (Rover)

<u>Designation</u>	<u>Location</u>	<u>Principal Nuclear Contractor</u>	<u>Type</u>	<u>Power</u>		<u>Start- up</u>	<u>Dis- mantled</u>
				<u>Plant</u>	<u>Reactor</u>		
				<u>Net</u>	<u>KWT</u>		
				<u>KWE</u>			
KIWI-A	NTS, Nevada	LASL	Open Cycle, Hydro- gen Gas Cooled			1959	1959
KIWI-A Prime	NTS, Nevada	LASL	Open Cycle, Hydro- gen Gas Cooled			1960	1960
KIWI-A3	NTS, Nevada	LASL	Open Cycle, Hydro- gen Gas Cooled			1960	1960
KIWI-B Series	NTS, Nevada	LASL	Open Cycle, Hydro- gen Gas Cooled			1961	1961
KIWI-B2A	NTS, Nevada	LASL	Open Cycle, Liquid Hydrogen			1962	

Naval Propulsion Experiments and Prototypes

SIG	West Milton, New York	GE	Sodium			1955	1957
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Advanced Reactor System Experiments

LAPRE-1	Los Alamos, New Mexico	LASL	Aqueous Homogeneous (Phosphoric Acid)	None	2000	1956	1957
LAPRE-2	Los Alamos, New Mexico	LASL	Aqueous Homogeneous (Phosphoric Acid)	"	1000	1959	1959
LAMPRE-1	Los Alamos, New Mexico	LASL	Fast Molten Plutonium Fueled, Sodium Cooled	"	1000	1961	
MSRE	Oak Ridge, Tennessee	ORNL	Single Region Graphite Moderated	"	10,000	1964	
UHTREX	Los Alamos, New Mexico	LASL	Helium Cooled	"	3000	1964	
EBOR	NRTS, Idaho	GDC	Gas Cooled, BeO Moderated	"	10,000	1964	
FRCTF	Los Alamos, New Mexico	LASL	Fast Molten Plutonium Fueled, Sodium Cooled	"	20,000	1964	

b. Specific Reactor Systems

(1) Gas-Cooled Reactor - ML-1

Designation ML-1 (Experiments: GCRE)
Company Aerojet General
Sponsor AEC - DOD (Army)
Location NRTS, Idaho
Use Relatively Mobile Power Source - Military Compact
Date Critical March, 1961
Status Initial power operation in first week of September
@ 100 kw. In 1963, power operation at 300 kw is
expected.
Power Thermal 3.3 mw
Electrical Net 0.33 mw
Power Conversion. . . . Closed Brayton cycle - 800° F in, 1200° F out
Air-cooled precooler
Clark Brothers axial compressor TC-Set
Coolant Nitrogen gas with some oxygen content
Efficiency 10%
Fuel Elements Tubular element in pressure tubes 61 fully en-
riched elements, fuel pin bundle

Core I

Lifetime 4000-5000 hours until shim removal
Fuel Pins Hastelloy with UO₂ pellets and UO₂-BeO
Elements Liner Hastelloy, outer structure: stainless steel
Control Semaphore control blades - 6 pair - radial
Insulation Thermoflex between fuel element and water moderation
Fuel Operating weight 53 KgU
Enrichment 93.15%
Material UO₂ & 70% UO₂-BeO
Shape Pellets
Center Temp (max/avg) 2,900/?
Cladding Thickness 30 mils
Material Hastelloy X
Shape Tubes
Surface Temp (max/avg) 1,700/1,350° F

Core II

Fuel Pins	All 19 pins contain diluted $\text{UO}_2\text{-BeO}$ finned on lower 2/3 length
Insulation	Refrasil tape
Cooldown time	22-23 seconds
Moderator	Weight 3,750 tons Material H_2O
Coolant	Loops 1 Material $\text{N}_2 + .5\% \text{O}_2$ Temp in/out 791/1,200° F Outlet pressure 291 psia Reactor operating press. 22 psi
Secondary	Cycle Direct cycle Coolant Nitrogen Flow 89,640 lb/hr
Plant Control	Bypass flow
Crew	10 set up; 7 operating
Transportability	C-130 airplane, truck, railway
Packaging	On modular skids --- 40 tons' total weight
Modules	1. Reactor 30,000 lb minimum 2. Power Conversion 30,000 lb for 3. Control 6,000 lb operation 4. Gas Storage 5. Cable Reel 5,000 lb 6. Spare Parts & Support 7. Gas drying for 8. Refueling support
Refueling	By single elements only
Shielding	Lead-lined borated water tank
Mounting	Shock mounted
Startup	60-70 kw gasoline generator required (45 kw design value)

General Design Characteristics

Design performance at 100°F

Gross electrical output	400 kw
Net electrical output	330 kw
Reactor thermal power	2.9 mw to gas; 3.3 mw total
Cycle efficiency	13.3%
Plant thermal efficiency	10%
Coolant flow	92,000 lb/hr

Dose rate at control cab @ 500 ft during full power operation	5 mr/hr (with expedient shielding as needed)
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Dose rate at 25 ft. 24 hr after shutdown (direction of transport vehicle driver with P-C skid in place)	15 mr/hr
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Overall plant dimensions	279 x 113 x 93 in. high
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Overall plant weight and dimensions .	Weight	Dimensions (in.)
Reactor package	30,000 lb	111 x 110 x 93 high (plus ion exchange column on end)
Power conversion package	30,000 lb	168 x 113 x 93 high
Control cab	6,500 lb	145 x 82 x 81 high
Auxiliary equipment	15,000 lb	- - - - -

Operating supplies (startup and 90-day operation)

Demineralized water	2900 gal
Nitrogen (with 0.5 vol% oxygen)	1800 scf
Oxygen	200 scf
Anhydrous boric acid (B ₂ O ₃)	1200 lb
Mixed bed ion exchange resin	900 lb max
Lubricating oil	180 gal
Filter elements	6
Plant startup time	12 hr

Auxiliary power requirements

Prestartup	30 kw max
Normal startup	45 kw max
Normal shutdown	45 kw max, 3 kw avg
Emergency shutdown	none
Reactor drying	45 kw max

Reactor Thermal Characteristics

Power density	700 kw/ft ³
Maximum heat flux	137,500 Btu/hr/ft ²
Average heat flux	80,500 Btu/hr/ft ²
Heat transfer surface	126.5 ft ²
Maximum: average heat flux ratio .	1.71
Maximum fuel center temperature (including hot spot factors) . . .	2240° F
Maximum moderator temperature . . .	200° F
Maximum surface temperature of fuel cladding (nominal, average)	1441° F
Maximum surface temperature of fuel cladding (including hot spot factors), reference	1750° F

Reactor Nuclear Characteristics

Average thermal neutron flux (fuel)	1.9×10^{12} neut/cm ² -sec
Average fast neutron flux (fuel). .	1.7×10^{13} neut/cm ² -sec
Maximum: average thermal flux ratio (fuel)	3.9
Hydrogen to U235 atom ratio	36
Core buckling	0.0053 cm ⁻²
Fermi age	92 cm ²
Square of thermal diffusion length, L ²	2.56 cm ²
Thermal utilization, f	0.82
Infinite multiplication factor, k .	1.70
Neutron lifetime	2.4×10^{-5}
k _{eff} , cold clean core; no shims or burnable poison	1.081

k_{eff} , cold clean core, with shims no burnable poison	1.040
k_{eff} , cold clean core, with shims and burnable poison	1.015
Core life, full power	3000 hr min; 10,000 hr design
Burnup (U235), average	3.6% in 10,000 hr
Prompt temperature coefficient, $\Delta k/k-^{\circ}C$	$+3 \times 10^{-6}$ @ $0^{\circ}C$ $+2 \times 10^{-6}$ $90^{\circ}C$

Reactor Vessel

Materials

Tube sheet	Stainless steel, Type 304, 2.94 in. thick
Pressure tubes	Stainless steel, Type 321
Source tube	Stainless steel, Type 321
Gas ducts, plenums and baffle . . .	Stainless steels, Types 304-L, 321 and 347
Outside diameter	30.960 in. max (exclusive of upper flanged connection)
Overall height	79.5 in.
Pressure tube length	24 in. between inside surfaces of tube sheets
Design pressure	345 psia
Design temperature	400° F
Wall thicknesses	Tubes 0.020 in. Plenum 2.12 in. min
Source tube	0.020 in. wall thickness; 0.500 in. OD

Reflector

Composition, top	2 in. H_2O ; 4.5 - 5.0 in. stainless steel: 1.5 in. W
Bottom	3-4 in. stainless steel; 3 in. W
Radial	1.8 in. Pb; 2 in. W; 180° segment 4 in. Pb; 180° segment
Total heat generation	6×10^5 Btu/hr
Maximum power density	360 Btu/hr-in. ³

Biological Shielding

Composition 3-1/2 to 4 in. lead plus 30 in.
of borated water (10 wt% boric acid)

Core (Excluding Reflector)

Diameter 22 in. equivalent
Height 22 in.
Number of fuel elements 61
Number of coolant passages 61
Number of coolant passes 1
Type of geometry of fuel elements Cluster of 19 pins (18 fueled)
Cold, clean critical mass, U235
no shims, no burnable poison 25 kg
U235 loading 49 kg
Enrichment, inner 6 pins 93% U235
outer 12 pins 31 vol% UO,
69 vol% BeO

Core composition

Materials	Volume %
UO ₂	4.3
BeO	3.3
Stainless steel	3.6
Hastelloy-X	7.0
H ₂ O	58.6
Insulation	7.0
Gas void	16.2
Total	100.0

Fuel Element

Dimensions 1.72 in. OD x 32 in.
Fuel material UO₂
Number of pins per element 19
Pin outside diameter 0.241 in.
Pin cladding material Hastelloy-X
Pin cladding wall thickness 0.030 in.
Pin spacer 0.040 in. OD Hastelloy wire

Heat transfer material	He
Pellet diameter	0.176 in. (nominal)
Type burnable poison	Cadmium
Reactivity worth of burnable poison	2.2% at startup
<u>Control Elements</u>	
Type	Tapered blades
Location	Moderator
Number: Shim blades	3 pairs (3 actuators)
Safety blades	2 pairs (2 actuators)
Regulating blades	1 pair (1 actuator)
Absorber material: Safety and shim blades	5 wt% cadmium- 15 wt% indium- 80 wt% silver
Dimensions (each blade)	4 x 10 x 0.25 to 0.62 in.
Regulating blades	Stainless steel
Dimensions (each blade)	4 x 10 x 0.25 to 0.62 in.
Cladding material	None
Reactivity worth of control elements:	
Safety and shim blades. . .	0.050 $\Delta k/k$
Regulating blades	0.005 $\Delta k/k$
Total	0.055 $\Delta k/k$
Actuating time for regulating blade:	
Drive	13.3 sec for full insertion or withdrawal
Scram	0.35 sec for full insertion from signal
Safety and shim actuator:	
Drive	4.0 min for full insertion or withdrawal
Scram	0.35 sec for full insertion from scram signal
<u>Moderator</u>	
Type	Water
Reactor inlet temperature	180° F

Reactor outlet temperature 190° F
 Pressure 32.5 psi max
 Flow rate 300 gpm
 Type of flow circulation Forced

Purity:

Total solids 1 ppm
 Resistivity 10^5 to 10^6 ohm-cm
 Total heat removal rate 1.5×10^6 Btu/hr

Reactor Working Fluid Flow

Working fluid 99.5 vol% N₂ + 0.5 vol% O₂
 Reactor inlet temperature 800° F nominal
 Reactor mixed-mean outlet temperature 1200° F max
 Average velocity in core 160 ft/sec
 Maximum velocity 180 ft/sec
 Inlet pressure 315 psia (max)
 Core ΔP 15 psi
 Reactor ΔP 22 psi

Power Cycle

Type Brayton cycle with regeneration
 Total volume of working fluid system 120 ft³
 Total system working fluid inventory
 full load at 100° F 52 lb
 Working fluid transit time 2.0 sec

Cycle characteristics:

Ambient temperature	<u>100° F</u>	<u>0° F</u>	<u>-65° F</u>
Net power, kw . . . , . . .	330	330	330
Reactor inlet, °F	791	597	597
Turbine inlet, °F	1200	990	990
Compressor inlet, °F	132	24	24
Compressor inlet, psia . . .	117	93	93
Compressor outlet, psia . . .	320	294	294
Reactor inlet, psia	313	288	288

Turbine-Compressor Set

	<u>Stratos T-C Set</u>	<u>Clark T-C Set</u>
Speed, rpm	18,338	22,000
Turbine stages	2	2
Turbine rotor material . .	Incoloy 901	Incoloy 901
Turbine blade material . .	Inco 713	N 155
Turbine stator blade material	Inconel	N 155 or 19-9DL
Expansion ratio	2.38	2.38
Compressor stages	2	11
Compressor material	Al 355 T71	403 stainless steel
Rotor shaft	SAE 4340	SAE 4340
Compressor ratio	2.72	2.72
Case material	304 stainless steel	304 stainless steel
Seals		
At journals	Buffered labyrinth	Buffered labyrinth
Interstage	Plain labyrinth	Plain labyrinth
Shaft	Buffered labyrinth	Double "L" ring seal oil buffered
Bearings		
Journal	Tilting pad	Plain babbit
Thrust	Kingsbury type (in low-press. area)	Kingsbury type (in low-press. area)
Support	Overhung turbine	Turbine and compressor supported between bearings
<u>Alternator</u>		
Output	<u>60-Cycle Operation</u>	<u>50-Cycle Operation</u>
Rating	500 kva 3 Ø	417 kva 3 Ø
Voltage	2400/4160 v	2000/3467 v
Rotor shaft speed	3600 rpm	3000 rpm
Case		
Diameter, without water jacket	39 in.	
Diameter, max. with water jacket, at starting motor	40.6 in.	

Length, without starting motor 26.8 in.
 Length, with starting motor 29.4 in.
 Weight, alternator only
 (including water jacket) 4700 lb
 Starting motor 360 lb
 Temperature, operating (hot spot) 275° F internal max

Recuperator

Length (including insulation) 81 in.
 Outside diameter (including insulation) . . 49.25 in.

Headers

High-pressure inlet 8 in.
 High-pressure outlet 8 in.
 Low-pressure inlet 20 in.
 Low-pressure outlet 14 in.

Effectiveness 79%

Pressure loss

High-pressure $\Delta p/p$ 2.24%
 Low-pressure $\Delta p/p$ 1.21%

Type Shell-and-tube regenerator
 Tubes 4 passes x 840 tubes
 Shell 1 pass
 Surface External fins
 Materials 300 series stainless steel

Pre-cooler, Moderator Cooler And Oil Cooler Assembly

Dimensions:

Length, overall 166-15/16 in.
 Precooler 122-5/16 in.
 Moderator cooler 32-1/8 in.
 Oil cooler 11-5/16 in.
 Width 113 in.
 Thickness, overall 32 in.
 Core 15 in.
 Fans and plenums 17 in.

Materials

Tubes and fins Series 1100 aluminum
Headers Series 2219 aluminum
Weight 6500 lb

Precooler:

Header, inlet One, 14 in.
Header, outlet One, 10 in.
Effectiveness 92.01%
Total Δ p/p 1.65%
Air flow 247,500 lb/hr
Type Fin fan air-to-gas exchanger
Tubes 1105 tubes, single pass
Surface Internal and external fins

Moderator cooler:

Headers, inlet and outlet 4 in.
Total Δ p 2.77 psi
Water temperature
In 190° F
Out 180° F
Air Flow 73,250 lb/hr
Type Fin fan air-to-water exchanger
Tubes 88 tubes per pass, three passes
Surface External fins

Oil cooler:

Connections, inlet and outlet 1 1/2 in.
Total Δ p 9.38 psi
Oil temperature
In 180° F
Out 150° F
Oil flow 18,900 lb/hr
Air flow 27,500 lb/hr
Type Fin fan air-to-oil exchanger
Tubes 45 tubes, 2 passes
Surface Internal and external fins

(2) Gas-Cooled Reactor - ML-1A

Designation	ML-1A - production model of ML-1
Status	DOD to purchase in 1963 procurement
Size	Same as ML-1
Optimization	System optimized better than ML-1 but not for size
Power	500 kw at - 65° F 100 kw at - 100° F
Cost	\$4 x 10 ⁶

(3) Gas-Cooled Reactor

Date Critical	CORE 1 2/23/60	CORE 2 9/61
Power - Thermal	2.2 mw	2.2 mw
Electric	0	0
Fuel		
Optimum Weight	46 kgU	46 kgU
Enrichment	93.15%	93.15%
Material	30 UO ₂ -316SS Cermet	UO ₂ & 70 UO ₂ -BeO
Shape	Curved plates	Pellets
Center Temp max/avg (°F)	2400/1775	2400/1775
Cladding		
Thick (mil)	6	30
Material	318 SS	Hastelloy-X
Shape	Curved plates	Tubes
Surface Temp max/avg (°F) .	1700/1320	1700/1320
Moderator		
Weight (lb)	200	200
Material	H ₂ O	H ₂ O
Coolant		
Loops	1	1
Material	N ₂ + 0.5O ₂	N ₂ + 0.5O ₂
Temp in/out (°F)	800/1200	800/1200
Outlet psia	192	192
Reactor op psi	18	18
Secondary		
Flow	N ₂ -H ₂ O-air dump 63,000 lb/hr N ₂	N ₂ -H ₂ O-air dump 63,000 lb/hr N ₂
Designer	AGN	AGN
Builder	AGC	AGC
Owner	USAEC	USAEC
Operator	AGN	AGN

(4) Gas-Cooled Beryllium Oxide Reactor - EBOR

Designation	EBOR - Experimental Beryllium Oxide Reactor (formerly BORE, MGCR)
Company	General Atomics (Westinghouse for Power Conversion)
Sponsor	AEC - Maritime Commission
Use	Formerly ship propulsion - now consider small plant
Experiment	EBOR - Idaho - to test BeO as material - no power conversion
Construction	Begin Winter 1962 Completion for Spring 1964
Power	10 mwt
Coolant	Helium for first experiment - later other coolants; 1300°F gas, 1075 psi
Steam Temperature	1000°F
Fuel	Uranium mixed with BeO
Burnup Experienced	Up to 100,000 mwd/ton 35 atom %
Cladding	Hastelloy at present, ceramic in future
Expected Advances	With high gas temperature clad element to 1900°F surface temperature; unclad element to 3600°F center temperature
Power Conversion	Direct cycle - gas turbine - helium (none in EBOR) Indirect cycle - steam turbine
Costs	Beryllium oxide price now \$28/lb formerly \$50/lb
Power Prototype Schedule. .	Not possible till 1966 or 1967

Reactor Core and Reflector

Power rating	10 mw(t)
Average neutron flux:	
Fast	3.1×10^{13} (>0.2 mev)
Thermal	4.8×10^{12} (<0.68 ev)
Power peaking factors:	
Axial	2.10
Radial	1.120
Excess reactivity (k) to compensate for	
Temperature changes	4%
Xe and Sm at rated power	2%
Burnup	3%
Maximum excess reactivity with dry core	9%
Fuel:	
UO ₂ in BeO	22 vol%
Uranium enrichment	70%
Pellet diameter	0.33 in.
Pellet length	0.43 in.
Pellets per fuel pin	178
Cladding	0.020 in. Hastelloy-X tubing, 0.335 in. ID by 80 in. long
Core dimensions (active core)	23.43 in. square by 76 in. long
Reflector	7 in. of BeO
Number of fuel elements	36
Lattice	Square, 3.64-in. pitch
Core loading	100 kg U235
Average power density in core	14.4 kw/liter = 1.40×10^6 Btu/(hr)(ft ³)
Total BeO in active core	2350 lb
Total BeO in reflector	5279 lb
Heat flux at fuel-element surface:	
Average	5.8 cal/cm ² -sec = 77,100 Btu/(hr)(ft ²)

Maximum	15.0 cal/cm ² -sec = 200,000 Btu/(hr)(ft ²)
Film-temperature drop, design	43° C = 109° F
Fuel-element temperatures, design maximum:	
Fuel, nominal value without hot-channel factor	910° C = 1670° F
Cladding, nominal value without hot-channel factor	730° C = 1350° F
Coolant flow area:	
Each fuel element	7.5 cm ² = 1.165 in. ²
Total	270 cm ² = 41.94 in. ²
Channel velocity of coolant:	
Minimum	50 m/sec = 166 ft/sec
Maximum	75 m/sec = 246 ft/sec
Heat-transfer coefficient	0.134 cal/(cm ²)(sec)(°C) = 989 Btu/(hr)(ft ²)(°F)
Coolant mass flow rate	6.3 kg/sec = 13.88 lb/sec
Coolant temperature:	
Inlet	400° C = 750° F
Outlet	700° C = 1300° F
Hot-channel factors: *	
Temperature rise of coolant . .	1.152
Temperature drop of film . . .	1.20
Overall temperature drop for cladding and fuel	1.594
<u>Control</u>	
Shim safety rods (cruciform):	
Number	4
Active length	76 in.
Worth of shim safety rods . . .	25% δ k
Shim safety rod speed:	
Insertion or withdrawal . . .	0.15 cm/sec
Insertion only	0.30 cm/sec

* Factors are for the lowest-powered channel, which gives the maximum cladding temperature.

Regulating rods (shim safety rods used
for regulation):

Worth of regulating rods 25% δk

Regulating-rod speed:

Insertion or withdrawal 0.15 cm/sec

Insertion only 0.30 cm/sec

Maximum reactivity insertion rate $1.5 \times 10^{-4} \delta k/k/sec$

Scram time and type of mechanism:

50% insertion 0.7 sec

100% insertion 2.0 sec

High-speed, pneumatic, motor-driven
lead screw, positive scram blade
insertion

Temperature coefficients, $\delta k/k/^\circ C$:

Cold:

Prompt -0.54×10^{-4}

Delayed $+0.14 \times 10^{-4}$

Total -0.40×10^{-4}

Hot:

Prompt -0.25×10^{-4}

Delayed -0.02×10^{-4}

Total -0.27×10^{-4}

Hot equilibrium poison:

Prompt -0.25×10^{-4}

Delayed $+0.10 \times 10^{-4}$

Total -0.15×10^{-4}

Reactor Vessel

Configuration Cylindrical shell, hemispherical
heads

Material SA-387B

Shell wall thickness 5-1/4 in.

Vessel pressure:

Working pressure 1120 psia

Design pressure 1250 psia

Test pressure 1875 psia

Overall reactor dimensions:

Overall height of reactor, including
control rod drives 34 ft

Outside diameter of pressure vessel
over primary flange 13 ft

Containment type and material No containment vessel; reactor
is located in a vault that
has concrete walls and bottom
and that is covered with
removable concrete slabs.

(5) Military Compact Reactor - MCR

Sponsor Army Reactors - AEC

Organizations Allison Division, General Motors
NDA (United Nuclear)

System Compact, lightweight, mobile nuclear reactor
power plant, capable of operation while in
motion. To be applied to use as a mobile
power source, with minimum vulnerability.

Designation MCR (MM-1)

Program Preliminary studies completed 1961
Development work started 1962
Pilot plant construction and operation
Prototype possible in 1966

Reactor High temperature
Liquid-metal cooled

(6) SNAP-2 (3-Kwe Size)

Program	Space Reactor System for Air Force - 1955 AEC chose Atomics International in 1956 Develop 3-kwe unit Mercury vapor turbogenerator conversion equipment is being developed by TRW under contract to Atomics International.
Materials	Metal hydride technology
Power	Thermal - 50 kw Electric - 3 kw
Converter	Hg Rankine
Power Source	ZrH reactor-thermal reactor
Maximum Operating Temperature	1300° F
Weight	750 lb (250,220-lb reactor) - unshielded
Approximate Life	12 months
Prime Contractor	Atomics International (reactor) Thompson, Ramo & Wooldridge (converter)
Application	Power source for satellite and other space craft
Core Volume	0.3 ft ³
Core Configuration	Hexagonal arrangement of 61 cylinder elements
Coolant	Liquid sodium, NaK
Temperature	1000° F in, 1200° F out
Fuel Elements	ZrH - U 235 10" long - 1" diameter, 37 rods (1.25" OD)
Secondary System	Mercury boiler and turbogenerator (TRW) tested superheated Hg single shaft power conversion
Overall Efficiency	6%
Core Power Density	165-170 kw/ft ³
Primary Pump	Rotating permanent magnet pump
Reflector	Beryllium - outside vessel
Control	Reflector thickness - 2 Be control drums 2 Be safety plates
Fuel Cladding	Stainless steel (SER) - Hastelloy N (SNAP-2)
Core Vessel	9" diameter 10 mils - 14 mils 1.25"

Turbine	2 stage, axial-flow, impulse Hg turbine														
Alternator	P.M. - sealed station - 3.5 kw at 110-2000 cps														
Hg Pump	Centrifugal pump														
Available	1964														
Power Conversion	NaK pump, turbine, alternator, Hg pump on single shaft - 40,000 rpm														
Boiler	Concentric tube counterflow - once through NaK in annulus, Hg in inner tube helical configuration														
Radiator Condenser	Hg condensed 600° F and 6 psia														
Radiator Area	100 ft ²														
Weights (approx lb)	<table border="0"> <tbody> <tr> <td>Reactor</td> <td>- 200-250</td> </tr> <tr> <td>Boiler</td> <td>- 100</td> </tr> <tr> <td>Combined rotating unit</td> <td>- 50</td> </tr> <tr> <td>Radiator and liquid</td> <td>- 150</td> </tr> <tr> <td>Controls</td> <td>- 20</td> </tr> <tr> <td>Structure</td> <td>- 50</td> </tr> <tr> <td>Piping</td> <td>- 30</td> </tr> </tbody> </table>	Reactor	- 200-250	Boiler	- 100	Combined rotating unit	- 50	Radiator and liquid	- 150	Controls	- 20	Structure	- 50	Piping	- 30
Reactor	- 200-250														
Boiler	- 100														
Combined rotating unit	- 50														
Radiator and liquid	- 150														
Controls	- 20														
Structure	- 50														
Piping	- 30														
Unshielded Total System . . .	600-700 lb														
Shadow Shield for Transistors	300 lb (depends on vehicle design)														
Shielded Total System	900-1050 lb														
Specific Weight System	300-350 lb/kw (shadow shielded for transistors) 200-250 lb/kw (unshielded)														
Design Environment	10 g steady-state vessel 60 g shock 15 g vibration														
Reactor Size	16 x 14 dia (inches)														
U in ZrH _x -wt %	9.85 (x = 1.85)														
U 235	4.3 kg														
N _H , H atoms/cm ³	6.5 x 10 ²²														
Average Heat Flux	17,000 Btu/hr/ft ²														
Neutron Flux	3.1 x 10 ¹¹ n/cm ² sec														
Reflector	Axial - 1.5" BeO Radial - 2.3" Be														
Hg Boiling	900° F														
Hg Condensing	600° F														
Turbine Inlet Temp	1150° F														

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 flow rate	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 flow rate	44 lb/min
Hg system flow rate	20 lb/min

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	
<u>Heat Rejection</u>		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	

(7) SNAP-2 (11-kwe Size) (AI-7950)

Purpose Power source for cylindrical space station
Power 165 kwt, 11 kwe
System SNAP-2 reactor at higher power
Three SNAP-2 system type rotating power conversion units and radiators

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

Weight Summary for Cylindrical Space Station (lb)

Reactor System	320	
Secondary NaK Systems (3)	150	
Power Conversion Systems (3)	1080	
Radiator-Condenser (3)	<u>660</u>	
Total unshielded power system	2210	210 lb/kwe
Uranium	950	
Zirconium hydride	1890	
Lithium hydride	460	
Stainless-steel containers	<u>500</u>	
Conical Shadow Shield, Total	3800	
Shutdown Shield	2000	
Support Structures	<u>1300</u>	
Total System Weight	9310	845 lb/kwe

(8) SNAP-8

Program Based on SNAP-2 metal hydride technology NASA is developing power conversion equipment through AGN. NASA also is responsible for overall system integration.

Schedule includes demonstration of 90-day system life in 1964, operation for 10,000 hours of complete system in 1965.

30-60-kw turboelectric unit

To develop 30 lb/kw in 90-150-kw power

Power Thermal - 300-600 kw
Electric - 30-60 kw

Converter Hg Rankine

Power Source ZrH reactor

Maximum Operating Temperature 1450° F

Weight 1400-1500 lb (unshielded system)

Approximate Life 12 months

Prime Contractor Atomics International (reactor)
Aerojet (converter)

Application Satellite and electric propulsion power source

Team NASA - System sponsor
NASA-JPL - Mission requirements
NASA-Lewis - Technical direction system development
AEC - Reactor development adviser to NASA
A. I. - Reactor contractor
AGN - Contractor to entire reactor system including including conversion machinery

Marshall Flight Center - Flight test

Overall Efficiency 10-11.5%

Available 1965 for 30 kwe - 1966 for 60 kwe

Reactor Size 19" x 15" diameter

Core Volume 0.48 ft³

Power Density Average	1.25 mwt/ft ³
Weight, Reactor . . .	400 lb (unshielded)
Size, Reactor	"5-gallon can"
Specific Weight	45-60 lb/kwe (unshielded)
Fuel-Moderator Rods . .	211 rods, 0.53 x 14"
U in ZrH _x , Wt%	9.85 (x = 1.7)
U 235:	7.0 kg
N _H , H atoms/cm ³ :	6.0 x 10 ²²
Cladding	Hastelloy N, 1 mils, 0.56" OD
Heat Flux Average . . .	55,000 Btu/hr/ft ²
Neutron Flux	2.3 x 10 ¹² n/cm ² /sec
Reflector	Axial - None Radial - 3" Be
Coolant	NaK (Na78K) 1100° F in, 1300° F out
Turbine Inlet	
Temperature.	1250° F
Hg Boiling	1070° F
Hg Condensing.	700° F
Radiator, Average	
Temperature	700° F
Radiator Area	400-800 ft ² (520 ft ²)
Endurance Expected . .	10,000 hr
Control	Be reflector drums, 2Be safety plates

(9) SNAP-10

Contractor	Atomics International
Application	Space Auxiliary Power Unit
Reactor	UZrH (cf SNAP-2)
Power	Thermal - 15 kw Electric- 300 watts
Cooling	Conduction
Converter	Thermoelectric, mounted on reactor surface
Reactor Diameter	12"
Reactor Length	18"
Thermocouple Hot Junction Temperature	100° F
System Weight	330-400 lb
Specific Weight	1100-1333 lb
Status	Study completed 1959 Canceled (replaced by SNAP-10A program)

(10) SNAP-10A

Prime Contractor	Atomics International
Application	Satellite Power Source
Availability	1963
Converter	Lead-telluride thermoelectric - heated by reactor coolant. Two concepts: a) TE converter directly coupled to radiator b) Secondary heat transfer loop to radiator from TE cold junction
Output	0.5 kwe
Thermal kwe	25-30
Efficiency	2.0%
Power Source	U-ZrH reactor (SNAP-2)
Available Year	1963
Minimum Life	1-3 years
Reactor size	16" by 14" dia
Core Volume	0.3 ft ³
Power Density	0.099 mwt/ft ³
Reactor wt	250-325 lb
Converter wt	250 lb
System wt (unshielded)	525 lb
Specific wt	1050-1200 lb/kw
Specific wt (shielded for Transistor Payload)	1450 lb/kw
Fuel Moderator	UZrH
Rods number	37
U 235	4.3 kg
N _H , H atoms/cm ³	6.5 by 10 ²²
Max temp	1050° F
OD	1.25 in.
Heat flux	10,200 Btu/hr/ft ²
Neutron flux, n/cm ² /sec	1.8 x 10 ¹¹
Radial Reflector, in.	2.3 Be

Be control drums	2 (startup)
Be safety plates	2
Reactor coolant	Na78K
Temp Out	900° F
Temp In	800° F
Temp of TE Hot Junction	859° F
Temp of TE Cold Junction	590° F
Radiator Area	62.5 ft ²
Radiator	580-600° F (avg)

Atomics International Convection Cooled Thermoelectric System (Similar to SNAP 10A)

Power Output	300 watts
Reactor Power	12 kw
Hot Junction Temperature	900° F
Converter Efficiency	10% of Carnot
Radiator Area	35 ft ²

Weight

Reactor	250 lb
Shield for transistors	250 lb
Converter-radiator	100 lb
Miscellaneous (pump, structure, etc:)	100 lb
Total :	700 lb

Specific weight: 2333 lb/kwe

Power Output	1000 watts
Reactor Power	35 kw
Hot Junction Temperature	900° F
Converter Efficiency	10% of Carnot
Radiator Area	100 ft ²

Weight

Reactor	250 lb
Shield for transistors	300 lb
Converter-radiator	300 lb

Miscellaneous (pump,
structure, etc.) 150 lb
Total: 1000 lb
Specific weight: 1000 lb/kwe

(11) Oak Ridge Intermediate Reactor

Organization Oak Ridge National Laboratories
Purpose Intermediate Size Space Power System
Power Output 1000-kw thermal
100-300 kw electric
Reactor Cycle Direct boiling Rankine
Coolant Potassium
Working Fluid Potassium
Operating Temperature . 1500° F
Conversion Unit Potassium turbine
Structural Materials . . Probably stainless steel
Fuel Probable UO₂ with stainless-steel cladding
Specific weight About 25 lb/kw including shield for space
instrumentation
Experimental Reactor
Operation Expected 1965
Schedule For development beyond SNAP-2, 8, but utilizing
a more conventional technology than SNAP-50/SPUR.

(12) Lunar Reactor Plant - Boiling Mercury System - ANL

Designer Argonne National Laboratory (Report ANL-6261)
Capacity 1 mwe
Specific Power 9.8 kg/kw
Total Weight 9800 kg
Overall Thermal
Efficiency 12.2%
Dimensions of Radiator
No. 2 folded Overall Length : 485 cm
Diameter : 214 cm
Reactor Fast reactor cooled by boiling mercury
Core Diameter 41 cm
Core Length 41 cm
Critical Mass 492 kg
Enrichment 35.6%
Power Density 150 kw/l (core vol)
Heat Transfer Surface . 1.495 x 10⁵ cm²
% Voids at Full Power . 73%
Speed of Mercury
Entering Core 50.5 cm/sec
Reactivity Worth of
Reflector 17%
Materials Surfaces in contact with Hg: Steel with 5% Cr,
1/2% Mo
Conversion System . . . Direct, 2-phase mercury Rankine cycle
Generator Type - AC brushless
Continuous rating - 1000 kva
Power factor - 0.75, lagging to 1.00
Overload 2 min. - 1500 kva
Overload 2 sec. - 2000 kva
Speed - 5700-6300 rpm
Overspeed - 7600 rpm

POWER PLANT WEIGHTS (in Kg)

	1	2	3
	<u>All Steel, Except Mg Base and Supports</u>	<u>Steel Re- actor Ti Radiator Mg Base</u>	<u>Combination Steel and Ti Reactor - Mg - Clad Radiator Mg Base</u>
<u>Reactor</u>			
Pressure vessel, ducts, grids, separation plates, and miscellaneous hardware	1062.2	1062.2	920.0
Core	686.5	686.5	686.5
Reflector controls and operating mechanisms . .	637.8	637.8	574.0
Turbine and pumps	318.0	318.0	318.0
Generator	454.0	454.0	409.0
Compressor	<u>27.0</u>	<u>27.0</u>	<u>24.0</u>
Reactor total	3185.5	3185.5	2931.5
Base and radiator scaffold	133.0	133.0	133.0
<u>Radiator</u>			
Radiator tubes	3450.8	1929.0	1052.0
Manifolds	807.0	451.0	451.0
Miscellaneous hardware .	429.7	429.7	429.7
Valves and hoses	<u>477.0</u>	<u>477.0</u>	<u>477.0</u>
Radiator total	5164.5	3286.7	2409.7
Mercury inventory	<u>1315.0</u>	<u>1315.0</u>	<u>1315.0</u>
Total plant weight . . .	9798.0	7920.2	6789.2
Weight, kg/kwt	1.225	.990	.849
Weight, kg/kwe	9.798	7.920	6.789

(13) SNAP-50/SPUR

Sponsors AEC, Air Force, NASA
Companies Pratt & Whitney, ORNL, BNL, ANL
Program Longer range than SNAP-2, more advanced technology,
for developing reactors for space of up to 1000 kw
and beyond.
Power Range 300-1000 kwe and greater
Development Time 8-10 years
Flight Test Capability . 1970 or before
Operating Temperature . 2000° F and greater
Specific Weight 10 lb/kwe
Conversion System . . . Rankine turboelectric
Operating Time 1 up to 5-10 years

Reference Design Criteria

Power Thermal : 8-10 mw
Electric : 1 mw
Maximum Operating
Temperature 1500-2000° F
Weight 10 lb/kwe (stated as 10-20 lb/kwe 1/27/62)
Approximate Life 10,000 hr unattended
Prime Contractor Pratt & Whitney

Current System Concept

Cycle Two loop indirect
Reactor Coolant Lithium
Turbing Working Fluid . Boiling potassium (probably)
Fuel UO₂BeO, UO₂, UC being considered
System Materials Refractory metals, columbium, etc.
Experimental Reactor . . LCRE (reactor with primary loop only)
Criticality Expected . . 1964
Power Operation 1965
Development Cost \$250,000,000
Current Funding FY 1963 \$25,000,000 (mostly LCRE)
FY 1964 expect increase

(14) SPUR - Space Power Unit Reactor (Garrett & Aerojet)

Program Reactor powered, turbogenerator type APU for
electric propulsion and as satellite power source

Sponsor United States Air Force

System Manager and
Power Conversion
Equipment Garrett-AiResearch

Reactor and Reactor
Loop Components . . . Aerojet General Nucleonics

Generator and Electrical
Components Westinghouse

Power Thermal
Electric : 150-300 kw (ground test unit)

Operating Temperature . 2000° F

Coolant Sodium or lithium - primary
Potassium - secondary

Specific Weight 7.9 lb/kw unshielded

Specific Weight 10 lb/kw shielded

Weight 3000 lb- shielded

Packaged Diameter . . . 10 feet

Packaged Length 36 feet

Operating Life 1 year

Radiator Area 3 sq ft/kwe

(15) STAR - Space Thermionic Auxiliary Reactor (General Electric)

Program A thermionic, reactor-powered APU being developed
by General Electric with its own funds.

	<u>Radiation-Cooled STAR-R (final)</u>	<u>Convection-Cooled STAR-C (final)</u>
Power . . . Thermal Electric	13-70 kw	LCT - 5500 kw 1000 kw
Temperature:		
Fuel	2200-2430° C	
Cathode	2000-2230° C	
Reflector Suf..	1000-1550° C	
Specific Weight	100-19 lb/kw	5 lb/kw
Length	26 inches	
Diameter	13 inches	
Cooling	Radiation	Conduction (liquid metal)
Weight (Unshielded)	1300-1400 lb (w/ no power conducting equipment)	5000 lb (w/ no power conducting equipment)
Length	12 inches	
Output	4.3 kwe	
Specific Weight	140 lb/kwe	

(16) In-Core Thermionic Reactor - Martin Study

Use Terrestrial - lightweight, small plant for use
in remote areas

Sponsor Army Reactors Branch, AEC, AECNYO

Power 1000 kwe

Power Conversion Low-temperature thermionic emitter on fuel

Emitter 1400° C emitter temperature - up to 1500° C

Coolant Water - forced circulation boiling

Moderator Solid - zirconium hydride

Fuel UO₂ pellet coated with molybdenum

Length of Fuel
Element - Diode . . . 44 inches

No. of Elements per
Bundle 35

No. of Bundles per
Core 37

Core Diameter 40 inches

Conversion Efficiency
Expected 10% overall

Study 4 months

Report Release November 15, 1962

Subjects Technical Feasibility
Economic Feasibility
Materials
Core Design
Plant Operational Requirements

Standards Fuel core - 15-month life at full power
Rapid, single reassembly
C-130 aircraft delivery

Shipment 4 or less modules at less than 30,000 lb each

Cost to Build Reactor . \$2,500,000

(17) In-Core Thermionic System - B&W

Organization:	B&W and RCA for Bureau of Ships (Report BAW-1261)
System	Thermionic Converters in Natural Circulation Boiling Water Reactor
Power	1 mwe
<u>Characteristics of Reference Core</u>	
Core Diameter	44.08 in.
Core Length	42.00 in.
Metal-to-Water Ratio	0.70
Power Density	9.5 kw/liter
Pressure Tube Heat Transfer Area	745 ft ²
Average Pressure Tube Heat Flux (9 mwt 1 mwe)	41250 Btu/hr ft ²
Cathode Surface Area	441 ft ²
<u>Fuel Loading</u>	
Fuel	UO ₂
Enriched fraction	0.124
UO ₂	991 kg
U 235	107 kg
Fuel Burnup (mwd per ton of ceramic per full power year)	3340 mwd/ton
Calculated k _{eff} at 400° F	1.10
<u>Coolant</u>	
Temperature	400° F
Pressure	250 psia
<u>Fuel Elements</u>	
Total	826
Element composition	8 converters stacked vertically within a Zircaloy pressure tube
<u>Fuel Cell</u>	
Total	6608

Composition

<u>Region</u>	<u>Materials</u>	<u>OD, in.</u>	<u>Area Friction</u>
Fuel	UO ₂	0.05000	0.10631
Cathode	Mo	0.72000	0.11414
Anode, insulation and gaps	Nb, Al ₂ O ₃	0.92000	0.134
Pressure tube	Zr	0.09400	0.05182
Coolant	H ₂ O	1.53347	0.58825

Converter

Total	6608
Length	5.25 in.
Cathode fuel length	4.25 in.
Cathode surface area	62.006 cm ²
Assumed electrical characteristics	
Output voltage	0.5 v
Current density	5 amp/cm ²
Power density	2.5 watt/cm ²

Radial Reflector

Material	Beryllium
Thickness	4.0 in.
Volume fraction	1.0

(18) In-Core Thermionic Reactor Study - AI

STUDY: Atomics International for AEC, for several years

Specific Weights	4-5 lb/kwe
Power	> 300 kwe
Conversion Efficiency Possible	5 to 20% (17.5%)
Ready for Flight Test	10-13 years
Power Output	15-20 watt/cc
Temperature of Operation	3000° F or higher
Coolant	Liquid metal
Heat Rejection Temperature	1475° F

(19) In-Core Thermionic Reactor Study - GA

Study	General Atomics
Power	300 kwe
TI Cell Efficiency	14%
TI Cell Power Density	11 watts/cm ²

Reactor Core with TI

Diameter	13 inches and Reflector 2 inches
Height	11 inches
Fuel element cells	546
Weight	582 lb
System Efficiency - Overall	12%
Coolant	Lithium
Reactor Outlet Temperature	2000° F
Reactor Inlet Temperature	1800° F
Radiator Area	190 ft ² @ 2.3 lb/ft ² , 2 sides
Radiator Weight	220 lb
Specific Weight, System	4.2 lb/kwe
Radiator Temperature	1900° F

Variations

Radiator Temperature	1700° F
Cell Output	8 watts/cm ²
Efficiency	6%
Specific Weight of System	7-8 lb/kwe
Radiation Temp	1500° F
Overall Efficiency	12%
Specific Weight of System	6-7 lb/kwe

(20) In-Core Thermionic Reactor Study - AGN

Organization Aerojet General - Nuclonics
Power Output 300-2000 kwe
Cathode Current Density $>10 \text{ amp/cm}^2$
Cathode Power Density 30 watt/cm^2
Reactor UC ZrC reactor (LASL)
Specific Weight Range 12.5 to 8.5 lb/kw
(competitive with Dynamic System)
Temperature 3300° F

(21) Advanced Reactor Program - Beyond SNAP 50 and ORNL Inter-
mediate

Sponsor AEC, Air Force, NASA
Organizations Pratt & Whitney, ORNL, LASL, GA, GE
Systems under Study . . 1. SNAP-50 reactor with thermionic converters
on heat rejection radiation
2. SNAP-50 type reactor with metal vapor
magnetohydrodynamic conversion system
3. Direct-boiling SNAP-50
4. Gas-cooled reactors - Brayton cycle
5. Thermionic direct conversion technology -
GA and GE research on thermionic fuel
element - LASL support

2. Conversion Systems

(1) SNAP-1 Turbine

Turbine Three-stage assembly
Fluid Mercury
Inlet Pressure 210 psia
Inlet Temperature 1300 ° F
Rotational Speed 40,000 rpm
Exhaust Pressure 2.06 psia
Weight Flow Rate 0.311 lb/sec
Power 0.875 hp
Design Efficiency 40%
Test Efficiency (Avg) . . 42%
Bearings Mercury hydrosphere

(2) SNAP-2 Turbine

Design-Point Operating Conditions

<u>Requirement</u>	<u>Original Turbine Requirement</u>	<u>Increase in Output to achieve 3 KW Electrical Power</u>	<u>Later Version</u>
Rotational speed, rpm	40,000	40,000	40,000
Inlet total pressure, psia	100	105	115
Inlet total temperature, °F	1,150	1,150	1,150
Exit static pressure, psia	6.2	6.5	7.1
Required power output, kw	4.86	5.16	5.67

(3) 3-KW Turbine (AiResearch)

Organization AiResearch Corporation.
Turbine Tested 1957-1958
Working Fluid Mercury
Flow Rate 16 lb/min
Power Level 3 kw
Duration 400 hr

	<u>Inlet Press.</u>	<u>Exhaust Press.</u>	<u>Inlet Temp</u>	<u>Dia</u>	<u>RPM</u>
First Stage	125 psia	14.2 psia	1100° F	2.86 in.	36,000
Second Stage	14.5 psia	6.3 psia	1100° F	2.86 in.	36,000

(4) 15-KW Vapor Turbine Power Unit - with Solar Energy

Source (Sunstrand)

Organization Sunstrand Turbo
Output 15 kwe
Turbine 3 stage
Working Fluid Rubidium
Temperature of Boiling . 1200° F @ 10 psia
Temperature of Superheat 1750° F
Temperature of Subcooled
Fluid 675° F
Bearings Journal bearings - rubidium lubricated

(5) Stirling Engine Generator

Sponsor Air Force Wright Development Division
Organization Allison Division, General Motors
Rated Power 3 kw (400 cycle, 220 volts, 3 phase, ac)
capable of 5 kw output
Rated Efficiency 30.5% (brake thermal) minimum
Rated Generator
Efficiency 80%
Speed 3000 rpm (engine and generator)
Working Medium dry helium
Heater Temperature . . . 1250° F (NaK in)
Coolant Temperature. . . 150° F (water in)
Mean Cycle Pressure . . 1500 psi
Mean Crankcase Pressure. 1430 psi
Weight 285 lb (engine and generators)
Estimated weight of prototype engine.
Considerable reduction in weight is anticipated
for the fully developed flight configuration.
Generator Power 120/208 volts, 400 cps. 3 phase, 2 generators
of 2.5 kva each.
Frequency Control . . . ± 1%; voltage control -- ± 1%
Generator Power Output . 5.5-kva output capability; constant load --
5.0 kva; waste power load for 500 watts for
speed and frequency regulation.
Program Design, fabrication and performance testing
of advanced Stirling cycle engine suitable
for application to a flight prototype system.

Estimated Engine-Generator Assembly

Weights and Dimensions

(Allison Data)

Dimensions (Inches)

<u>Power</u>	<u>Weight lb</u>	<u>Height</u>	<u>Length</u>	<u>Width</u>	<u>Specific weight lb/KWE</u>
1.0	83	14.5	18.8	11.0	83
4.0	186	20.0	23.0	13.5	46
10.0	330	25.0	27.4	16.2	33
15.0	420	27.3	30.0	17.8	28
20.0	500	29.5	32.1	19.1	25
30.0	630	33.0	35.8	21.2	21

(6) ML-I Gas Turbine-Compressor System

	<u>Stratos T-C Set</u>	<u>Clark T-C Set</u>
Speed, rpm	18,338	22,000
Turbine stages	2	2
Turbine rotor material . .	Incoloy 901	Incoloy 901
Turbine blade material . .	Inco 713	N 155
Turbine stator blade material	Inconel	N 155 or 19-9DL
Expansion ratio	2.38	2.38
Compressor stages	2	11
Compressor material . . .	Al 355 T71	403 stainless steel
Rotor shaft	SAE 4340	SAE 4340
Compressor ratio	2.72	2.72
Case material	304 stainless steel	304 stainless steel
Seals		
At journals	Buffered labyrinth	Buffered labyrinth
Interstage	Plain labyrinth	Plain labyrinth
Shaft	Buffered labyrinth	Double "L" ring seal oil buffered
Bearings		
Journal	Tilting pad	Plain babbit
Thrust	Kingsbury type (in low-press. area)	Kingsbury rype (in low-press. area)
Support	Overhung turbine	Turbine and compressor supported between bearings
<u>Alternator</u>		
Output	<u>60-Cycle Operation</u>	<u>50-Cycle Operation</u>
Rating	500 kva 3 Ø	417 kva 3 Ø
Voltage	2400/4160 v	2000/Ø467 v
Rotor shaft speed	3600 rpm	3000 rpm
Case		
Diameter, without water jacket . . .	39 in.	
Diameter, max. with water jacket at starting motor	40.6 in.	

Length, without starting motor . . . 26.8 in.
 Length, with starting motor . . . 29.4 in.
 Weight, alternator only (including water
 jacket) . . . 4700 lb
 Starting motor . . . 360 lb
 Temperature, operating (hot spot) . . . 275° F internal max

(7) Fluid Engine (Cleveland Pneumatic)

Organization . . . Cleveland Pneumatic
 Cycle . . . Brayton
 Working Fluid . . . CO₂ - supercritical
 Engine Speed . . . 1000 rpm
 Temperatures
 Heater in . . . 600° F
 Heater out . . . 500° F
 Cooler in . . . 35° F
 Total System Weight . 4 lb/hp at 1000 hp
 Engine Weight . . . 2.4 lb/kw @ 500 kw
 1 lb/kw at 2000 kw

(8) Out-of-Core Thermionic System - GA

Study . . . General Atomics
 Power . . . 100-1000 kwe
 TI Cell Power Generation . . . 10 watts/cm²
 Specific Weight . . . 8-10 lb/kwe
 Efficiency . . . 10% 20%
 Radiator Temperature . 3100° F 2500° F
 Power . . . 1-50 kwe
 Heat Rejection Temperature . . . 2000° F
 Overall Efficiency . . 15%
 Specific Weight . . . 20-30 lb/kwe

(9) Out-of-Core Thermionic System - AGN

Organization Aerojet General Nucleonics
Sponsor Primarily Air Force
Power 300-kwe range
Operating Temperature . 2,600 - 2,700° F
Coolant Lithium
System Efficiency
 . Expected 10-12%
Power Output 2-3 watts/cm²
Specific Weight 10 lb/kwe
Current Funding \$300,000 level
Hardware Now Under
 Development Li Pump, purification loop
Ground Test Reading . . . Late 1960's
System Developed 1970's
Development Cost 1/4 cost of Rankine K cycle

APPENDIX F
COMPACT REACTORS UNDER DEVELOPMENT

1. Gas-Cooled Reactors

a. Gas-Cooled Water-Moderated Reactor - ML-1

The ML-1 reactor is a water-moderated, nitrogen-cooled unit which generates approximately 3300 kw of heat. The reactor is fueled by enriched uranium dioxide, Hastelloy-X clad, in 61 fuel-bearing pressure tubes which connect the inlet (upper) and outer (lower) plenums. Reactor heat is transferred to the coolant (99.5 vol% N_2 + 0.5 vol% O_2) as it flows past the fuel pins. Coolant enters the reactor at 800° F and leaves at 1200° F. The demineralized water moderator, surrounds the pressure tubes and flows counter to the gas flow, under forced convection. The core, integral shielding and pressure vessel are contained in a tank of borated water. Six semaphore control blades actuated by electromechanical devices effect reactor control. The lifetime of the first ML-1 core, for a reactor outlet temperature of 1200° F and an ambient air temperature of 100° F, is estimated at 3000 hours, limited by corrosion.

A reactor system representing an advance in the state of the art of gas-cooled water-moderated reactors has been described by AGN. As in the ML-1 reactor, the core consists of pressure tubes which contain the fuel elements mounted between tube sheets. Gas flows from top to bottom and moderator water surrounds the core. Semaphore-type control blades are used. This plant was considered for use with several power cycles and it was concluded that a direct air cycle plant was most favorable under current conditions. These studies envisioned a 500-psi maximum gas coolant pressure and a peak gas temperature of 1600° F.

b. Gas-Cooled Solid-Moderated Reactor - MLX, EBOR

A later AGN study (September, 1962) has considered an advanced gas-cooled and gas-motivated power system using air or nitrogen as a working fluid and a solid moderator. For operating temperatures up to 1600° F, BeO , ZrH_x and YH_x were considered to be satisfactory moderators. Above 1600° F only BeO appears capable of satisfactory operation. Several ar-

rangements of fuel and moderator were studied. A configuration with uniform moderator continuing coolant passages, in each of which a dilute fueled pin is placed, was chosen for the reference design. Fuels investigated included UO_2 , UO_2 -Cr cermet, UO_2 -BeO, UN and UN-Cr cermet. Possible cladding and coating for fuel were studied.

Another GCR under development is General Atomic's EBOR, a beryllium-oxide-moderated, helium-cooled reactor.

2. Liquid-Metal-Cooled Reactors

a. Solid Hydride-Moderated Liquid-Metal-Cooled Reactors - SNAP - 2, 8 and 10A

The SNAP-2, 8 and 10A systems utilize the same type of reactor as heat source. For SNAP-2 and 10A, the reactor units are practically identical except for minor differences in power output and operating temperature. The SNAP-8 is a scaled-up version of these with a thermal power output more than 10 times that of the SNAP-2 and operating at a higher temperature. The important parameters of these reactor systems are shown in the tables in Appendix E and the cycles are shown graphically in Appendix G.

The SNAP-2 reactor core consists of a solid homogeneous fuel moderator, uranium-zirconium hydride, with enriched uranium constituting about 10 percent by weight of the mixed hydride. The 9-inch-diameter core vessel contains a bundle of fuel-moderator elements, each clad with steel specially coated on the inside. This barrier serves to exclude liquid-metal coolant from the fuel and to prevent loss of hydrogen from the hydride. Outside the core vessel there is a beryllium reflector which acts to minimize the core size. Rotating beryllium drums in the reflector, which change reflector thickness, are utilized for control.

The reactor output of 50 kw (30 kw in SNAP-10A) is removed by NaK-78 which flows axially through the core between the fuel elements. The coolant enters the reactor at 1000°F (800°F in SNAP-10A) and leaves at 1200°F (900°F in SNAP-10A).

The SNAP-8 reactor is designed to operate at about 600 kw. To bring its power up to this level the size of the reactor was increased and

the operating temperature was raised. The liquid-metal coolant temperature is 1300° F at the reactor outlet, which is the maximum practical temperature compatible with the materials used in the system.

b. Direct-Cycle Liquid-Metal-Cooled Reactors - ORNL Intermediate, ANL Lunar

Two of the major reactor designs proposed utilize a direct-cycle power conversion system. A boiling liquid metal -- mercury in the ANL reactor and potassium in the ORNL reactor -- serves as the reactor coolant and also the cycle working fluid.

The ORNL reactor itself has not been fully described. However, it has been stated that the reactor will operate at a temperature of approximately 1500° F and utilize conventional materials. The fuel may be stainless-steel clad UO₂. The neutron spectrum of the reactor has not been mentioned, but it is probable that it will be a fast or perhaps intermediate reactor.

The ANL boiling mercury reactor design utilizes a steel containing 5 percent chromium and 1/2 percent molybdenum, for all surfaces in contact with mercury. This is a fast reactor and is of a construction quite similar to that of EBR-I. The core diameter is 41 cm. The reactor is fueled with 492 kg of uranium at 35.6 percent enrichment. Fuel rods are assembled into uncanned subassemblies of 37 or 61 rods each. Control is effected by the movement of four stainless-steel or iron reflector segments. These elements are cooled by hydrogen gas.

The mercury coolant is pumped upward through the core, where boiling takes place. The liquid-vapor mixture is separated in two arc separators above the core and the vapor is discharged to the turbine system.

c. Advanced Lithium-Cooled Compact Reactors - SNAP-50/SPUR

The SNAP-50/SPUR reactor is the most advanced compact system currently under development. The SNAP-50 design concept is a two-loop system employing lithium as a reactor coolant. In a heat exchanger the lithium gives up its energy to another fluid (e.g., potassium) circulating in a high-temperature vapor turboelectric cycle. Because of its high fuel density and thermal conductivity monocarbide fuel is preferred. Uranium

nitride is also being considered. However, until further irradiation data are obtained, final choice of fuels cannot be made. Oxide fuels, UO_2 and $\text{UO}_2\text{-BeO}$ show promise of long endurance and satisfactory performance at some increase in reactor size and weight over that of a carbide-fueled reactor. The reactor will operate at high temperatures up to 2000°F and the use of refractory metals throughout the system is required. Columbium must be used for all surfaces exposed to lithium.

APPENDIX G

POWER CONVERSION SYSTEMS UNDER DEVELOPMENT-DESCRIPTION

1. Thermomechanical Conversion

a. Brayton Cycle Systems

(1) Direct Closed Cycle Gas Turbine Power Conversion System - ML-1

The ML-1 utilizes a direct closed gas cycle (modified Brayton cycle) power system. Gas heated in the reactor core drives a turbine to produce power and is then recirculated through the core. The major components of the power conversion system are: turbine-compressor set and reduction gear; alternator and starting motor; precooler with fans; recuperator; switchgear and connecting piping and valving. Hot gas from the reactor expands in the turbine. It then passes through the low-pressure side of the regenerative heat exchanger (recuperator) and through an air-cooled precooler which rejects heat to the atmosphere. The gas is compressed in the compressor, preheated to 800° F in the high-pressure side of the recuperator, and returned to the reactor. The direct-coupled compressor and alternator are driven by the turbine through a gear-box. A brushless alternator operating at 3600 rpm is driven by the turbine shaft to produce the net electrical output of 300-500 kwe at 60 cps.

(2) Brayton Cycle Fluid Engine - Cleveland Pneumatic Tool Company

The Cleveland Pneumatic Tool Company's Brayton Cycle engine, variously referred to in their literature as a "Liquid Thermal Engine" or as the "Thermex Engine," operates on a Brayton, or Joule, closed gas cycle.

The ideal Brayton cycle, shown in the diagrams below, consists of an isentropic compression, a constant pressure heat addition, an isentropic expansion, and a constant pressure heat rejection.

The Cleveland Pneumatic engine, shown schematically below, consists essentially of a piston and cylinder, a cooler and a heater. As the piston arrives at its topmost position, compressed heated fluid is admitted into the expansion chamber and, in expanding, forces the piston downward. During the downward stroke, cooled fluid below the piston is compressed and forced into the heater. On the upward stroke, cooled expanded fluid flows from a cooler into the compression side of the cylinder while, simultaneously, fluid expanded during the downward stroke is forced into the cooler. The engine operates, therefore, on a two-stroke cycle, with the reciprocating engine replacing the compressor and the turbine of a conventional gas cycle.

At the present time, Cleveland Pneumatic has in operation a demonstration engine of 1 hp which uses carbon dioxide as the working fluid. Operation is in the supercritical region. Other fluids with low critical temperatures are being investigated, particularly the various freons. Cleveland presently has a contract with the AEC for performance of parametric analyses for an engine to operate with the ML-1 and PM-2A reactor systems. This work includes evaluation of various operating fluids, determination of optimum temperatures and pressures, and optimum system configuration. No hardware study is involved, and the results will be based upon theoretical calculations only. The analysis is, in essence, a feasibility study.

The original application for this system was envisioned as propulsion for submarines utilizing a pair of 7500-hp engines. Other power generation applications are now being sought. Cleveland considers their engine to be most applicable at relatively low temperature. It is not expected to be competitive with conventional engines in the normal range of steam or gas turbine operating temperatures.

A number of calculations were made to evaluate the performance of the Brayton cycle engine operating with carbon dioxide. Since the T-S diagrams for CO₂, -- available in the literature -- do not cover the high-pressure regions of interest, fluid properties outside the range of the curves were calculated using generalized compressibility methods. Calculations were made for the reversible cycle, assuming isentropic expansion and compression.

Most of the data in Cleveland's brochures are for presentation purposes, and are not suitable for system performance calculations. However, two sets of data, trials No. 101 and 102, are of sufficient detail to serve for cycle calculations. Both sets of data are for a pressure range of 900 to 2500 psia, with a compression intake temperature, T_1 , of 70° F. This temperature is based upon seawater coolant at 35° F, which is unrealistically low for our purposes. Therefore, assuming air as a coolant, T_1 was raised to 100° F. On this basis, for trial No. 101 which has a maximum temperature of 400° F, the thermal efficiency of the ideal cycle was calculated to be in the order of 16 percent. This is considerably below the values of 21 to 34 percent indicated elsewhere in their brochures. It should be noted that with T_1 at 100° F. the cycle is completely within the supercritical region, whereas at 900 psia the cycle will enter the two-phase region at temperatures below 75° F.

Cleveland's computer output data for the Brayton cycle operating with CO₂ has been analyzed. All the data are based upon a minimum fluid temperature of 120° F, assuming air at 70° F as coolant. Calculations are for 500, 800 and 1100 psia pressures after expansion and 3500, 5000 and 10,000 psia pressures after compression. The maximum temperatures at the start of expansion range from 550 to 1200° F. Reported thermal efficiencies

for the various cases vary from 25 to 44 percent.

The effect of irreversibility on the thermal efficiency was estimated from equations for a real process for a perfect gas. Although the assumption of a perfect gas is not representative for CO_2 , the results serve to indicate the variation of thermal efficiency with expansion and compression efficiencies. The tabulation below shows the results of the calculations for the case considered above.

$\eta_c = \eta_e$	Thermal efficiency %
1.0	32.6
0.95	26.8
0.90	20.4
0.85	13.4

Expansion and compression efficiencies are assumed equal. The real cycle shows a severe decrease in thermal efficiency compared to the ideal cycle when irreversibilities are taken into account. Thus, while the computer calculations show very attractive cycle performance, the efficiency of the actual engine could be expected to be no better than that of conventional gas cycle systems.

Aside from thermodynamic aspects, considerable development work remains to be done on the Brayton cycle engine. Cleveland Pneumatic lists the following areas requiring research and development:

Valving

- i. Minimization of pressure drop, noise, unswept volume
- ii. Evaluation of wear characteristics and life assurance
- iii. Development of final valve timing and actuation system

Fluid Characteristics

- i. Analysis and evaluation of additional fluids
- ii. Laboratory evaluation of certain characteristics vs present extrapolation
- iii. Analysis of fluids with regard to contamination, breakdown, life

✓

Engine Configuration and Performance

- i. Evaluation of seals and bearing designs
- ii. Arrangement to balance high loads and load reversals
- iii. Study power converter to optimize system
- iv. Reduction of fluid and mechanical friction
- v. Evaluation of engine flexibility and optimum configuration
- vi. Analysis of environmental effects on engine
- vii. "Off-design" operating performance

✓

Development of Theory

- i. Throttling theory
- ii. High-pressure heat transfer analysis

Research in the areas indicated above is needed before an engine operating with a conventional heat source can be developed. Additional work is required if the engine is to be used with a nuclear reactor heat source. Transition from the present 1-hp demonstration engine to a large power operating engine competitive with other systems appears to be many years away.

b. Rankine Vapor Cycle Systems

(1) Snap-2 and 8 - Indirect Mercury Rankine Cycle Power Conversion System

The SNAP-2 system utilizes a Mercury Rankine Thermomechanical Energy Conversion Cycle.

The liquid metal reactor coolant, circulated by a rotating permanent magnet pump, flows from the reactor outlet at 1200°F to the boiler superheater where it gives up some of its energy to heat, evaporate and superheat mercury, the power cycle working fluid. The boiler superheater is a concentric tube, counter flow, once-through unit, with NaK flowing in the annulus. Mercury is boiled at 900°F and 100 psia, superheated to 1150°F, and then expanded through a two stage axial flow impulse turbine. The exhausted mercury is then condensed at 600°F and 6 psia in a radiator-condenser which radiates its heat to space. The condensate, subcooled to 420°F, is returned to the boiler by a miniature centrifugal pump operating in combination with a jet booster pump. The turbine drives an alternator--a permanent magnet machine with a sealed stator which delivers up to 3.5 kw at 110 volts and 2000 cps. All the moving parts of the power conversion system are mounted on a conversion shaft. This combined rotating unit (CRU) is sealed to prevent the loss of cycle fluid. The shaft of the CRU is supported by mercury lubricated bearings and rotates at 40,000 rpm. The mercury boiler feed pump also supplies pressurized mercury to the bearings.

As visualized at the present time, the SNAP-8 system power conversion unit will consist of one or two 30 kwe mercury turbo-alternator units.

SNAP 2 POWER CONVERSION SYSTEM

ADVANCED SNAP-2 COOLANT FLOW DIAGRAM
11-Kwe SYSTEM WITH 3 CRU LOOPS

SNAP - 1

SNAP - 2 POWER CONVERSION EQUIPMENT

(2) Boiling Liquid Metal Direct Rankine Cycle Power Conversion System

In the ANL Boiling Reactor approximately 10 percent of the coolant in vapor form goes to the two-stage impulse turbine. The remainder of the mercury is recirculated by a pump. The turbine discharges mercury vapor into a plenum from which the vapor goes to the radiator. The radiator condensate lines are piped upward to the condensate pump. A 400-cycle a-c brushless exciter type generator is an extension of the turbine shaft. The main bearings are journal bearings lubricated with mercury. The mercury which runs out of the ends of the bearings is continuously blown down to the condenser. Since the generator and turbine are on the same shaft, one thrust bearing is required. It is of the Kingsbury type, mercury lubricated.

Two mercury pumps are needed--a recirculating pump and a condensate pump. The recirculating pump handles a large volume of mercury at low pressure ratio and the condensate pump handles a small volume at high pressure ratio. Since the output of both pumps goes to the inlet header of the reactor, their impellers are integral with separate inlets and a common volute, thus saving weight and space. A radial flow impeller is used for the recirculating pump. This unit is mounted on an extension of the main turbine shaft.

Hydrogen is used to cool the winding of the main generator and the windings and bearings of the auxiliary motor as well as the reflector control elements. An axial flow compressor mounted on the main shaft circulates the hydrogen. A radiator is provided to cool the gas.

Very little information has been given on the power conversion system of the ORNL reactor. It will operate at about 1500°F, utilize a potassium Rankine cycle and will be based on conventional materials and technology.

SIMPLIFIED POWER CONVERSION CYCLE
BOILING REACTOR

$h_{gt} - 75.84 \text{ cal/gm}$
 $Se^t - 0.128 \text{ cal/gm}^\circ\text{K}$
 $P_{gt} - 1.286 \text{ atmos}$
 $T_{gt} - 371^\circ\text{C}$
 $X_{gt} - 0.915$

$S_d - 0.120 \text{ cal/gm}^\circ\text{K}$
 $P_d - 11.91 \text{ atmos}$
 $T_d - 582^\circ\text{C}$
 $h_d - 86.28 \text{ cal/gm}$

$Se - 0.120 \text{ cal/gm}^\circ\text{K}$
 $Pe - 1.286 \text{ atmos}$
 $Te - 371^\circ\text{C}$
 $he - 71.4 \text{ cal/gm}$
 $x - 0.85$

$So - 0.0279 \text{ cal/gm}^\circ\text{K}$
 $Pa - 1.286 \text{ atmos}$
 $To - 371^\circ\text{C}$
 $ha - 12.05 \text{ cal/gm}$

ANL FORCED CIRCULATION
BOILING REACTOR

(3) SNAP-50 System Power Conversion - High Temperature
Liquid Metal Indirect Rankine Cycle

a. The SNAP-50 advanced reactor system provides the potential for development of a direct boiling reactor if this approach proves more attractive than a two-loop system. There is no specific design for this reactor available at present but the cycle would probably be similar to that of the ORNL reactor system, utilizing more advanced materials technology.

Liquid lithium leaving the reactor enters the boiler where it gives up energy to boil the potassium working fluid. The lithium leaving the boiler is recirculated through the reactor by the reactor loop pump. The working fluid is boiled to about 90 percent quality. Entrained liquid is removed in a separator, cooled, and returned to the circulating pump. The vapor is expanded through a turbine and condensed in the radiator.

For a 1000 mwe system studies have been made of a SNAP-50 type reactor system coupled to four 250-kw rotating machinery packages. This concept reduces system weight and size and increases reliability.

Each of the rotating units consists of turbine, alternators, and jet-boosted centrifugal condensate pump. The shaft rotates at 24,000 rpm. The alternators are mounted between the journal bearings, with the turbine and pumps cantilevered from the opposite ends of the shaft. One alternator produces 250 kw of 1600-cps power. An auxiliary alternator produces 2-1/2 kw of 400-cps power to drive the motors and pumps. There is also a control alternator. These alternators are maintained at 800°F by the coolant. The bearings are potassium-lubricated. Potassium condensation takes place inside a compact heat exchanger from which sodium conducts the heat to a space radiator.

ONE OF FOUR 250 KW ROTATING MACHINERY PACKAGES

SPECIFIC PERFORMANCE OF HIGH-TEMPERATURE,
 RANKINE-CYCLE NUCLEAR APUs.

SNAP - 50, SPUR CYCLE CONDITIONS (300 Kwe)

Reduce to $4\frac{1}{2}$ "

on Page G14
Top Section

c. Stirling Cycle Engine

The major components of the Stirling engine are the engine heater, the regenerator, the engine cooler; a displacer piston to control the movement of the working fluid through the heater, regenerator, and cooler; and the power piston to compress and expand the gas.

The heater for the space version of the Stirling engine consists of thin-walled U-tubes which are uniformly spaced around the cylinder. These high-temperature steel tubes are enclosed in a stainless steel housing through which liquid metal is pumped to transfer heat to the heater assembly.

An effective regenerator must store approximately 6 Btu for each unit of heat input and must do it in a very short period of time with minimum pressure loss. The regenerator used consists of cylindrical assemblies, containing stacks of stainless steel wire, placed about the periphery of the cylinder wall.

The cooler assembly, directly below the corresponding regenerator, consists of many small diameter tubes located in an annulus through which a cooling fluid circulates.

The displacer piston, a hollow, stainless steel shell, fits loosely in the cylinder. Its motion is controlled by the displacer piston rod, connecting it to the drive mechanism in the crankcase. There is only a slight pressure differential between the top and bottom of the displacer piston and, since some leakage of gas between the displacer and the cylinder wall can be tolerated, piston rings or other means of sealing the piston to the cylinder are now required. The top of the displacer piston operates at approximately the temperature of the cooler. The piston walls are made as thin as possible to reduce conduction heat losses and two internal, horizontal baffles are provided to reduce radiant heat losses.

The power piston provides the work for compression of the gas and must deliver the work of expansion. Therefore a considerable pressure difference can exist between the top and bottom of the piston. This piston must fit closely into the cylinder and must be sealed to prevent gas leakage. Since this piston is located in the cold zone of the engine there is a greater choice of sealing means available than for internal combustion engine pistons.

which experience much higher temperatures. Forces are transmitted from the power piston to the drive mechanism by means of the power piston rod. This rod is hollow and the displacer piston rod passes through it. The Stirling engine would generally have a low specific power output if it were operated at low mean cycle pressures. Accordingly the engine is designed to operate at higher pressures, and a buffer space is provided so that a thermodynamically inert volume of working fluid can be trapped below the power piston. The mean pressure of the gas in this buffer space is maintained at a level near the mean pressure of the working fluid above the power piston. This arrangement reduces the loads on the drive mechanism. Existing designs by the Allison Division of General Motors for space application utilize a buffer mean pressure of 1500 psi. The fluctuations about this mean in the actual working medium above the power piston are of the order of ± 500 psi.

The motion of the two pistons is controlled by a rhombic drive mechanism, consisting of two similar contrarotating shafts with one crank on each crankshaft. The shafts are timed together by a pair of helical gears.

An important feature of this drive mechanism is its symmetry, which permits complete dynamic balancing of all moving parts of the engine by the use of suitable sized counterweights located on the crankshafts.

2. Thermoelectric Conversion Systems

a. Conduction Cooled Direct Conversion - SNAP-10

One direct conversion scheme that has been studied for reactor units up to a few hundred kilowatts is the use of thermoelectric couples located on the surface of a solid reactor, with heat transferred to the surface by thermal conduction. Control is maintained by reflector movement. The original SNAP-10 unit utilized this type of conduction thermoelectric conversion.

SNAP-10 was designed for 300 watts output. The reactor fuel was uranium-zirconium hydride similar to that used in the SNAP-2 reactor, but in the form of stacked circular plates rather than rods. The core was beryllium reflected. Heat transfer was by conduction, with the power level limited by geometry from the core to an annular array of thermoelectric converters and thence to radiating fins attached to the cold junction of the thermocouples. Operating temperatures for such unit are limited by the maximum capability of presently available thermoelectric materials (up to 1100° F).

NUCLEAR REACTOR WITH AN EXTERNAL DIRECT-CONVERSION GENERATOR

NUCLEAR REACTOR WITH DIRECT-CONVERSION NUCLEAR FUEL ELEMENTS

DIRECT-CONVERSION NUCLEAR REACTORS

Reduce to 3.7"

on June 6/7

b. Convection Cooled Direct Conversion - SNAP - 10A

The SNAP-10A power conversion system utilizes thermoelectric couples. The liquid metal reactor coolant (NaK) transports heat to the thermoelectric converters, where some of the heat is converted to electrical output, while the rest is rejected by radiation to space. Since an electromagnetic pump is used to circulate the coolant, there are no mechanically moving parts in the system during operation. The lead telluride thermoelectric converters are mounted in series between the NaK coolant tubes (hot junction) and the radiator (cold junction) but are electrically insulated from them. Temperature limitations of the thermoelectric material used (Pb-Te) limit the efficiency attainable from the couples to about 10 percent of Carnot. The overall power conversion system efficiency is about 2 percent. The NaK, leaving the reactor, flows within small diameter tubes where heat is given up to the thermoelectric elements attached to the tubes. These elements are series connected to produce an electric output of 500 watts. Waste heat conducted from the cold junction of the thermoelectric elements is rejected to space by radiation from individual radiators.

SNAP-10A THERMOELECTRIC SYSTEM

Reduce to 6"

(G-1) 151

c. In-Core Direct Conversion

The design of nuclear systems utilizing thermoelectric elements in a reactor core has been investigated. One of these is a Westinghouse reference design for a thermal circulation, light-water, low-pressure, 500-kwe d-c power reactor. This unit uses in-core Pb-Te series-type thermoelectric couples with the thermoelectric material wrapped around the nuclear fuel element. Present estimates indicate that a 5 percent overall core electrical efficiency is feasible utilizing the best thermoelectric materials now available. The reference fuel element design utilizes Fe-U as fuel material; however, the use of Nb-U or BeO-U is also considered.

A thermoelectric in-core system has also been studied by Texas Instruments Co. The reactor is a UO_2 fueled, water-cooled reactor containing PbTe thermoelectric elements. The study indicated that it is possible to build a PbTe thermoelectric reactor with a uranium enrichment less than 20 percent. Analysis of the transient behavior of thermoelectric fuel elements indicates that because of the low thermal conductivity required of thermoelectric elements, the fuel will be thermally isolated from the water coolant for relatively large time lags. This means that the thermoelectric nuclear fuel element is unsafe, if reliance is placed wholly on the negative coolant-moderator temperature coefficient to control transients. The use of UZrH fuel is one suggested solution to this problem.

The reference fuel element design is cylindrical, built around 0.5-in. diameter UO_2 fuel pellets. It consists of two sections each 3 feet long joined to form a total 6-foot length. Cladding is stainless steel. Al_2O_3 is used for electrical insulation. The core consists of 4118 elements. The study indicated that a reliable core of such element cannot now be fabricated at reasonable cost. Because of pumping power requirements, a forced-convection plant with only direct conversion power is not attractive or even feasible at present thermoelectric efficiencies.

The natural circulation direct conversion reactor is feasible but the thermal performance of such a plant is poor compared with conventional plants. This plant can be utilized to generate steam for space heating or other use.

3. Thermionic Conversion Systems

a. Conduction-Cooled Direct Conversion - STAR-R

The General Electric STAR-R is a radiator-cooled nuclear thermionic conversion system which does not require heat transfer fluid. A hollow cylindrical fast reactor is surrounded by thermionic converters. Reactor heat passes through the converters, where part of it produces electricity while the rest is radiated from the TI cell anodes. A typical 70-kwe STAR-R power plant is about 3 ft long by 1 ft in diameter. Uranium carbide fuel is contained in a central ring. There are two end reflectors to which bus bars are attached. Thermionic converter cells are located on the outside surface of the reactor. Each cell has its own cesium reservoir. During operation the reactor may have no moving parts. Fuel element temperature is maintained by the negative coefficient of reactivity or by reflector control if necessary. Heat is conducted from the nuclear fuel directly to the cathode of each converter. The heat that is not directly converted to electricity is rejected by radiation from the anodes. The anodes also serve a neutron reflector for the core.

NUCLEAR REACTOR WITH DIRECT-CONVERSION NUCLEAR FUEL ELEMENTS

Strick in
p. 6-20

b. Out-of-Core Direct Conversion - SNAP-50/SPUR

In the advanced SNAP-50 program, an improvement or backup for the turbomachinery conversion unit is the use of thermionic converters located in the heat exchanger or heat rejection radiator. The feasibility of this approach depends on development of reactors operating at increased temperatures and development of reliable, efficient, low-temperature thermionic converters.

The reactor coolant can be circulated directly to the radiator where thermionic converters are mounted on the radiator tube wall. A cathode power density of $1\text{-}1/2 \text{ watt/cm}^2$ results in an overall plant efficiency of 9% and a plant specific performance of less than 10 lb/kwe. In the AGN design, thermionic converters are arranged in a cylindrical configuration concentric to the radiator tubes. The cathode cylinders are sleeves mounted on the radiator tubes on a thin layer of electrical insulation. The anode is another concentric cylinder attached to the radiator fins. Energy flows from the liquid metal in the tubes through the tube wall and converter to the radiator fin. Cathode and anode surfaces are nested with internal electrical connections providing structural support. Concentricity and alignment are provided by the radiator tube. A cesium diode requires an anode-cathode gap of between 0.004 and 0.1 inch depending on the type of cathode used. Seals are located so that there is no thermal gradient across a seal. The temperatures of the system depend on the temperature required for diode operation and reactor materials technology (possibly in the range of 2100-2800°F.)

Other studies have utilized an additional convection loop to transfer heat from the anode to a radiator. If the reactor coolant is an electrical conductor, additional electrical insulation is required between it and the inside surface of the emitter. Similarly, if the cold fluid is an electrical conductor, electrical insulation is required on the outside of the collector tube.

THERMIONIC RADIATOR CYCLE CONDITION
SNAP-50

THERMIONIC CONVERSION ON RADIATOR
(SNAP 50/SPUR)

c. In-Core Direct Conversion

A typical design for in-core thermionic reactors is General Electric's STAR-C plant. The 1000-kwe liquid-metal-cooled thermionic system consists of a reactor, liquid metal headers which connect the reactor to the heat exchanger, and the primary liquid metal coolant pump. In this system, waste heat which passes through the converter is removed by the liquid metal coolant and rejected in a radiator. The thermionic converters located in the reactor itself are able to take advantage of the high temperatures available in the core. The thermionic cells are contained within the fuel elements. A cesium reservoir for each element is bound at the top of the reactor. Bus bars at the top collect the power produced.

A General Atomics study of a similar type of in-core thermionic system of 300-kwe output resulted in the design of a reactor core 11 inches high by 13 inches in diameter (with 2-inch reflector) containing 546 fuel element cells, weighing 582 lbs. Overall cycle efficiency was found to be 12% when thermionic cells of 14% efficiency and 11 watts/cm² power density were used. Lithium coolant, circulated by a d-c Faraday type electromagnetic pump enters the reactor at 1800°F and leaves at 2000°F. It then flows to the radiator where heat is rejected to space.

The In-Core thermionic unit designed by Atomics International utilizes a converter with emitter structure of nuclear fuel containing high enrichment uranium mounted concentrically within the collector. Operating temperature of the diodes is expected to be 3000°F or greater.

Each thermionic cell is capable of producing approximately 1 volt. Therefore, in order to obtain a useful high voltage, many cells must be connected in series.

A cell consists of a cathode or emitter, an anode or collector and an interelectrode space filled with cesium vapor. Electrical insulation provides the proper electrical configuration and the entire element is isolated from its external environment by an outer sleeve which also serves to maintain the structural integrity of the unit.

The cathodes are formed by solid cylinders of fuel material, held in place by insulation separators. Heat resulting from nuclear

SCHEMATIC DIAGRAM OF A HIGH-POWER REACTOR,
THERMIONIC SPACE POWER PLANT

REACTOR SPECIFIC WEIGHT vs POWER
IN-CORE THERMIONIC CONVERSION

fission causes the cathode to emit free electrons into the interelectrode space. The anode, which collects emitted electrons, is in the form of a thin shell of metallic conductor surrounding the cathode. Electrons impinging on the anode heat the surface. This heat must be removed if the electrode temperature difference is to be maintained. The anode must be in contact with the outer sleeve which is cooled by circulating liquid metal. To prevent the electrical shorting of adjacent anodes, a layer of electrical insulation must be interposed. Cesium vapor is ionized in the high temperature environment, and serves to neutralize the space charge produced by electrons emitted into the interelectrode space.

Reactor control is maintained by movable reflector sections. A cesium circulation system is provided. This design utilizes two sets of converter elements, the main power supply and an auxiliary supply providing power to drive the EM pump. Two cesium reservoirs for the two converter sets, are located outside the reactor. Cesium gas is continuously bled to the cells and removed to minimize fission product gas concentration in the diodes. NaK is the reactor coolant in this design. The use of EM pump allows operation of the coolant system as a completely sealed unit.

A nuclear thermionic unit is currently under development by RCA and B&W for The Bureau of Ships. A typical device has been run for more than 300 hours at an average power of 135 watts and efficiency of about 10%. This cylindrical thermionic converter operates at an emitter temperature of 1,350°C and produces 2.6 watts/cm² at 327 amps and 0.48 volts.

4. Other Conversion Systems

a. Magnetohydrodynamic (MHD) Power Conversion

A very advanced power conversion system utilizing the SNAP-50 reactor would involve the use of a metal vapor magnetohydrodynamic cycle.

APPENDIX H

STATUS OF POWER CONVERSION SYSTEMS UNDER DEVELOPMENT

1. Mechanical Conversion System

<u>Company</u>	<u>System</u>	<u>Status</u>	<u>Reference</u>
<u>Rotating Systems</u>			
Thompson Ramo Wooldridge	Rankine Cycle Mercury Turbine for SNAP-1, -2 SPUD & Sunflower	Output 0.53 kw 2510 hr, SNAP-1 System run 1960 500 hr SNAP-2	ARS-2166-61
	Mercury Materials Program		
Aerojet-General Nucleonics	Rankine Cycle Mercury Turbine for SNAP-8	Under development 1 yr, component work No complete units	ARS-2166-61
	Mercury Materials Program		
Sunstrand Turbo Sunstrand Machine Tool Company	Rankine Cycle - Rubidium Turbine for 15-kw solar system	Applied Research and Component Development	ARS-2166-61
AiResearch-Garrett Corp.	Potassium Power System Study for AF-ASD 300-1000 kw	Conceptual design and materials data studies; starting component development	ARS-2166-61
<u>Reciprocating Systems</u>			
Allison Division of General Motors	Sterling Engine - Solar Powered	Second phase of develop- ment; redesigning into flight type hardware - no firm program. Experimental en- gine - operated 100 hr	ARS-2166-61
Cleveland Pneumatic	Liquid Thermal Engine Brayton Cycle	Conceptual design - 1-hp demonstration engine operated. Parameter studies being conducted.	ARS-2166-61

2. Comparison of Operating Conditions of Closed Cycle Dynamic Engines

The cycle efficiencies of various dynamic cycles and working fluids are affected by many variables. The actual conversion efficiencies are about 30 to 50% of these ideal cycle efficiencies when the turbine, alternator pump, and accessory losses are taken into consideration. It is evident that the Stirling cycle and the mercury binary cycle have higher ideal and actual energy conversion efficiencies than do any other practical energy conversion systems.

Many other working fluids and cycles have been studied recently but none has been found to be particularly attractive for use in a practical power conversion system. The Brayton cycle has also been studied extensively in the past few years. The problems of rejecting heat through a radiator containing a gas rather than a condensing fluid result in a major weight disadvantage, and for equal efficiencies, the Brayton cycle energy source must operate approximately 600°F higher than does that of the equivalent Rankine cycle. These disadvantages have led to the elimination of the gas cycle from consideration for current space power systems. An advantage of the Brayton cycle using a noble gas working fluid is the elimination of many materials problems. This advantage is partially offset however by the higher temperatures required and the associated materials problems resulting from the large thermal gradients produced. Among the working fluids studied briefly for use in the Rankine cycle are sulphur, phosphorous, aluminum bromide, cesium, lithium, bismuth and lead. Complex phase changes take place in phosphorous in the range of desirable cycle temperatures making it a relatively unattractive working fluid. Materials containment problems eliminate sulphur and aluminum bromide from further consideration. The state of development of mercury engines is such that even if these materials problems were solved, the use of sulphur, phosphorous or aluminum bromide would not result in a change in cycle performance great enough to warrant development of power systems utilizing these working fluids. Lithium vapor, bismuth and lead cycles require reactor and dynamic machinery capable of operating at approximately 3000°F. The development of dynamic engines utilizing these three working fluids is remote.

Cesium has favorable pressure-temperature properties at moderate temperatures as well as a low turbine inlet and exit specific volume. The use of cesium has been considered recently and further work on the merits of cesium systems is expected to follow current efforts on potassium and sodium cycles.

3. Dynamic Space Systems

Of the many combinations of cycle, working fluid, and power size only three power systems are now being developed to the point of demonstrating useful operation in space. They are SNAP-8, a 30-kw nuclear turboelectric mercury vapor power system; SNAP-2, a 3-kw nuclear turboelectric mercury vapor power system; and sunflower, a solar turboelectric unit. All other closed cycle power systems under development are limited to applied research, component development to demonstrate practicality, or the development of a system to demonstrate ground operation. SNAP-2, Sunflower, and SNAP-8 are expected to be available for orbital operation in the mid-sixties.

The most extensive development of the Rankine cycle for space use has been with mercury engines having an output of 0.5, 1.0 and 3.0 kw per turboalternator unit. This work has been done under the SNAP-1, SNAP-2, SPUD, and Sunflower programs at Thompson Ramo Wooldridge, Inc. An experimental SNAP-1 power conversion system underwent a 2510-hour endurance run in 1960. The SNAP-2 power unit has been operated numerous times in continuous checkout runs during development tests for periods up to 500 hours.

The development of SNAP-8 mercury power conversion systems in the range of 20 to 40 kw has been under way for approximately one year at Aerojet-General Corporation. Component work has been devoted to experiments and tests to determine the validity of design concepts. In the near future, the results of current work are expected to lead to the design and test of complete components for ground operation. During the next few years, these components are to be developed into a system for ground operation and eventually into flight type hardware.

The development of mercury power systems in the range of 30 to 60 kw is at the present time highly dependent upon NASA plans related to the 30-kw SNAP-8 machine and its successful development. Present NASA plans

are for power systems using multiples of the 30-kw turboalternator unit. Work to date has been limited to conceptual and preliminary design. For powers between 100 and 300 kw, limited studies have been conducted on the mercury engine. They indicate the practicality of using mercury at larger powers. Only cursory investigations have been conducted to date for power sizes above 500 kw for a mercury engine. There is a need for sophisticated study in comparing the mercury engine with potassium and sodium engines to determine the potential of the mercury engine in terms of power and operating temperature.

All the significant work on the rubidium Rankine cycle has been limited to applied research and state-of-the-art component development for a nominal 15-kw solar turboelectric power system. This work is being done at Sundstrand Machine Tool Company. In comparing the rubidium power system with potassium and sodium for large nuclear turboelectric power, there seems to be little advantage in employing rubidium. Consequently, applied research in the large power systems has favored potassium and sodium over rubidium.

The majority of work under way in the potassium and sodium power systems has involved the study of power system concepts at the conceptual design level in applied research related to thermodynamic properties of working fluids and materials problems. A systematic study of 300- to 1000-kw potassium power system is currently entering a second phase at Ai-Research Division of the Garrett Corporation. This program is being conducted for the Aeronautical Systems Division of the Air Force. Some component development has been undertaken, particularly in bearings, to investigate the problem areas associated with components. Component development has not progressed to the point of operating components of large size which may be used in 100-to 1000-kw power systems.

The development of the large high-temperature nuclear turboelectric power systems is critically dependent upon the development of reactors capable of operating at approximately 2000°F for greater than 10,000 hours. The implementation of a development program for large alkali metal power systems depends to a great extent on the applied research recently initiated and currently planned.

The Stirling engine being developed by the Allison Division of the General Motors Corporation for ground demonstration of an engine capable of being powered by solar, or possibly nuclear energy, is in the second phase of development. The current effort is oriented toward re-design of the engine into flight type hardware; however, no firm program exists to develop the Stirling engine to an operational system for orbital use. The near term goal of the program is to develop a 3-kw power unit capable of operating at or above 30% efficiency and demonstrating 1500 hours of endurance. Current development efforts have as a goal the reduction of the power unit package from 237 pounds to 135 pounds for a 5-kw unit.

Recently, potential requirements for power systems operating up to 15,000 hours have been indicated for space missions which are being considered for this decade. This long operating life requirement for large power systems such as the potassium engine points up the overwhelming problem in high temperature nuclear turboelectric power system development. The problem is that of ensuring high reliability. Past development experience has indicated that it takes a minimum of 5 to 8 years to develop somewhat similar advanced power systems for considerably less stringent life specifications.

4. Availability of Dynamic Space Power Systems

An estimate of the probable availability of dynamic power systems through 1972 is shown on the following page. The availability of the simple mercury, solar and nuclear power systems is reasonably firm for power up to 30 kw.

Above 30 kw, power systems may be developed in the years indicated if a logical development proceeds without going to crash programs. The nuclear mercury power system can be developed for powers as high as 500 kw, probably by the end of 1968.

The nuclear potassium power system may fulfill power requirements between 300 kw and a few megawatts. Availability of a potassium system is given as late 1972. This date assumes that a major development effort for an operational one-megawatt power system will begin in 1963 after significant results have been obtained in materials, and heat transfer tech-

nology. During the nine-year development period, it is assumed that reliable operation for 10,000 hours can be achieved. It is expected that the development of the megawatt-sized power system operating at temperatures near 2000°F will be as difficult and as costly as the development of the indirect liquid metal aircraft nuclear propulsion system which is no longer funded in the U.S. budget.

ESTIMATED DATE OF PROBABLE AVAILABILITY OF DYNAMIC SPACE POWER SYSTEMS

5. Dynamic Power Systems - Development Problems and Programs

Materials problems for the larger mercury engine under development are apparent because the maximum mercury temperature is 1300° F as compared with 1200° F for the SNAP-2 engine. The increase of 100° F has a significant effect on corrosion rates and formation of corrosion products. At the present time, lack of data makes the prediction of limitations on the operating life of the power systems at 1300° F due to materials degradation difficulty.

There are many unknowns related to the potassium and sodium systems. Materials problems are among the most important. Data now available are not sufficient to allow a realistic estimate of the effect of materials properties on the operating life of power systems to be made. Since these systems are intended for use in missions of more than 10,000 hours duration, comprehensive and long-term materials test efforts are required.

The development of generator materials, particularly materials for the bore seal, represents a major area of effort in the near future. It is desirable to use as a seal a non-conductor, such as a ceramic material, to keep internal generator losses low. The condensing temperature of the potassium and sodium engines is expected to be higher than the maximum allowable generator temperature, therefore it is necessary to introduce the use of auxiliary subcooled potassium or sodium loops to cool the generator.

An important aspect of potassium and sodium engine development is system performance and environmental testing. The use of columbium as a structural material makes the testing of these engines considerably more difficult than the testing of mercury engines. Columbium reacts easily with air and impure noble gases, therefore, it is highly desirable to test alkali metal engine components, and the systems themselves, in a high vacuum. In mercury engine development, test rig components have required considerably more maintenance than the components under test. This is expected to be true for alkali metal engine development as well. The test programs will have an even greater complexity because maintenance will probably have to be done in large vacuum test facilities. It may be necessary for repair work to

be performed by technicians wearing "space suits" which are connected to the outside atmosphere. Alternative methods include maintenance by remote control and repeated pumping down of the test facility after repair.

Major development areas for the Stirling cycle engine include sealing, lubrication, balance and vibration. The seals represent a problem in that they can allow the lubricant to cause obstruction in the heat exchanger with a resulting degradation in performance. Current plans at Allison are to continue systematic development of seals where geometries and materials are varied with the goal of long life and essentially zero leakage. To date, according to Allison, no problems have been encountered which forestall or limit the future development of the Stirling engine. No scientific break-throughs are required. The present state-of-the-art, supported by past engine experience, should ensure reliability.

6. Status of Thermoelectric Conversion Devices

The SNAP systems developed by Martin testify to the advanced state of the isotope thermoelectric devices. Devices rated in the kilowatt range are still in the developmental stage. Lead telluride is the standard material for thermoelectric use and efficiencies are of the order of 7%.

The upper operating temperature limit is 700° C -- 600° C if long life is required.

This binary compound melts at 922° C and can be made P or N type by suitable doping (i.e., with sodium for N and potassium iodide for P). Voltage per junction is between 60 and 200 mv, requiring a large number of cells in series for reasonable voltage outputs. Lead telluride is probably not the best choice for in-pile reactor use because of the 4-barn cross section of tellurium. Typical materials under development include LiNiO, GeTe, BiGeTe, LaBaTiO₃, NaUO₂, ZnSb, CeSe₄, and Bi₂Te₂. Irradiation tests so far indicate that radiation damage can be annealed, but tests with higher fluxes will be required before a definite conclusion can be drawn. It is believed that within about two years, thermoelectric materials will be developed that can operate at 1500 to 1800° C. Some such materials already exist, but their electrical and thermal properties have not yet been measured. It is difficult to maintain a high figure of merit at these elevated temperatures;

therefore, the major research effort at present is directed toward materials investigation. Westinghouse has the only large-scale program on irradiation testing. The theoretical maximum efficiency for thermoelectric devices is now 25% and has been extrapolated by some workers to 35%. The thermionic devices have theoretical efficiencies of 30% with extrapolations to 40%. These efficiencies can be misleading however, since in some cases part of the heat lost can be recovered to produce useful power, the amount depending on the heat sink temperature.

Liquid thermoelectric materials are at present in the realm of ideas and conjecture. A better understanding of the basic phenomena is needed. The Seebeck coefficient of Cu_2S in the liquid state has been measured as 300×10^6 v/°C at 1200° C, which is comparable to that of PbTe at 400° C. Higher temperature operation implies higher Carnot efficiency. However, the big disadvantage of liquids in connection with the large temperature gradient required for efficient operation is the possibility of high heat transfer by convection. In addition, corrosion is a problem when such liquids must be contained at high temperatures.

7. In-Core Thermoelectric Reactor

Studies of water-cooled direct-conversion thermoelectric reactor for the Office of Naval Research (AD-267442), carried out by Texas Instruments, have indicated that a typical reactor unit would consist of a 6 ft. by 6 ft. cylindrical core, with UO_2 fuel elements of about 10% U^{235} enrichment separated from the water coolant by about 0.100 inches of PbTe.

The comparison at various thermal cycles shows that the low efficiency of present day thermoelectrics will necessitate the use of natural circulation cooling. The output power is only slightly larger than the pumping power required for a forced circulation system.

The system studied can produce about 4-mw electrical output with about 7% efficiency.

The thermoelectric nuclear fuel element will be expensive to build, costing perhaps five to ten times as much as a standard nuclear fuel element. The reliability of the element must be determined experimentally. The U^{235} inventory is much larger in this system than in a conventional reactor.

At least two important advantages are gained at the expense of the above disadvantages: the reactor power system is silent and there are no moving parts -- therefore, a "promise" of reliability and unattended operation. If a direct-conversion power generation system were perfected with a 15% efficiency, there would certainly be a strong incentive to develop such reactor systems.

At present, development effort is aimed at perfecting techniques for reliable, low-cost fabrication of thermoelectric elements with thicknesses less than 0.050 inch, and perfecting higher temperature and higher efficiency thermoelectric materials.

Studies have shown that the limiting factor in the design of direct-conversion thermoelectric elements is not nuclear design, but the difficulty of element fabrication and the low output power of the reactor.

8. Status of Thermionic Converter Power and Systems

<u>Organization</u>	<u>Reactor</u>	<u>Power Output</u>	<u>Temperature</u>	<u>Specific Weight of System</u>	<u>Prototype Available</u>
<u>Out-of-Core Systems</u>					
Aerojet-General	Reactors in 300 Kwe range - SNAP-50 type	2-3 watts/cm ²	2600-2700°F	--	late 1960's
Ford Instrument	--	3 watts/cm ²	2300°F	--	--
<u>In-Core Systems</u>					
Martin Nuclear	Liquid Metal cooled reactor - 60 Kwe	10 watts/cm ²		--	1968
General Electric	Fast Reactor	10 watts/cm ²	3100-3300°F	10 lb/Kwe	by 1970
Atomics International	Liquid Metal cooled reactor	15-20 watts/cm ²		4-5 lb/Kwe	1973-1976
General Atomics	Fast Reactor	10 watts/cm ²	3000-3300°F	--	--
Babcock & Wilcox and R.C.A.	Boiling Water Reactor	2.5 watts/cm ²	2200-2500°F	--	--

9. Thermionic Conversion Devices

Thermionic devices, although capable of being produced in quantity, have yet to reach practical levels of reliability, lifetime, and efficiency. The present requirement of 2000°C cathode temperature for reasonable efficiency, (about 12%), leads to formidable materials problems. Methods of reducing the cathode temperature without excessive loss of efficiency are being studied at present.

Thermionic devices are technologically similar to electronic vacuum tubes. It is thought that knowledge of vacuum tube techniques and mass production methods will be required to bring device costs to a competitive level. This is based on the assumption that thermionic devices will, like vacuum tubes, be of two types, one rugged, with a 5000 to 10,000-hour life (like present military types), but extremely expensive, and the other having a 1000-hour life, but inexpensive and easily replaced.

The bulk of thermionic research is on the cesium diode, including both low- and high-pressure varieties. Only two companies seem to be developing the closely spaced vacuum diode, and apparently no one is now working on the magnetic triode.

Research on uranium carbide and uranium carbide-zirconium carbide has been initiated at LASL, which has the largest program in this area. The scope of these studies is extensive, including plasma theory, experimental device development, and in-pile radiation testing.

Several organizations are doing in-pile experiments on cesium cells. One is preparing a combined thermionic-thermoelectric unit for in-pile insertion. The emphasis is on applications to space power systems.

RCA has already investigated solar and chemical energy sources for thermionic devices. Working with Thiokol, RCA has built and tested thermionic devices for insertion in rocket motor chambers and nozzles. Such devices would supply 30 kw for control and guidance during powered flight. The 3.5 lb laboratory model produced 270 watts and achieved 8% efficiency operating at 2200°C. At 3200°C, the theoretical efficiency is 38% with a tungsten cathode, but the tungsten evaporation rate is

67 mm/hr. Better cathode materials would allow efficiencies up to 60% for short periods of time (about 10 minutes).

RCA is also working with B&W to develop a nuclear thermionic unit. One such device under test has operated for over 300 hours at close to 10% efficiency. This converter operates at 1350°C and has delivered a total output of 36.5 kwh.

10. Future Development of Thermionic Systems

a. Vacuum diode

Unless a breakthrough in emitter and structural materials is made, it is likely that there will be relatively little progress in the vacuum diode in spite of its attractive characteristics of simplicity and low-temperature operation.

b. The high-pressure ceramic emitter

Using uranium-zirconium carbide fuel element with a nonporous zirconium carbide coating to limit or prevent fission-product diffusion, it is likely that fuel elements of this or equivalent type will demonstrate substantially improved life under actual nuclear conditions.

c. High pressure metal emitter

The most difficult problem areas which will be studied intensively are the engineering aspects of packaging these converters into useful systems. Achieving the proper balance between the strongly interrelated features of efficiency, power output, spacing, and life, is the central area of development. Adequate metal-ceramic cell performance and techniques for the maintenance of metallurgical characteristics of the emitter material will also be significant.

d. Volume ionization devices

Packaging, optimization of cells, and metallurgical problems for the volume ionization converter will be similar to those of the high-pressure metal emitter device discussed above. Since the maximum operating temperature will be 300 to 400°C lower, these problems are expected to be somewhat reduced.

Nuclear systems (in-core) and indirect (out-of-core) cycles have been under study for some time.

e. Converter Performance Trends

No thermionic technique currently under serious development is likely to produce a major increase in efficiency. A slow, gradual improvement based upon a careful pushing of materials to their performance limits seems most likely.

One development could materially alter the situation. Very low work-function surfaces (~ 1.0 ev) have been produced in the laboratory. Those studied to date will probably not be stable under current densities greater than 2 watts/cm^2 which are needed for efficient conversion.

An important parameter affecting thermionic system development is output power density which will probably be chiefly influenced by emitter temperature as follows:

<u>Temperature range - °C</u>	<u>Power Densities watts/cm²</u>
1300	1 - 5
1700	5 - 20
2200+	20+

Thermionic energy conversion has made extremely rapid and substantial progress in a very short time compared to other thermodynamic power cycles. In spite of difficulties inherent in this rapid progress, thermionic power conversion systems are expected to be operational in the near future.

APPENDIX I
POWER CONVERSION SYSTEMS-PRINCIPLES OF OPERATION

1. Dynamic Systems

a. Turbomechanical Systems

(1) System Description

A typical turbomechanical power plant consists of a heat source, a turbine for converting the energy to shaft work, an alternator for converting the shaft work to electrical energy, various condensers and heat exchangers, and controls.

The turbine is used in several types of systems, including the open working cycle and the closed working cycle with single or double fluid loops. In the one-loop direct cycle, the working fluid of the turbines passes directly through the energy source (e.g. reactor core or solar absorber). In the indirect cycle, the energy is transferred to the working fluid through a heat exchanger.

In the open cycle, gases are heated and expanded through a turbine; the exhaust is discharged.

In closed-cycle applications, the problems are such that Rankine vapor cycles seem more useful at present than Brayton gas cycles. The choice between the two basic variants of the Rankine cycle (i.e. with and without superheat), is influenced by the properties of the working fluid and the problem of its expansion from the saturated state which may yield a quality too low for efficient turbine operation because of droplet impingement. Since not all of the heat is added at maximum cycle temperature, superheat must reduce the cycle efficiency. The temperature can be kept more uniform by staging the turbine and reheating. The use of a superheat and reheat cycle requires an additional heat exchanger. Most compact power systems are large enough to require a staged turbine. However, the weight of the heat exchanger may limit the applicability of the superheat or reheat cycle.

(2) Working Fluids

The choice of working fluid for a Rankine Cycle System used in a compact power plant, is an important consideration. Among the

substances that have been suggested for this use are: mercury, rubidium, potassium, sulphur, aluminum bromide and sodium. Certain characteristics of the working fluid are necessary for efficient turbine performance.

To assure that the normal expansion process is along a line of constant quality, it is desirable, for example, to have the slope of the vapor dome nearly infinite. Expansion from the saturated state in this case, results in a high quality at the exhaust, eliminating wet-vapor losses and erosion. If the slope is positive, superheated vapor is produced but some thermodynamic losses occur. The vapor pressure should be low to minimize stresses in the condenser and boiler tubes. Cavitation problems in the pump are possible but under these conditions they can be solved by other means such as the use of jet pumps and inducers. Heat addition across a large temperature difference is associated with losses due to the highly irreversible nature of this process. Therefore, it is desirable that the temperature be raised to boiling with a minimum of heat addition and that the specific heat of the liquid be low. The fluid should be noncorrosive and nonpoisonous, easily handled, storable and relatively inexpensive.

The physical properties and corrosion characteristics of various fluids useful for space compact power applications are currently being investigated.

One criterion for selection of the best working fluid is its boiling point. The following table compares various liquid metals and non-metals. Materials with temperatures higher than about 2000°F are eliminated because of materials creep stress problems.

<u>Liquid Metals</u>		<u>Liquid Non-Metals</u>	
	B.P.-(°R)		B.P.-(°R)
Mercury	1134	Sulphur	1292
Cesium	1720	Phosphorous	996
Rubidium	1730	Aluminum Bromide	950
Sodium	1860		
Lithium	2093		

Each of the possible working fluids can be utilized most effectively only within a certain range of temperatures. Condensate pump cavitation limits, turbine exit stage design, and radiator size determine the lower cycle temperature for each fluid. The upper cycle temperature is limited by the vapor pressure of the fluid and the resulting increase in pressure vessel weight, and corrosion. At 1000°F, for example, mercury has a vapor pressure of 180 psia, while at 1400°F it is 1140 psia. The vapor pressure in a cycle using rubidium at 2000°F is approximately equal to that of one using mercury at 1000°F.

Mercury is a suitable working fluid for power plants with a maximum temperature under 1500°F. The following summarizes the mercury cycle conditions in present space power system designs.

Cycle Conditions of Mercury Power Plants

	<u>Boiling Temp., F</u>	<u>Maximum Mercury Temp., F</u>	<u>Condensing Temp., F</u>	<u>Net Electrical Power, kw</u>
SNAP-1	1025	1300	500	0.5
SNAP-2	900	1150	615	3
SPUD-1	1025	1300	550	1
SNAP-8	1070	1200	706	35
Sunflower 1	1050	1250	600	3

To achieve higher powers, heat must be rejected at higher temperatures and the upper cycle temperatures must be raised. These maximum cycle temperatures are limited by the availability of materials for the turbine and reactor capable of satisfactory operation for long duration.

Developments may permit increases of maximum cycle temperatures to 1800°F, with potassium the most likely working fluid. As the state of the art in the reactor technology and materials developments advances further, a sodium vapor cycle utilizing maximum cycle temperatures over 2000°F will become practical. A very high-temperature sodium vapor cycle will probably be used only for powers in the multimegawatt range and will not be available for operation until after 1970.

(3) Pumps

The Rankine cycle turboelectric system includes a pump which transfers the condensed liquid to the boiler. In double loop systems, pumps are also needed to circulate the primary fluid through the reactor. Dynamic pumps are generally used for Rankine cycle plants; however, positive displacement and electromagnetic pumps may also be used.

Another pump type frequently found in Rankine cycle systems is the jet pump. In compact systems the jet pump is most useful as a device for suppressing cavitation. In principle, this device utilizes the kinetic energy of one jet to accelerate another.

Since the compact system pump is mounted on the same shaft as the turbine, it cannot be operated at its optimum speed; design compromises are made to match the pump to the turbine. This is in contrast to Brayton cycle systems, where the power input to the compressor represents a sizable fraction of the turbine output and the matching problem can only be solved by separately mounting both components. In well-designed Rankine cycle systems, pump power added to the fluid represents but a small portion of the power extracted in the turbine. Component matching can therefore be achieved by selecting pump speed to be similar to the optimum turbine speed.

(4) Development Problems

The mechanical design areas which are of interest are associated with materials and high temperatures. Thermal stress and creep with their effect on clearances, seals and bearings represent the major problems.

The problem of sealing fluids on high-speed shafts has not been solved completely. In conventional plants, leakage is detrimental but the closed-cycle inventory can be replenished. In plants for unattended or remote operation, the inventory loss becomes critical. Because of the limitations on the physical size of the package, there are also problems of contamination of components and interference with engine performance by the leaking fluid.

The rotating unit is usually sealed as a whole, with only electric wires piercing the sealing surface. The whole rotating assembly is subjected to the working fluid vapor and, since it is impossible to have two fluids inside the sealed chamber, the working fluid must be used as the bearing lubricant and as coolant for the alternator rotor.

Many specific types of seals can be used. Labyrinth seals, used in conventional application for many years, have been thoroughly studied. Face seals and shaft seals are also available. Bearings have been run with liquid-metal lubrication.

Another major problem is predicting and maintaining clearances. The changes in size of engine components must be accurately assessed.

For higher powers, the turbomachinery weight becomes a major fraction of the total system weight. The turbomachinery of a megawatt system can comprise about 30% of the system weight. While the turbine weight is only a small part of the total system weight, its speed characteristics have a great effect on the weight of the electrical generator.

To permit electrical generator operation at the high rotational speeds desirable, it is necessary to use multiple rotating units of small diameter to reduce stress limitations or else use generators with large length-to-diameter ratio.

Another problem in development of large power systems is attack by the alkali metal working fluid on alternator windings. Insulation compatible with the alkali metals which will not decrease the performance of the generator must be developed. In these systems, an auxiliary coolant loop is needed to allow the alternator to operate below the minimum cycle temperature.

b. Stirling Engine

The performance of the Stirling engine is dependent upon the working fluid. Hydrogen, with its high volumetric heat capacity, high-heat transfer coefficient, and high gas constant, is a very desirable working fluid. Its use is complicated, however, by the tendency of hydrogen

to diffuse through metals at high temperatures. Although the use of metal coatings (e.g. molybdenum) can greatly reduce this diffusion, the difficulties inherent in extended, unattended life operation, limit the feasibility of hydrogen utilization. The thermodynamic properties of helium (which are nearly as good as those of hydrogen) its availability, and its low diffusion rate, make helium the preferred working fluid.

The generally less desirable thermodynamic characteristics of air eliminate it from consideration as an engine working fluid when high efficiency is required. The following table presents the representation efficiencies of the Stirling engine operating on air, helium, and hydrogen.

Effect of Working Fluid
On Advanced Stirling Cycle Performance

<u>Working Fluid</u>	<u>Estimated Efficiency (%)</u>
Hydrogen	34.5
Helium	30.5
Air	25.5

c. Electric Generators

The effect of variation in important electrical system parameters such as speed, operating temperatures, and input power must be evaluated in terms of total power system performance. The design of a minimum weight a-c generator will not necessarily result in a minimum weight conversion system.

Many design analyses of brushless, non-rotating winding-type generators have been carried out, including the inductor, inductor-Lundell-inductor, Bekey-Robinson and permanent magnet generators.

Studies indicate that a-c generator weights in the range of 0.5 to 0.9 lb/kv-amp are obtainable for electrical generators.

Generators

Generating devices can be divided into seven types:

(1) Conventional d-c generators using brushless and mechanical commutation.

(2) Salient-pole a-c generators using wound d-c fields on the rotor, as used in most industrial a-c generating systems (with or without brushes).

(3) Solid rotor a-c generators in a wide variety of configurations representing various versions of the inductor alternator and/or the Lundell alternator, but all characterized by absence of any rotating windings.

(4) Permanent-magnet generators (PMG) deriving their field excitation from permanent magnets.

(5) Electrostatic generators of various types having in common the characteristic that the conversion process depends on energy stored in electric rather than magnetic fields.

(6) Acyclic or homopolar d-c generators using collectors but no commutator.

(7) Cryogenic generators using any of the foregoing approaches but taking advantage of the unique characteristics associated with very low temperatures.

The first four of these generator types have been subjected to considerable development and application. The last three types listed have not progressed so far, and quantitative information regarding them is limited.

In advanced systems, the surface of the generator will be at high temperatures (1200°F or more) to prevent condensation of metallic vapors. In general, the generator will be operated at the high speeds of the turbine shaft. Problems inherent in high-temperature, high-speed operation greatly influence the choice of generator type.

In modern high-speed generators which use brushes, the attainment of even 500-hour life requires a special atmosphere. The need for long life and the severe environment conditions require that the generator be brushless.

Permanent magnet alternators have received considerable attention. Their specific weights range from 5 to about 1 lb or less per kw, the lower figure representing the higher speed machine (48,000 rpm);

their efficiencies are of the order of 85-90 percent. For operation at very high speeds or high temperatures, the unit is limited by the properties of the permanent magnet rotor. Electrostatic machines may be considerably lighter, running as low as 1/5 lb/kw. However, the practical voltage range is quite high. For a variable capacitance bridge type, operating voltages would be in the 10-15 kv range. For the charge transport type about 1000 kv is the usual output.

Of the many types of electromagnetic generators, the homopolar inductor-alternator has been almost universally selected as the best configuration at power levels above a few kilowatts. This design eliminates rotating windings, sliprings and brushes, and utilizes a solid steel rotating structure capable of withstanding maximum mechanical and thermal stresses. At present, it is felt this machine can be designed to operate reliably with coolant temperatures up to about 1000°F and at speeds to about 24,000 rpm.

The main operational limitation of the inductor generator is the mechanical strength of the rotor magnetic material at the maximum desired speeds and temperature. In general, weight of the generator decreases with increase in speed until such factors as limited rotor diameters or poor armature slot combinations cause the weight to increase.

In addition to mechanical stresses, the Curie point of the magnetic materials limits the maximum operating temperatures. The Curie point of cobalt irons, such as Westinghouse Hiperco 27, is about 1700°F, and that of high-strength carbon steels, such as SAE 4340, is 1300°F.

The use of an alkali metal working fluid introduces an electrical insulation problem. Because of imperfect rotating seals, the interior of the a-c generator is exposed to metallic vapors. The insulation must either withstand the high-temperature environment and corrosive effects of the vapor, or be enclosed within a sealed ceramic container.

Normally, a vapor has a relatively low electrical conductivity. If it is ionized by extremely high temperature or radiation, however, the electrical conductivity of the ionized vapor exceeds that of copper. Once the generator air-gap vapor ionizes, the induced losses increase significantly. The vapor ionization is affected by radiation exposure.

System design must therefore prevent ionization from occurring within the generator.

Generator excitation systems can be classified in two categories -- static and rotary. Studies and analyses indicate that the static exciter is more suitable for use with the inductor generator.

A static excitation system using high-power semiconductors can yield a minimum weight power system.

	<u>Rotary Exciter</u>	<u>Static Exciter</u>
Weight (lb)	32	12
Losses (watts)	1300	700
High-temperature losses	1100°	500°
Low-temperature losses	200°	200°

Static excitation systems must supply generator field power of about 0.5-1.0% of the generator rating. For a one-mw generator, the excitation power is approximately 10 kw. This is a sizable amount of power to be handled by solid-state control devices.

The method and temperature of heat rejection from the exciter must be considered. Semiconductors used in the static exciter are not capable of operating at generator temperatures. At temperatures suitable for silicon rectifiers, the heat rejection area and weight become large. Thus it is necessary to either minimize the power losses or develop high-temperature semiconductors for use in these applications.

Finally, the long-term damage and degradation of these components when exposed to radiation environments must be considered in designing solid-state devices.

A general summary of the present state-of-the-art knowledge of generator characteristics is presented in the following table.

Summary of Generator Characteristics

<u>Generator Type</u>	<u>Polyphase</u>	<u>Operating Speed</u>	<u>Efficiency</u>	<u>Weight</u>	<u>Reliability</u>
Lundell	Yes	Low	Fair	Low	Fair
Homopolar Inductor	Yes	High	Fair	Low	High
Heteropolar Inductor	No	High	Fair	Low	High
Inductor Lundell	Yes	Low	Fair	Low	Fair
Flux Switch	No	High	High	Low	High
Secsyn	Yes	Low	Fair	Low	Fair
Synchronous	Yes	High	Fair	Low	High

Characteristics of Lundell-Induction Generator

<u> Ratings (kw)</u>	<u> Shaft Speed (rpm)</u>	<u> Length (inches)</u>	<u> Diameter (inches)</u>	<u> Weight (lb)</u>	<u> lb/kw</u>	<u> Eff. percent</u>
80	30,000	11	11.25	120	1.5	94
125	26,000	12	12.5	193	1.55	95
275	19,000	15.5	17	468	1.7	96
1000	8,000	28	29	2750	2.75	96.5

SNAP-2 utilizes a permanent magnet generator. The rotor is constructed of Alnico V permanent magnets -- unaffected by the high-temperature mercury environment. The stator-wound field coil is sealed from the mercury environment with a glass seal. The generator is a two-phase unit connected externally to produce a single-phase output.

The SNAP-8 unit has a homopolar inductor generator. A 15-kw generator for use in a solar power system of the homopolar inductor type has also been designed. These units operate in mercury vapor environments.

2. Direct Conversion Systems

a. Thermoelectric Conversion

Among the materials showing promise for use in thermoelectric conversion devices are: lead telluride, bismuth telluride, zinc antimonide, germanium telluride, manganese telluride and cerium sulfide.

The factors to be considered in the design of thermoelectric generators include the temperature of the heat source and sink, the electrical output required, the properties of materials used, and the environmental conditions of operation. At the present time, certain problem areas such as the methods and materials used in electrical connections, require additional development work.

The thermoelectric generator can be designed to operate with any heat source, a solar concentrator, a combustion reaction, a nuclear reactor or radiosotopic source. By-product and waste heat from other processes can also be utilized.

In general, the design of thermoelectric generators for terrestrial applications is somewhat simpler than for space applications since weight considerations are not so critical and heat removal can be accomplished by convection as well as radiation. However, the use of convection cooling may introduce pumping requirements and other complications into an otherwise simple system.

The use of nuclear fission and isotopic decay heat sources is of great interest. At the present time, radiation damage to semiconductor materials limits the length of service of the thermoelectric elements and the allowable proximity of the elements to the nuclear fuel. The future of direct nuclear thermoelectric power generation hinges on the solution of these problems and the development of appropriate radiation resistant materials.

Existing thermoelectric generators are nowhere near being economically competitive with steam turbogenerator units. However, light-weight, inexpensive power sources with lifetimes up to 5 years are being built now. These are low-temperature units, which operate at efficiencies in the range of 6-8% overall, have a materials efficiency of about 17%. Devices with overall efficiencies greater than 10% are in the offing. The output of generators now ranges up to several hundred watts, with kilowatt units due soon. As the development of advanced thermoelectric materials continues, materials efficiencies of the order of 30% are likely. Thus, overall efficiencies greater than 15% may be reached in the near future with 20 to 30% attainable eventually.

The thermoelectric generator is notable because of its versatility as to heat source and its relatively low operating temperatures (up to 1100°F at the present time). It is essentially a direct-current device.

Several thermoelectric generators are proving their usefulness today. SNAP isotopic units are being developed and used in space and terrestrial application such as satellite power plants, buoys, navigation lights, and weather stations. For several years, the Russians have operated radios on thermoelectric power, utilizing a kerosine-fueled heat source.

Economical and efficient thermoelectric devices can be developed to the performance levels which seem theoretically possible -- if progress is made in the development of efficient, high-temperature thermoelectric materials, in the design and fabrication of the devices themselves and in the improvement of heat source utilization and heat removal systems.

Advantages and Disadvantages of Thermoelectric Power Generation

Advantages

Compactness
Quietness
Heat source versatility
Relatively low-temperature operation
No moving parts
No maintenance required
Simplicity of construction
Redundancy of small component parts
Adaptability of various ambient conditions
Long life
Reliability

Disadvantages

Relatively inefficient
Relatively low-temperature heat source
High specific weight
Low voltage
Long time constant
D-C output

b. Thermionic Conversion

There are two types of thermionic generators. The first is the vacuum diode, so called because the inter-electrode space is evacuated. The operating principle of this device is well understood, and experimental units have been built which follow theoretical predictions. When the cathode is heated some of the electrons gain enough energy to overcome the "work function" -- the potential barrier that holds them to the surface of the material. If they also have enough energy to overcome the potential field of the "space charge" due to previously emitted electrons in the inter-electrode space, the electrons eventually reach the anode where they give up a part of their energy -- an amount equivalent to the work function of the anode surface. If this collector potential barrier is small, the energy level of the electrons in the anode is higher than it was in the cathode, and the resulting potential difference causes current to flow in an external circuit.

A major problem of the thermionic generator is posed by the presence of the space-charge field which limits the current flow. In the vacuum diode, this difficulty is overcome by maintaining extremely close electrode spacing. Such spacing -- 0.01 mm between electrodes -- is quite difficult to maintain in large-scale rugged devices. The development of vacuum thermionic generators is proceeding, but alternate means of solving the space-charge problem are being studied. For example, it has been suggested that a magnetic field be used to direct the electrons from cathode to anode in a circular path to avoid the space charge region. Another scheme would utilize a grid between the electrodes to accelerate electrons through the space charge region. The auxiliary electric and magnetic fields required in such schemes consume power, however, reducing the efficiency of the system.

The second type of thermionic generator is the one which has been receiving the most attention recently and appears to offer the most promise for practical power generation. It operates with ionized cesium filling the interelectrode space. Several modes of diode operation are possible depending on the pressure of the cesium, and the device may be called a

cesium diode, a plasma diode or a plasma thermocouple. The basic theory of these devices is not well developed, but units have been built and have demonstrated system feasibility. The negative space charge between electrodes is counteracted by positive cesium ions. If more ionized gas is used than required for charge neutralization, that is, if the gas pressure is increased, the anode and eventually the cathode will become coated with the gas. This lowers the electron work function and improves the performance of the unit. When the gas pressure is great enough, collisions between the charged particles become important. In this mode of operation the unit is called a plasma diode. At still higher gas pressure, the device can be considered a gas-phase thermoelectric generator -- a plasma thermocouple.

Important in the design of the thermionic generator are such parameters as the collector (anode) work function and temperature, the interelectrode spacing, the emitter temperature and the cesium pressure. The cathode temperature, depending on the materials used, will be between 900 and 2500°K (1600-4500°F). Any heat source that produces the appropriate temperature can be used. Systems which utilize solar concentrators have been proposed for terrestrial and space applications. Combustion, catalytic and nuclear heat sources are also under study. One type of nuclear system envisages the incorporation of nuclear fuel into the cathode itself, eliminating the need for any heat transfer fluid. Materials are being developed, and this system seems quite promising for future advanced reactor power unit.

At the present time, laboratory devices are operating at 10 to 15% efficiency. This will probably be increased to 20 to 25% in the next few years, with 35 to 40% theoretically possible. The output of a thermionic diode is about 1 to 2 volts with current densities of 1 to 50 amperes/cm². Some of the prototype devices have operated for hundreds of hours, and this will be extended to thousands of hours in the near future.

There are many problems to be solved before large-scale, efficient and reliable devices become available. The mechanical design of thermionic generators presents a problem. Extremely close electrode spacing must be maintained under the conditions of thermal and mechanical stresses to which

a vacuum diode unit may be subjected. Seals and joints must resist failure in vacuum or in a cesium atmosphere. Severe materials problems include the choice and development of components which minimize cathode evaporation, lower the anode work function, and eliminate corrosion by cesium. Another problem in the engineering design of the thermionic generator is the minimization of heat losses by direct thermal radiation from the high-temperature cathode.

Within a few years it is probable that small and medium-sized thermionic generators will be in use for special applications such as space auxiliary power plants and remote, unattended, terrestrial power units. The thermionic generator has the advantage of being compact, silent and without moving mechanical parts, and offers the possibility of long, maintenance-free operation. It is appropriate for use with high-temperature heat sources and may, therefore, eventually find a place as a topping unit for a vapor cycle plant. The high-temperature thermionic generator might be used in combination with the lower-temperature thermoelectric generator as a wide temperature range, small or medium-sized power generator. An advantage of the thermionic diode yet to be explored, is its possible use for a-c power generation. The electron current produced could be modulated in the generator itself, by varying an external magnetic field or the potential on a grid between the electrodes.

(1) Thermionic A-C Power Generator

A thermionic cell able to generate alternating current has been under test for the past six months at Ford Instrument Co. This unit does not need inverters to change direct current to alternating current. The cell operates at about 1250°C, a relatively low temperature compared with the usual 1800°C of most other thermionic units. The low-temperature operation of the cell permits a voltage pulse to readily extinguish as well as fire the unit.

Except for a pulsing circuit, the a-c cell is identical with the low-temperature, wide gap d-c cesium cell. A low-temperature d-c cell has run continuously for nearly 3600 hours and is putting out

nearly one watt of power at a power density of about 3 watts/cm².

When d-c thermionic cells are used, several units must be strung in a series to generate the two or three volts needed to energize an inverter. Loss of a single cell could significantly decrease performance of a thermionic converter system. Low-voltage inverters are today notoriously inefficient devices. They are heavy and must operate at relatively low temperatures. If, however, a pulsing current can be generated within the thermionic cell, the a-c component could be taken off at the desired voltage by a single-turn primary transformer.

A voltage pulse can be used to fire a converter but it is difficult to extinguish the converter with a second pulse. However, because this unit operates at a relatively low temperature, nearly all the positive ions produced are a result of electron collision. The device is therefore analogous to a gas-discharge device such as a thyratron, and has similar properties: momentarily increasing the applied voltage causes the cell to conduct, while momentarily decreasing the applied voltage causes the cell to stop conducting. High-temperature thermionic cells, on the other hand, produce positive ions by surface ionization and internal switching by an applied pulse is difficult. This cell uses about 10% to 20% of its power output for pulsing, but there is hope that this amount can be reduced. The present cell shows its best efficiency at 400 cps output. The device has been run up to 1700 cps.

Operation of the low-temperature converter is based on setting up an ultra-low-voltage arc between the emitter and collector. A gap up to 0.2-in. between emitter and collector is used. This gap is one or two orders of magnitude large than those used in most other thermionic cells.

The a-c thermionic cell, based on the phenomenon of arc discharge in a cesium vapor, has two modes of operation -- an arc mode and a secondary mode. The arc mode is characterized by a high current density and a correspondingly low internal resistance due to the formation of cesium ions that neutralize the internal space charge. The secondary mode is characterized by very low attainable currents, due to an appreciable

space charge that results in a high internal resistance.

The a-c cell operating into a load that does not approach an open-circuit condition can be pulsed into and out of the arc mode of operation through the application of momentary pulses of positive and negative voltages. A converter pulsed in this way would deliver a pulsating d-c voltage across its transformer, since the d-c components of current will cancel and prevent saturation of the transformer core. The transformer secondary may be wound for any desired voltage, thereby permitting the generation of usable a-c voltages from a pair of thermionic converters. The two converters may be combined into a single-cell using a common emitter for both collectors. This will result in a more efficient design since standby and end losses will be reduced.

APPENDIX J
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