

Power Generation from Nuclear Reactors in Aerospace Applications

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POWER GENERATION FROM NUCLEAR REACTORS IN AEROSPACE APPLICATIONS

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SUMMARY

This survey paper focusses on power generation in nuclear powerplants in space. In particular, the states of technology of the principal competitive concepts for power generation are assessed. The possible impact of power conditioning on power generation is also discussed. For aircraft nuclear propulsion, the suitability of various technologies is cursorily assessed for flight in the Earth's atmosphere; a program path is suggested to ease the conditions of first use of aircraft nuclear propulsion.

INTRODUCTION

Although this Symposium on Advanced Compact Reactor Systems principally emphasizes just nuclear reactors (as well as their potential applications, competitive concepts for their design, their safety, their shielding and their means for cooling), equal emphasis is warranted on the means for converting reactor heat into a useful product, the useful product generally being electric power. In "Radioisotope Thermal Generators and Thermoelectric Conversion" (ref. 1), Stapfer does treat the history, current status and potential of thermoelectric power generation in space. The organizers of this symposium assigned all other forms of power generation to this paper. Power conditioning for all these generators was also to be assessed. And the potential applicability of this space - power technology to aircraft nuclear propulsion was to be considered as well.

Given the enormous scope, I chose to emphasize power generation in space. My comments on power conditioning are limited to the influence that power conditioning exerts on the power generator itself. For aircraft nuclear propulsion, the service conditions are so different from those in space that very different technology is required. The influences of required power level, of required reactor life, of atmospheric oxygen and of nuclear safety are briefly discussed. A rationale is advanced for considering the air-cushion (or ground-effect) vehicle as the progenitor of nuclear-powered aircraft.

The methods considered for power generation are thermoelectric, Brayton cycle, Rankine cycle, Stirling cycle, magnetohydrodynamic (MHD), and thermionic.

Program Philosophy

The philosophy for managing the program to create a long-lived reactor power system stems from the characteristics inherent in the powerplant itself. First, development of such a nuclear powerplant is both costly and lengthy. Several hundred million to perhaps a billion dollars are required, and a period of at least 10 years. Second, the desired lifetime sought for the power system is itself of the order of 10 years. Although a lifetime of

perhaps five years might be acceptable for a few initial applications, many other applications will require the full 10 years. Design of the powerplant for the longer useful life would save both time and money.

"Optimization" of one powerplant design for, say, a five-year life followed by "optimization" of a second powerplant design for, say, a 10-year life would require a succession of developments and demonstrations. Not only would the mission for the second, long-lived powerplant be delayed until the appropriate development and demonstration had been completed but also about \$100 million would also be required. (On p. 285, ref. 2 gives \$50 to 60 million.) At this time, planning for successive development of two such powerplants thus seems extravagant in its use of both time and money. Instead, a single design fulfilling a range of application appears more prudent. If, following development of a multipurpose powerplant, a particular mission justifies development of its own "optimized" powerplant, the powerplant could then be developed specifically for that need, the mission then, of course, bearing the budgetary burden for development of that "optimized" powerplant.

In supporting this philosophy of the general rather than specific need for nuclear power in space, this paper emphasizes "enabling technology." Thus, much of the technology described is directed toward solving problems, toward defining potentialities of various concepts and toward providing a design basis for future power systems.

A second consequence of the required powerplant lifetime of roughly 10 years is that a life demonstration prior to use in space is impractical. Instead, performance could be demonstrated for a year or two. If that were sufficient assurance for the mission manager, useful application in space could follow that brief test. Testing of that demonstration powerplant in the terrestrial laboratory could then continue at modest cost. The true endurance of the powerplant would probably be demonstrated only through in-service operation in space.

This raises the critical question: How can we estimate the powerplant's life before its useful application in space? Only in that way can we confidently avoid wasting a great deal of money. The crucial factor is to have available good backgrounds of constituent and component technologies that will provide a basis for life estimation. In addition, the approach to powerplant design should be conservative, in the light of these constituent and component technologies. This approach would give an air of validity to the life estimation and provide some substantive assurance that the required powerplant life can in fact be achieved.

This philosophy has markedly affected the content of this paper. Accordingly, great stress is placed on the following factors: (1) the body of constituent and component technologies, especially as they concern durability and life estimation, (2) recognition of and allowance for the range (or scatter) of the available data, and (3) conservatism in applying these data.

A truly crucial factor in useful application of nuclear power in space is its acceptance by the mission managers. Xenophobia itself is important. A trouble-free development program that holds to its schedule, its budget and its advertised performance should help to allay those fears; achieving that

depends on, as above, the prior evolution of an extensive, sound body of enabling technologies as well as on conservatism in utilizing that information.

Just as beauty lies in the eye of the beholder, so does mission suitability reside in the mind of the mission manager. The mission manager's congenital distrust of things nuclear will be allayed by a successful ground-test of an exact duplicate of the powerplant to be flown; independent tests of reactor and power-generating system might be acceptable, depending on the individual bent of the mission manager. A test of reactor and power generator combined is not only more costly but also more difficult and time consuming. But convincing the mission manager of the suitability of the nuclear powerplant may require just such a combined test. At this time, a prudently planned program would include such a combined test; if not needed, it could be deleted later.

Ground-testing an exact duplicate of the powerplant for use in space will also more likely convince the mission manager than would a powerplant specially adapted to testing on Earth. The suitability for ground-testing an exact copy of the flight article might, therefore, become a design specification for the powerplant. For example, the poor pumping characteristics of heat pipes may not allow the ground test to duplicate the flight powerplant; the one-g environment in the ground test might require either a planar version of the powerplant or gravity-assist for liquid flow. In either case, that compromised ground-test may influence the mission manager's assessment of heat pipes versus pumped loops.

An additional facet of ground testing a nuclear space power system is test interruption. These interruptions occur for a variety of reasons, and the powerplant design must accommodate them as well as the conditions in space. Some interruptions will be accidental; examples are shutdowns from loss of utility-supplied electric power or from failure or deterioration of test-supporting equipment. Other shutdowns are programmatic; during, say, a two-year endurance run, planned shutdowns and inspections might occur at, say 3, 6 and 12 months into the test. Such inspections could early detect incipient problems and permit corrective actions, even redesign and redevelopment of a troublesome component. Additional shutdowns may also occur for technical reasons, that is, as results of failure or malfunction of system components.

The vicissitudes of endurance testing of nuclear powerplants lead prudent project managers to acknowledge such test interruptions and to design the powerplant itself to tolerate these interruptions. In some ways this need for multiple safe shutdowns and restarts imposes conditions more severe than those in space; xenon override for reactor restart is an example. But in order for a ground test to be successful in the mission manager's eyes, the powerplant should tolerate multiple restarts, perhaps 10 or 20 in number.

This need to tolerate multiple stops and starts will affect both powerplant design and the assessment of competitive concepts. For example, welded assemblies must remain ductile even after a considerable period of service at operating temperature. Gas bearings, if used, should endure multiple starts and stops. Stable materials will likely be preferred over those unstable; for example, substantial grain growth at operating temperatures would raise questions concerning basic stability of the materials of construction and thus of their longevity. For concepts (such as heat pipes or Rankine-cycle systems) using two-phase fluids, the location of condensed liquid (or frozen solid)

following shutdown could alternatively permit or prevent restarting the powerplant. The need for stop and restart during ground testing imposes such a variety of constraints that each powerplant concept will require substantial evaluation on this score.

An additional factor, whose assessment is also for the future, is ability of the various powerplant concepts to tolerate acceleration of the spacecraft. For example, spacecraft acceleration in an inappropriate direction might produce liquid flow along heat pipes and out of the reactor, a condition impairing ability of the heat pipes to cool the reactor. Bearings must also tolerate spacecraft acceleration. Intolerance of these accelerations would automatically exclude some nuclear powerplants from, say, military spacecraft requiring agility or from scientific spacecraft to be boosted by the Shuttle-Centaur orbit-transfer vehicle. Not only do the mission managers distrust things nuclear but the view is also widespread that rotating machinery is, by its very nature, limited in its potential lifetime. Accordingly, a program relying on rotating machinery must frontally address this issue and stress both conservative design and durability demonstration of the rotating machinery. In assessing the current technology for rotating machinery, this paper similarly emphasizes design conservatism and demonstrated durability.

Specific Versus General Mission Needs

So far no mission group or agency has seen fit to specify a firm need for nuclear power in space. Although such a stipulation of a definite need for reactor power would give firm direction, urgency and purpose to the reactor-power program, that would at present be a mixed blessing. Even in the absence of direct monetary support by a mission group, the reactor-power program would likely flourish as a result of increased project funding from Congress, given merely the verbal support and urgency contributed by the mission group. On the other hand, every reactor-power concept faces questions of feasibility yet unanswered. A crash program with a fixed completion date could waste a lot of money.

The mission needs are still multiple rather than singular and diverse rather than unique. Inasmuch as the first useful application of reactor power in space is in the indefinite future, time is available to us for finding solutions to the outstanding problems in the most effective and most economical manner. This time might also allow us to choose a system concept having a broad range of application as well as the capacity for evolution to both much higher powers and higher performance. Thus the opportunity confronts us to address a range of applications and to create the capacity for utilizing nuclear power on a broad front in space.

This paper, therefore, emphasizes the enabling technologies to make practical several types of nuclear power-generating system. A program of constituent and component enabling technologies could provide both the information and the assurance that, once a firm mission need arises, a given nuclear-power system can be developed on time, on budget and with the performance and life claimed for it. Such a program of enabling technology in advance of a firm mission demand for reactor power could solve the principal problems before development begins. Not only would the enabling technology reduce development risk but also development cost would be reduced, the schedule could be accelerated and

performance improved. Too-early emphasis on powerplant demonstration may lead to poor realized performance as well as overruns of both time and money.

The hazards from inadequately demonstrated enabling technology before system construction are illustrated by the Department of Energy's Isabelle particle-accelerator project at Brookhaven National Laboratory (ref. 3). For Isabelle, 1100 high-performance superconducting magnets were crucial. Mark 5, the precursor of the Isabelle magnets, achieved 4.8 teslas in the spring of 1976. The Isabelle magnets were to be about 10% larger in physical size than Mark 5 and to operate at a magnetic induction of 5 teslas. No prototype magnets were built and tested before facility construction began in October 1978. No design margins were established for the magnets, and manufacturing variations from magnet to magnet were not taken into account. Concurrent with facility construction, Westinghouse under contract built coils for 12 magnets. Using these coils, Brookhaven assembled magnets and tested them, achieving 4 teslas instead of the necessary 5. In spite of a crash program on technology for superconducting magnets, problems continued until 1981, when a new design was evolved. Although the problems in magnet design have apparently now been solved, completion of the facility has been delayed by four years. Expenditures to date exceed \$160 million, and the total estimated cost for the facility rose from \$275 to \$500 million.

The risk inherent in advancing the technology of superconducting magnets is broadly recognized. For example, Vineyard states (ref. 3) "In any state-of-the-art endeavor on the scale of Isabelle, unforeseen technical problems will arise... We are pushing the state of the art...." Lederman states, "It is certainly true that superconducting accelerator magnet technology has proved to be enormously difficult.... The risks are high...." Drell writes, "(S)uperconducting magnet technology is proving to be more unforgiving than had been realized...."

Drell also writes, "(T)he Isabelle problems are of a technical nature, and are therefore not amenable to solution by management....changes alone...." With this I disagree. From a management viewpoint, the issue is how to avoid risking a large amount of money when there is substantial technical risk. The solution is simple: Solve the technical problems before spending the large amount of money.

In another DOE program on superconducting magnets, Fermi National Accelerator Laboratory (Fermilab) took that very path. According to Lederman (ref. 3), Fermilab built and evaluated 100 7-meter-long superconducting magnets in order to learn how to mass produce such magnets; parts for an additional 60 magnets were also manufactured. The cost of these magnets was about \$6.5 million, about 1.3% of the current price of Isabelle. For Isabelle, a similar program to build and evaluate magnets would have been a very cost-effective predecessor of facility construction.

So should it be with reactor space power. In that program, the enabling technology should be evolved and the issues of substantial risk all settled before a costly system demonstration or space flight are begun. For this very reason, this paper stresses adequate enabling technology in advance of powerplant demonstration or development, setting margins in powerplant design and using the available data in a conservative manner. Our fortuitous lack of a firm mission requirement provides the time for evolving the enabling tech-

nology which is a necessary precursor if the development program is to have low risk.

THERMOELECTRIC POWER GENERATION

Thermoelectric power generation is widely used commercially as a pilot-flame detector in gas-fired furnaces. In addition, thermoelectric power generation for use in space in combination with nuclear heat sources has been under active investigation for 25 years, more money having been spent on thermoelectrics than on any other form of space power generation. This large, continuing effort principally devolves from the fact that thermoelectrics are the only form of nuclear power generation so far used by the USA in space. The radioisotope thermoelectric generators (RTG's) have been used successfully in a variety of space missions, two examples of which are discussed below.

The Apollo missions placed six RTG's on the lunar surface in the Apollo Lunar Surface Experiment Packages (ALSEP). These SNAP-27 RTG's used the radio-nuclide Pu-238 as their heat source and produced about 63 W of electric power (ref. 4). Four of the six RTG's operated successfully on the lunar surface for periods ranging from 5 to 8 years until the experiments were terminated on command in 1977. A fifth RTG failed in 45 days (ref. 5), and a sixth operated intermittently (ref. 6).

The Multi-Hundred Watt (MHW) RTGs were flown on several spacecraft. Two were used by each Lincoln Experimental Satellite LES-8 and -9 in 1967, and three flew on Voyager 1 and three more on Voyager 2 in 1977. All 10 RTGs operated successfully (ref. 4). The thermoelectric generators used SiGe, just as contemplated for the SP-100 reactor powerplant. When this thermoelectric material was found early in the MHW program to be unstable at 1100° C, the peak operating temperature was reduced to 1000° C. At this temperature, generator output declined from its initial value of 150 W (ref. 1) to 125 W (ref. 4) after 5 years, corresponding to an average degradation rate of 3.65 percent a year. With its half-life of 89 years, the Pu-238 heat source contributed 0.78 percent to this annual decline, the remaining 2.87 percent resulting from instability of the SiGe at 1000° C.

The SNAP-10A reactor powerplant (figs. 1 and 2) also used SiGe thermoelectric power generation, in this case at 530° C (990° F) rather than 1000° C. From 43 kW of heat, this powerplant produced 56 W of power for an overall efficiency of 1.3% (ref. 7). Two such powerplants were built and tested. One operated successfully in the laboratory for 10,000 hr. The second powerplant was launched on a test flight on April 3, 1965. Unfortunately SNAP-10A ceased operating after only 43 days in space, apparently through failure of its voltage regulator. When the unregulated voltage climbed above its tolerable range, the powerplant's control system signaled the reactor to shed its reflector, which it dutifully did. That shut down the reactor and powerplant and terminated the experiment.

Figure 3 illustrates the performance potential of several classes of thermoelectric material. In each case, the area under the curve for a selected span of temperatures is proportional to efficiency of power generation. For SiGe, the technology is from the MHW program, its current temperature limit being 1270 K. Addition of GaP is predicted to boost efficiency somewhat, and peak operating temperature could conceivably be raised by 100° (ref. 1). Entirely

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CORRIGENDUM concerning page 3, para. 3

Only five RTGs were taken to the Moon, rather than six. For Apollo 11, the Surface Experiments Package was powered by solar cells, not an RTG, and operated for only 20 days (ref. 116, p. 2-3). The five RTG-powered experiments operated for periods from 1753 to 2872 days until they were shut down on September 30, 1977. For Apollo 14, the RTG power source operated successfully for four years but then operated only, intermittently for the succeeding 2.5 years. For Apollos 12 and 15, power output degraded more rapidly than for the other RTGs, the degradation rates being 7 and 12 percent a year, respectively (ref. 116, p. 4-42 ff). If we consider the harsh thermal environment on the lunar surface, even these degradation rates were comparatively successful, for the degradation rate of SNAP-19 on Nimbus III was 40 percent a year, its generated power dropping from 56.4 watts to less than 20 watts in 33 months (ref. 117).

ADDITIONAL REFERENCES

116. J.R. Bates, W.W. Lauderdale, and H. Fernaghan: ALSEP Termination Report. NASA RP-1036, April 1979.

117. G.L. Bennett, J.J. Lombardo and B.J. Roci: U.S. Radioisotope Thermoelectric Generator Space Operating Experience (June 1961 - December 1982). 18th. Intersociety Energy Conversion Engineering Conference, AIChE, 1982.

new classes of thermoelectric materials might raise both conversion efficiency and the peak temperature, but actual demonstration of such gains must await the outcome of the current technology program. The reactor temperature limit is that specified by Los Alamos National Laboratory (LANL, ref. 8).

The current concept for SP-100 would operate the reactor at about its temperature limit of 1500 K. Heat pipes from the reactor would thermally radiate to the thermoelectric converters operating at their temperature limit of the order of 1300 K. Two key advantages are gained by this approach: (1) the design and performance of the thermoelectric generators are independent of the reactor. And (2) absence of a high-temperature electrical insulator permits generation of voltages that markedly reduce the mass of the power conditioner and, concomitantly, the radiator area required to cool the power conditioner (see POWER CONDITIONING).

The performance potential of the SiGe thermoelectric material is shown in figure 4; efficiency of power generation was taken as 17% of Carnot cycle efficiency. Radiator area and efficiency are two measures of merit for reactor power generators. Not only would a large radiator be heavy but large size, per se, is disadvantageous. And efficiency of power generation measures how effectively a given heat source would be utilized.

Radiator area per kWe has a minimum for the following reason: Lowering radiator temperature raises efficiency of power generation and thus lowers the rate at which heat must be rejected. On the other hand, less heat can be rejected per unit area at the lower temperature. At the radiator temperature for which these two factors are in balance, radiator area is a minimum. If powerplant efficiency is a fixed fraction F of Carnot-cycle efficiency, then the temperature ratio producing the minimum radiator area is as follows:

$$\frac{T_{\min}}{T_{\max}} = 1 - \frac{5}{8F} + \sqrt{\left(1 - \frac{5}{8F}\right)^2 + \frac{1-F}{F}}$$

For $0 \leq F \leq 1$, the corresponding values of temperature ratio range from 0.75 - 0.8.

In the SP-100 program for reactor space power, thermoelectric power generation is the chosen concept. This principally results from successful use of SiGe thermoelectric power generation in the RTGs. The operating temperature for the RTGs was limited to 1000° C, or 1270 K in figure 3. The degradation rate of 0.0287 y^{-1} experienced at this temperature, if continued for 10 years, would result in a 25% power loss, that is, a 100 kWe power system would degrade to 75 kWe over 10 years. Addition of GaP to SiGe is currently being explored as a means of reducing or eliminating this power loss.

The mass of the SP-100 powerplant is currently 3900 kg (ref. 1). Correspondingly, specific mass is 39 kg/kWe at beginning of a mission and 52 kg/kWe after 10 years, 25% degradation (as above) being taken into account. Technology yet to be evolved may reduce these masses.

The overall assessment of thermoelectric power generation in figure 4 can itself be appraised by considering the performance resulting from the current-

ly nominal design conditions for SP-100, as follows: hot-junction temperature, 1300 K and cold-junction temperature, 600 K. Power generation at 0.17 of Carnot efficiency and power conditioning at 0.98 efficiency produce an overall efficiency of 0.09, a high value for thermoelectric power generation. This high efficiency is achieved by accepting a large radiator area, 1.9 m²/kWe in this case. Because this radiator area is more than 3 times the minimum, some emphasis herein is placed on smaller areas and, concomitantly, on accepting lower efficiencies. For radiator areas of 0.67 and 0.79 m²/kWe, overall efficiencies of 0.05 and 0.06, respectively, result, these radiator areas being 15 to 35% greater than the minimum.

Inasmuch as such a thermoelectric power system would have a high multiplicity of power-generating elements (220,000 elements in ref. 1), tolerance of failure is generally claimed. On the other hand, the manner in which this will be achieved in SP-100 has yet to be delineated. The history of single-point failures in use of thermoelectric power generation makes this delineation critical.

BRAYTON CYCLE

General Background

The Brayton cycle is a conventional, closed gas turbine (fig. 5) like that earlier proposed for terrestrial use with the high-temperature gas-cooled reactor HTGR. Through use of a highly effective recuperating heat exchanger, heat at the turbine discharge that would otherwise be wasted is recovered and added to the compressed gas at the compressor discharge. The cycle working fluid is generally taken to be a mixture of helium and xenon blended to achieve the average molecular weight best suiting the turbomachinery. Instead of a single gas such as argon, the blending of helium and xenon increases thermal conductivity and thereby improves heat transfer.

In figure 6 Brayton-cycle and thermoelectric power generation are compared for the peak temperature of 1300 K. On the basis of radiator area alone, Brayton and thermoelectric power generation are competitive. On the other hand, the Brayton cycle has overall efficiency about four times that of the thermoelectric concept. Thus from a given nuclear reactor, about four times as much power could be produced by Brayton.

For an uninhabited spacecraft requiring about 100 kWe, the reactor and shield, if at the end of a boom of modest length, would together have a mass of about two tons. For inhabited spacecraft, their combined mass will be much greater because of the greater shielding required. The shield weights in figure 7 are probably at the upper end of the range of shielding required because of operating conditions specified in the shield design; inasmuch as that reactor (SNAP-8) was cooled by NaK, the coolant was very radioactive. For this reason, the useful heat was extracted from the reactor coolant in a gallery located within the shield, a factor adding to shield diameter and weight. The maximum shield weight shown of 50 metric tons would be the dominant mass in the powerplant. Thus, a power-generating system of high efficiency would in that case have not only the highest power output but also the lowest specific mass (kg/kWe).

A second, often overlooked factor favoring high efficiency is that each nuclear reactor is limited in its ability to produce energy. The very smallest reactors are critically limited. As energy is produced, uranium is consumed and neutron poisons (fission products) accumulate. The reactor's controls can tolerate a limited loss in reactivity from consumption of uranium and from accumulating neutron poisons, and when this limit is reached, the reactor's useful life is over. Rather than die suddenly at that time, the reactor, if operated beyond its useful life, would die gracefully, gradually declining in its heat generation and in its operating temperature. Accordingly, power output would decline.

Larger reactors are burnup limited and fail structurally if operated beyond their useful lives. This damage results from several causes. Energetic neutrons born in fission damage the atomic structure of the fuel clad and thereby embrittle it. The fissioning itself produces two atoms from one, the two atoms occupying more space than their common ancestor. The product atoms (the fission fragments) depart the site of fission at high speed, leaving behind a vacancy in the fuel lattice as well as coming to rest at random locations within the lattice and producing dislocations there. In addition, roughly one fission in four produces an atom of inert gas (Xe or Kr). To the extent that these gas atoms exist as single atoms within the fuel lattice, they produce no more swelling than any other fission product. On the other hand, these gas atoms also diffuse within the fuel lattice and eventually congregate as gas bubbles; these bubbles can produce fuel swelling both rapid and large. Gas escaping the fuel itself no longer leads to fuel swelling but would exert hydrostatic pressure on the inner surface of any closed fuel container. The current reactor concept for SP-100 avoids this last problem by having a vent that permits this gas to escape into space. Some reactor fuel as well as volatile fission products (iodine, for example) will likely also escape through this vent.

For each burnup-limited reactor, the extent to which the embrittled fuel-clad can tolerate mechanical deformation will dictate the reactor's useful life. For these reactors, this life limit is reached not after a fixed period of time but after a given amount of energy has been produced.

Thus, each reactor (whether criticality-limited or burnup-limited) can produce a given amount of energy during its life; this limited energy product is, of course, the reactor's thermal power multiplied by its operating time. Thus, high efficiency of power generation can extend reactor life.

A Brayton power-generating system, having four times the efficiency of a thermoelectric system, would require only one-fourth the thermal power required by the thermoelectric concept, the generated electric power being the same in the two cases. Overall, the nuclear reactor's limited energy capacity will restrict each class of powerplant to generating a given (but differing) amount of electric energy (kWh) over its useful life. The number of kilowatt hours generated by a Brayton powerplant will be four times that of a thermoelectric because of its higher efficiency. This advantage could be exploited by the Brayton concept to produce four times the power at the same reactor life, four times the reactor life at the same power, or any combination of higher power and longer life producing four times the energy production of the thermoelectric concept.

Although the Brayton concept has great potential advantage in relation to the thermoelectric concept, critical issues are whether or not this performance is actually achievable and whether or not the required rotating machinery would have adequate life. Fortunately, gas turbines are widely used for aircraft propulsion and for power generation here on Earth, and much of their technology is directly applicable to space power. In addition, considerable relevant technology was evolved under the technology program on nuclear space power ten or more years ago. A review of those technologies is, therefore, appropriate.

Although the great body of current, ongoing technology for Brayton powerplants was evolved with airbreathing fuel-burning gas turbines, this technology is generally applicable to turbomachinery operating with any gas. This broad applicability is illustrated by an example; in the 1950's NASA Lewis applied this technology to aerodynamic design of the compressors in the AEC's gaseous-diffusion uranium-enrichment plants, the design techniques evolved in air being directly applicable to uranium hexafluoride in spite of the sonic speed of 95 meters per second for uranium hexafluoride instead of 340 meters per second for air. This technology is summarized in archival publications such as references 9 and 10 as well as in more detailed, specific documents.

The usual size at which this research is conventionally conducted is illustrated by figure 8, this compressor requiring about 10 MW to drive it under its design conditions; the person gives the scale of the photo. In figure 9, the large compressor for a wind tunnel requires 100 MW of drive power; again note the person's relative size. Each of these two compressors had a peak efficiency exceeding 0.90. Gas turbines in central-station service commonly produce 10 to 100 MW of electric power; gas turbines in baseload service have turbine inlet temperatures to 1500 K (2240° F, ref. 11). A great deal of directly relevant technology is thus available for design of Brayton-cycle space powerplants in the megawatt range and above. In fact, the Brayton-cycle gas turbines contemplated for use in the HTGR direct-cycle ranged from 275 to 367 MWe apiece. In addition, there is a large existing reservoir of skills and facilities should any Brayton-cycle space-power program run into trouble. Much of this knowledge resides in industrial organizations accustomed to designing and building nuclear powerplants. In a future program on Brayton space power, competition among several competent industrial organizations is thus assured, and an industrial base for either ongoing military or commercial applications could readily be established, a critical factor in Brayton's competition with alternate concepts.

Brayton at 10 kWe and 1150 K

In the space-power program, the critical issues were the applicability of these well-known principles at much, much lower powers and the general level of performance attainable. Both radial-flow and axial-flow compressors and turbines were explored (figs. 10 and 11); again the person gives scale to the photos. The turbocompressor in figure 12 incorporated a radial-flow compressor and a radial-flow turbine, both supported by gas bearings. The turboalternator in figure 13 had an axial-flow turbine and a synchronous alternator as well as gas bearings. Each of these was a component of a contemplated 10 kWe power system and performed successfully under test (See, for example, refs. 12 to 15). The turboalternator itself was rated at 12 kWe (ref. 14) although conservative design of the alternator permitted generation of up to

33 kWe within the alternator's temperature limit (ref. 15). At the highest powers, axial thrust imposed on the thrust bearing exceeded 200% of the design value; the thrust bearing supported this overload while still keeping bearing clearances within an acceptable range.

In this early program, the performances of the radial-flow compressor and turbine were so good (ref. 16) that that class was selected for further exploration at even smaller sizes (figs. 14 and 15). Component efficiency fell only a modest amount (fig. 16 and ref. 16) in spite of the size reduction.

On the basis of this exploratory research, a 10 kWe Brayton space-power system was designed and built as an experiment to explore the performance and the technology for such a power system. The powerplant's design conditions are as follows:

Working Gas	He and Xe Mixture
Gas Mixture Molecular Weight	83.8
Turbine Inlet Temperature	1600° F
Compressor Inlet Temperature	80° F
Rotor Speed	36,000 rpm
Compressor Pressure Ratio	1.90
Turbine Pressure Ratio	1.75
Recuperator Effectiveness	0.95
Waste Heat Exchanger Effectiveness	0.95
Rated Net Power	10 kWe

The turbine-inlet temperature of 1140 K (1600° F) was so chosen that the powerplant could be made of superalloys rather than refractory metals and so that the powerplant could be tested in air, if that were desired.

At this low power level, the heat source was assumed to be either a radio-nuclide or a solar mirror. In contrast with a nuclear reactor, these heat sources vary in size and mass almost in direct proportion to the thermal power demanded. For this reason, a powerplant overall efficiency of 0.25 was set as a programmatic goal, an 8% heat loss being assumed for a radionuclide heat source; thus, 0.27 was required of the power-generating system, five times the value achieved by thermoelectric conversion in the MHW RTGs.

A radial-flow compressor and turbine of 4.25- and 5-inch diameters, respectively, were incorporated into the so-called "Brayton Rotating Unit," or BRU (fig. 17). As shown by the schematic diagram in figure 18, this BRU had a compressor, turbine, and alternator on a common shaft rotating 36,000 rpm. In order that no centripetal loads would be imposed on the alternator's organic (plastic) insulation, the modified Lundell alternator selected has electrical conductors in the stator but not the rotor. The alternator rotor is a monolithic assembly of 4340 (magnetic) steel and Inconel 718 (non-magnetic) superalloy. The four-pole alternator generated 1200-Hz 3-phase ac power at 120/208 V. A voltage regulator and exciter (a single device) rectified and regulated a small fraction of alternator output, the dc being supplied to the alternator's field windings. Instead of a throttle valve for speed regulation, the powerplant employed a parasitic load to which unneeded power was diverted. The power output was thus regulated in both frequency and voltage, and the voltage-regulator exciter automatically shared reactive loads among power generators operated in parallel (ref. 17).

Early testing of the BRU revealed alternator hot-spot temperatures in the stator of 213° C (415° F, ref. 18) and 257° C (495° F, ref. 19) instead of the specified 180° C (356° F). For this reason, a new stator was designed and built for improved cooling. This redesign of the alternator as well as powerplant testing were so successful that powerplant rating was raised from 10 to 15 kWe, a change leading to some confusion concerning publications about this program.

The BRU's rotor was supported by journal bearings and a double-acting thrust bearing lubricated by the powerplant's inert-gas working fluid. The powerplant was, therefore, hermetically sealed, and no oil was present to possibly contaminate the cycle working fluid. The concept for the journal bearings is shown in figure 19; three such bearing pads constitute a single bearing. Because normal running clearance between shaft and pad (or shoe) is about 0.2 mil (5 μ m), each pad is loaded by a spring that presses the pad toward the shaft. During normal operation, hydrodynamic forces generated within the gas film support the shoe off the shaft and thereby flex the spring. In this way, bearing clearance is maintained within an appropriate range despite differential expansions between shaft and bearing that exceed this clearance. Thus a gas film everywhere separated the moving rotor from its stator so no rubbing or wear occurred.

When the rotor was at rest, the springs clamped the shoes against the shaft. During powerplant startup, high-pressure gas was injected through the pads' pivots in order that hydrostatic pressure would lift the pads off the shaft. After sufficient rotational speed was achieved, the gas injection was stopped. This strategy prevented bearing wear and thereby eliminated that possible wearout mode. Because bearings of this design are somewhat complicated, they have been superseded by foil gas bearings (discussed below), but the Brayton system performance and endurance to be described were obtained with this pad type of bearing.

The recuperating heat exchanger had a compact core with a large amount of extended surface (fins) for improved heat transfer (fig. 20). With the plate-and-fin construction, the hot and cold gases counterflowed in alternate layers of the multilayer core. At the edge of each flow channel, a header bar sealed the channel. The entire heat exchanger was brazed into a monolithic assembly.

The concept labeled "BHXU" was used in the tests to be described, the brazed header bars being exposed to space. Because of concern for the possibility of leakage at those brazed joints, the "BHXU-A" was also built; following flat-milling of the brazed plates and header bars, an additional metal plate was brazed onto that flat surface so that that metallic membrane would prevent gas leakage from any of the brazed joints beneath it. The measured performance of the BHXU recuperator is presented in figure 21. In every case, effectiveness is at least 0.90. The higher values of heat-exchanger effectiveness with the He-Xe mixture result from that gas's higher thermal conductivity than for simply Kr and display why the mixture is preferred over the pure gas. The design effectiveness of 0.95 was attained at rated pressure, and concomitantly, rated power, lower pressures producing even higher values of effectiveness (ref. 20).

This Brayton power-generating system was initially tested in a large vacuum chamber (fig. 22) for about 2500 hr (ref. 21); the power-generating system

itself is at the bottom-center. An organic liquid coolant (DC 200) was pumped through a waste heat exchanger and to a cylindrical radiator above the power-generating system (ref. 22). Two completely independent cooling systems were provided: two waste heat exchangers, two motor-driven pumps and two separate coolant passages in the radiator that shared common fins; one cooling loop was redundant. The radiating surface was coated with Z93, a radiator coating having thermal emissivity of 0.88 but still effectively reflecting the bulk of incident sunlight as well as Earth albedo in a near-Earth orbit.

Measured powerplant efficiency is shown by figure 23 (ref. 21). For power output of 10 kWe or more, net overall efficiency was 0.26 (rounded); this efficiency includes an estimate of 8% for heat loss from the radionuclide heat source itself. Thus, measured overall efficiency of the power-generating system itself was 0.29 (rounded). These efficiencies are based on the net electric power output from the powerplant after deduction of electric power for all the powerplant's auxiliaries such as voltage regulator and exciter, speed control, coolant pump and its power supply, and powerplant controls. These controls permitted unattended operation, and the powerplant was operated using these automatic control system throughout its 38,000-hr test. The demonstrated efficiency of this powerplant is thus five times that for thermoelectric power generation in the RTG program.

Several of the components of this powerplant were also tested individually and then improved through this investigation (ref. 21). As built, compressor efficiency was 0.02 below our expectation. Addition of a flow splitter in the duct elbow just ahead of the compressor and resetting compressor-exit stator vanes by 3° raised compressor efficiency from 0.80 to 0.82. Replacement of the turbine-exit duct by one with more gradual divergence raised turbine efficiency from 0.894 to 0.907. Internally consumed power could be reduced 400 W by a 1200-Hz instead of 400-Hz motor to drive a coolant pump of reduced head rise, by raising efficiency of power conditioning for the control system's signal conditioner (ref. 23) and by redesign of the speed control (ref. 24). Any new technology for realization of these improvements was demonstrated at the component level. Had these demonstrated improvements been incorporated into the Brayton powerplant, we estimate that powerplant net efficiency would rise by 0.035, that is from 0.26 to 0.30. Similarly, engine efficiency would have risen by 0.038 from 0.29 to 0.32 (ref. 21).

Under the space-power technology program, the broad base of Brayton-cycle technology generally available was thus shown to be directly applicable, even at powers as low as 10 kWe, and to permit power-generating efficiencies substantially in excess of those demonstrated by alternative concepts.

But powerplant durability is even more important than efficiency. For rotating machinery, demonstrated durability is crucial because of the belief widespread among mission managers that rotating machinery is of inherently short life. In the Brayton space-power program, the endurance period initially selected for the BRU was 10,000 hr. After that was achieved, the period for demonstration was extended to 50,000 hr, in part because continuing an ongoing, unattended test cost so little. Unfortunately, the electric heater supplying heat to the powerplant (although not a rotating component) failed after about 38,000 hr. The need to rebuild the heater produced a reassessment of the resources required for continuation of the test. Because

of the demand for resources, NASA Lewis in consultation with NASA Headquarters decided to stop the test at that point.

Two Brayton Rotating Units were tested and inspected at various intervals (ref. 25). The accumulated test times preceding the inspections were approximately 700, 3000, 5700, 11,000 and 21,000 hr on each given BRU. The inspection following 38,000 hr of operation was interrupted by retirement of the responsible individual and is still incomplete. The biggest change in the BRU at 38,000 hr, in comparison with earlier inspections, was a substantial pit on the outer surface of the turbine casing. This appears to have resulted from chemical attack from a contaminant but just what has not been determined. This BRU was initially tested in air for 1000 hr followed by 2000 hr in vacuum. The additional testing was all in air. Throughout the test, the cycle working fluid was either krypton or the helium-xenon blend.

BRU performance was stable throughout the entire 38,000 hr at 1600° F although the hot parts were soon discolored. The gas bearings and creep of the turbine rotor are two areas of concern relative to "wearing out" of the BRU. Neither the bearing surfaces, the adjacent shaft nor the thrust runner showed any signs of wear, as they should not have. Inasmuch as there might be continual rubbing of the bearing shoes on their pivots (fig. 19), the faying surfaces of the pivots were closely examined for wear. Although minor wear was evident at 700 hr, there was none thereafter, the initial change apparently being wearing in of locally nonconforming asperities. After 21,000 hr of operation, measured rotor creep was 0.03%, this in a metal deforming perhaps 25% to rupture; the turbine rotor was designed for 0.5% maximum, local creep in 50,000 hr.

The BRU was not entirely free of trouble. At 21,000 hr, the turbine-inlet volute was buckled and cracked, and a helium leak-check revealed a small leak at that point. This problem was attributed to two factors. First, the geometry of the turbine-inlet scroll (fig. 24) resulted in a weld over a weld at the site of the leak. Second, repeated thermal cycling of the powerplant between room and operating temperatures imposed cyclic loads on the turbine-inlet scroll. In particular, the pipe leading from the electric heater to the turbine was long, and therefore differential thermal expansion was substantial. Bellows in the duct were to accommodate this differential expansion, and gimbals at each bellows carried the longitudinal loads from internal gas pressure but still permitted angular deflections of the bellows. Over many thousands of hours, the pivots in these gimbals diffusion-welded and lost their pivoting ability. The unaccommodated thermal deformations of the turbine-inlet pipe finally buckled the turbine casing. Following this failure of the gimbals, one faying surface of each pivot was plasma-sprayed with a ceramic coating as a diffusion barrier to prevent the diffusion welding.

Several changes would likely cure the problems leading to this leak in the turbine housing. Use of a torus rather than a volute for the turbine-inlet plenum would avoid the weld over a weld. A shorter pipe between heat sources and turbine in a real space-powerplant would reduce the differential thermal expansion. For the gimbal pivots, coating one face of the faying surfaces with, say, aluminum oxide should prevent the diffusion welding that occurred. And deliberately reducing the number of thermal cycles imposed during the test program would further diminish fatigue of the turbine housing.

Although the recuperator in that Brayton powerplant was very effective in transferring heat (ref. 20), mechanical problems were encountered. Three identical recuperators were built, and all three cracked and leaked following 20 to 30 deep thermal cycles between room and operating temperatures. During manufacture of the recuperator, brazing of the core made it into a monolithic structure. Before powerplant startup, the recuperator would, of course, all be at constant temperature. During powerplant startup, hot gas leaving the turbine at perhaps 1400° F would enter the hot end of the recuperator, and cold gas leaving the compressor at perhaps 100° F would enter the cold end. Equilibrium gas temperatures would be about 1230 and 278° F for the entering hot and cold gases, respectively (ref. 26). Corresponding to the effectiveness of 0.95 under design conditions, a 48° F temperature difference between adjacent hot and cold channels would persist during normal steady-state operation. These temperature variations imposed thermal stresses, and the thermal cycling led to cyclic fatigue.

Following the leaks in these recuperators, a technology program was begun on design and fabrication of such heat exchangers specifically to exploit the heat-exchanger-manufacturing skills of the Garrett Corporation and the background of the Lewis Research Center on low-cycle fatigue (ref. 27); this effort was later directed toward the Mini-Brayton system, a concept for space power that demonstrated power-generation efficiency of 0.25 at 1 to 2 kWe. The original recuperators were built of 347 stainless steel, a fairly weak material, and Nicrobrazes, which produced structures too brittle for this service. Following analysis of the low-cycle fatigue of these heat exchangers, Hastelloy X structural material and Palnbro brazes were chosen. A recuperator made of these materials was subjected to 100 deep thermal cycles between room and operating temperatures without incident. Following an additional 100 thermal cycles, a helium mass spectrometer detected a tiny leak between the hot and cold passages, a leak so small that system performance would be unaffected (ref. 28). Because its outer surface was perfectly sound, no gas would leak from the recuperator to space.

The 100 cycles demonstrated are more than sufficient for Brayton recuperators intended for use in space. A total of 100 to 200 deep thermal cycles is most likely also adequate for a ground-test program for a space-power system. The basic technology for recuperators is thus pretty soundly based for turbine-inlet temperatures up to 1150 K (1600° F).

Recapitulation - Overall, this 10 kWe Brayton powerplant exceeded its programmatic goals for both efficiency and power output. Although the rotating machinery of this experimental powerplant ran without incident for 38,000 hr, the powerplant encountered problems with its static components. In particular, diffusion welding of gimbal pivots rigidified bellows in the turbine-inlet duct, thereby imposing thermal-deformation loads on the turbine casing and eventually buckling and cracking the casing; adding a ceramic diffusion barrier prevented further welding of these pivots and eliminated the problem. Cracking of the recuperators following 20 to 30 deep thermal cycles was solved in a technology program, but this technology has not been demonstrated at the 100 kWe power level. In general, raising Brayton power level above 10 kWe moves the concept toward a region of much broader experience and makes the potential problems easier to solve. Structural materials for 1150 K (1600° F) are conventional and can be readily employed in design of Brayton space-power systems.

Gas Bearings

Although the gas bearings in the Brayton Rotating Unit (BRU) performed without problems for 38,000 hr, the general complexity of these bearings (fig. 19) has resulted in increasing emphasis on a foil-type of gas bearings. Figure 25 shows various applications of foil bearings by the Garrett Corporation; both journal diameter and rotor weight are given. Although a few of the applications shown are experimental, others are in regular use on 727, DC-10, C-5A and F-4 aircraft, for example.

The foil-bearing concept pursued by Garrett is shown in figure 26. Curved, flexible foils are first inserted into the journal-bearing housing; a slot in the housing retains a tang on each foil and thereby positions the foil. Inserting the shaft into the housing requires deflecting and further curving the foils, so that each foil then contacts both the shaft and an adjacent foil. Rotation of the shaft draws gas between foil and shaft, thereby lifting the foil off the shaft and further flexing the adjacent foil. During normal operation, the foil rides on a film of gas and does not touch the shaft; bearing operation is hydrodynamic. Differential thermal deformation of shaft and housing are accommodated by flexing of the foils, the clearance between foil and shaft remaining nearly constant. Although there should be no bearing wear during normal operation, the reverse is true during starting and stopping.

If a BRU were shut down, rotor speed would gradually decrease, and finally the rotor would stop. During deceleration of the rotor, the clearance between shaft and foil would gradually decrease until the foil would rub on the shaft, this rubbing continuing until the rotor stops. Similarly, when the BRU is started, the foils rub on the shaft until a rotational speed is reached that is sufficient for a gas film to support the foil. (For the 10 kWe powerplant as well as for current concepts, the BRU and thus the powerplant are started by supplying electric power to the synchronous alternator and thereby driving it as a motor to perhaps one-third of rated speed, the BRU then accelerating on its own.)

Current experience in aircraft service shows the foil bearings' tolerance of this starting and stopping. Garrett currently has 4000 foil-bearing turbo-machines mass-produced and in service in the field. These machines have accumulated 45 million operating hours and have experienced mean times between failure ranging from 55,000 to 95,000 hr.

Figure 27 portrays a small Garrett gas turbine for which the temperature of the bearing at the turbine outlet was too high for oil lubrication; a foil type of gas bearing was, therefore, installed. Figure 28 shows the bearing itself as well as some operating conditions imposed on the bearing. This aircraft gas turbine was also subjected to flight maneuvering loads of 5, 10, 15 and 16 g's without mishap. In addition, a shaft load equivalent to a 16-g maneuver was imposed during ground test while the bearing was heated to 800° F (700 K). In three additional Garrett programs, foil gas bearings were successful in the following tests: Navy APU at 22.6 g's and 990° F bearing temperature; 100,000 rpm in three automotive gas turbines in a DOE program; and USAF turbojet at 15.8 g's and with 1000° F bearing temperature.

Under a NASA-sponsored program at Mechanical Technology Inc. (MTI), the following coatings for bearing foils and shafts were evaluated, although not all the possible combinations were explored:

For foils: B_4C , Cr_2O_3 , AFSL-28, TiC , Al_2O_3 , Cr_3C_2 , CrB_2 , WC , TiB_2 , MFL-5, Ag, Si_3N_4 , Kaman DES, and CdO plus graphite plus Na silicate.

For shafts: TiC , WC , borided A286, nitrided A286, B_4C , Si_3N_4 , Tribaloy 800, CrB_2 , Al_2O_3 , Cr_2O_3 , Cr_3O_2 , NASA PS-100, NASA PS-101, and CdO plus graphite plus Na silicate.

Additional coatings were evaluated at NASA Lewis. From this screening of coatings, NASA and MTI selected and qualified the following coating combinations:

Up to 925 K: Cr_2O_3 and Cr_3C_2
Up to 1075 K: ZrO_2 and CaF_2

As tested, these coatings were for application in air-breathing engines and have not been tested in the non-oxidizing environment of a space-power Brayton cycle. Based on their own program, Garrett has chosen the following coatings: SiC on bearing foils and SiO_2 plus Cr_2O_3 plus Al_2O_3 on the shaft.

Materials

Refractory Alloys - Although the 10 kWe BRU performed successfully at 1150 K (1600° F), the question arises concerning possibly higher temperatures, especially in combination with the SP-100 reactor at its contemplated use temperature of 1500 K. Fortunately, a considerable amount of past research emphasized development and evaluation of refractory alloys for use at high temperature in space. A considerable amount of high-temperature creep testing is reported in, for example, references 29 and 30. Although long-time creep-strength is emphasized here, the approach to alloy development and evaluation was broad and followed the approach advocated in reference 31.

Among the alloys evaluated was ASTAR-811C (Ta-8W-1Re-0.7Hf-0.025C). Reference 29 gives creep data from 70 separate tests of this alloy that spanned 1140 to 1920 K (1600 to 3000° F) and totalled 250,246 test hours, measured creeps ranging from 0.108 to 15.088%. Four individual tests exceeded 10,000 hr apiece and totalled 71,696 hr.

Herein these data are correlated for 1% creep by use of the Larson-Miller parameter (LM), where

$$LM = T (15 + \log_{10} t)$$

where

T temperature, °R
t time to reach 1% creep, hr

For good correlation, the data were divided into two groups, as follows:

- (1) Low-stress range: stresses not exceeding 10,000 psi (1 MPa = 145 psi).
- (2) High-stress range: stresses at least 5000 psi and test times over 3000 hr.

The overlap in the stress ranges was deliberate. For the low-stress range, the data were correlated for log-stress versus LM (fig. 29); in the high-stress range, stress versus LM was used (fig. 30). In each case, a straight line was fitted to the data by the method of least-squares, and the standard deviation of the data from that line was determined. A second straight line was then drawn parallel to the first but shifted to lower Larson-Miller parameter (and thus to lower stress) by twice the standard deviation (fig. 29 and 30). The probability that a given datum would lie on the high-stress side of the 2-sigma line is 0.977, a factor giving some confidence in application of these results in design. (The odds exceed 40 to 1.)

With the 2-sigma margins in figures 29 and 30, the stresses to produce 1% creep in ASTAR-811C over a period of 40,000 hr were estimated to be as follows:

Temperature		Stress	
K	°F	MPa	psi
1300	1880	155	22,000
1400	2060	96	14,000
1500	2240	36	5,000
1600	2420	7	1,000

The strength of 36 MPa (5000 psi) is adequate for design of heat pipes, ducts, heat exchangers, structure of nuclear reactors, and turbine housings but generally not for turbine rotors, especially so if the density of 16 g/cm³ for ASTAR-811C is taken into account.

Fortunately, rotor temperatures for a Brayton-cycle turbine are substantially below turbine-inlet temperature. For illustration of this point, the turbine was analyzed for the Brayton-cycle study sponsored by JPL (ref. 32); for the thermodynamic cycle selected in that study (fig. 31), turbine-inlet and -outlet temperatures are 1500 and 1169 K (2240° and 1644° F). Stagnation gas-temperatures relative to the turbine rotor (fig. 32) lie between these limits; because stagnation temperatures are, by definition, adiabatic values, actual rotor temperatures are expected to be below the stagnation temperatures. In the low-stress region at the rotor's outer radius, stagnation temperature is 1330 K (1934° F), 170 K (306° F) less than turbine-inlet temperature. Stagnation temperature falls to 1250 K (1790° F) at 70% of the outer radius and is roughly 1200 K (1700° F) for the central half of the radius, the highly stressed portion of the turbine rotor. As the table above shows, ASTAR-811C is fairly strong in this region. However, these temperatures are low enough that alternate materials might also be considered.

The creep testing of the molybdenum alloy TZM (Mo-0.5 Ti-0.08Zr-0.03C) is summarized in reference 29. Eleven specimens were tested for temperatures

ranging from 1145 to 1440 K (1600° to 2130° F). The time totalled 94,140 hr, and the longest single test was 45,354 hr (5.17 years). These data were correlated on the bases of (1) stress versus Larson-Miller parameter and (2) log-stress versus Larson-Miller parameter (fig. 33), just as for ASTAR-811C. However, the small number of tests of TZM made inadvisable any division of the data into two different sets; for each correlation, the entire set of 11 tests was therefore used. Inasmuch as the two correlation coefficients are 0.92 and 0.87, respectively, this factor does not provide a strong incentive to choose one approach over the other. On the other hand, the linear correlation probably underestimates creep strength at the highest temperatures.

For the 2-sigma lines shown, the following stresses would produce not over 1% creep for TZM in 40,000 hr:

Temperature		Allowed stress			
		Linear correlation		Logarithmic correlation	
K	°F	MPa	psi	MPa	psi
1169	1644	293	42,000	229	33,000
1200	1700	240	35,000	170	25,000
1250	1790	155	22,000	106	15,000
1300	1880	69	10,000	66	9,600
1330	1934	18	2,700	50	7,200

Thus, TZM is sufficiently strong (on either basis) for construction of turbine rotors of Brayton power systems at turbine-inlet temperatures up to 1500 K.

The substantial body of available data on these two alloys, ASTAR-811C and TZM, shows on a conservative basis that Brayton space-powerplants may be designed for long lives (up to 100,000 hr) at turbine-inlet temperatures up to 1500 K (2240° F).

Philosophy - The following factors were important in choice of materials for high-performance, long-lived Brayton-cycle powerplants for use in space. This approach is recommended for every component (e.g., the reactor) and for every powerplant.

- (1) A good data base: For the two refractory alloys discussed, a considerable amount of creep testing (344,000 hr, or 39 years) has already been performed. Each of these alloys is also compatible with the working fluids, both inert gases for the Brayton working fluid and the alkali metals, either lithium or sodium, for the reactor coolant. In addition, a considerable effort on ASTAR-811C has shown it to be both weldable and ductile postwelding. Inasmuch as TZM, along with many molybdenum alloys, is brittle postwelding, its joining should avoid welding; a turbine rotor could be bolted instead of welded to its shaft.

- (2) Allowance for scatter of the data: Following correlation of the substantial data base cited above, some allowances should be made for the scatter of the data, scattering not only among the data themselves but also their departure from the correlating concept. As presented here, straight lines were fitted to the data by the method of least-squares, the standard deviation of the data from the correlating line then being determined. Designing on the basis of the correlating line itself is inappropriate inasmuch as half the specimens tested were weaker than the line indicates. Instead, a "design" line was shifted to lower stress by two standard deviations.
- (3) Conservatism in applying the data: For appraisal of both ASTAR-811C and TZM herein, the strength criterion was 1% creep in 40,000 hr. For a 7-year (61,000-hr) mission, actual creep should then not exceed 2%. For a 100,000 hr (11.4-year) mission, creep would probably not exceed 3%. Inasmuch as these alloys deform 20 to 30% before rupture, rupture itself seems very unlikely.

ASTAR-811C was selected for emphasis herein for reasons other than just its creep-strength; these include the sheer volume of creep data, its weldability, its low ductile-to-brittle transition temperature (DBTT) postwelding and its compatibility with candidate working fluids. ASTAR-1411C (TA-14W-1Re-0.7Hf-0.025C) and -1611C (Ta-16W-1Re-0.7Hf-0.025C) are stronger alloys within the same family that benefit from increased solution-strengthening from their increased contents of tungsten. A comparison of ultimate strengths about 10 years ago showed the following (1 ksi is 1000 psi):

Alloy	Temperature		Ultimate tensile strength	
	K	°F	MPa	ksi
ASTAR-811C	1365	2000	390	57
ASTAR-1411C	1365	2000	670	97
ASTAR-1611C	1365	2000	810	118
ASTAR-811C	1535	2300	290	42
ASTAR-1411C	1535	2300	500	73
ASTAR-1611C	1535	2300	630	91

Similarly, the stresses producing a 1% creep in 10,000 hr were also estimated to be as follows:

Alloy	Temperature		Creep strength	
	K	°F	MPa	ksi
ASTAR-811C	1365	2000	140	21
ASTAR-1411C	1365	2000	220	32
ASTAR-1611C	1365	2000	275	40
ASTAR-811C	1535	2300	48	7
ASTAR-1411C	1535	2300	130	19
ASTAR-1611C	1535	2300	165	24

In comparison with ASTAR-811C, ASTAR-1611C has roughly three times the long-time creep strength at 1535 K. Alternatively, ASTAR-1611C could provide the same creep strength at 1535 K as ASTAR-811C provides at 1365 K. Even though some future technology program on welding of ASTAR-1411C and of -1611C might show that these stronger materials are usable in fabrication of Brayton powerplants, the approach herein emphasizes instead the greater knowledge of ASTAR-811C.

Although TZM is one of the strongest known alloys of molybdenum, its application herein was limited to the fairly low temperatures characteristic of the turbine rotor because of its low strength at high temperature. On the bases of the correlations in figure 23 and the 2-sigma allowances for scatter of the data, TZM has the following strengths at high temperature: For the linear correlation, TZM is limited to 1340 K (1950° F) if at a stress of 7 MPa (1000 psi) its creep is to be held to 1% over 40,000 hr. For the logarithmic correlation, TZM is limited to 1540 K (2310° F) for 7 MPa stress and 1% creep in 40,000 hr.

For the SP-100 reactor, a molybdenum alloy (currently Mo-13 Re) is planned for the heat pipes, reactor vessel and structure. Fuel hot-spot temperature is about 1730 K (ref. 8). The reactor vessel and structure, being nearly adiabatic, will operate close to that peak fuel temperature. Inasmuch as Mo-13 Re probably has lower creep strength than TZM, the data for TZM might provide an upper bound on creep strength of the Mo-13 Re planned for the reactor. The logarithmic correlation for TZM (fig. 33(b)) yields the estimate that at 1730 K (2650° F) and at 7 MPa (1000 psi), a reactor structure of TZM would creep 1% in only 260 hr; reducing stress to 3.5 MPa (500 psi) would increase the time for 1% creep to 1800 hr. Thus, the philosophy in design of the SP-100 reactor differs substantially from that suggested herein for Brayton power generation.

Advanced Materials for High Temperatures - Alternate materials offer the promise of higher temperatures in the future. In particular, SiC and Si₃N₄ are being actively investigated and developed for use as monolithic ceramics in automotive gas turbines at 1650 K (2500° F). Development of these materials for application in gas turbines was initiated by the Defense Advanced Research Projects Agency (DARPA). Following several years of DARPA support, DOE provided funds for a program on ceramic automotive gas turbines to be carried out by NASA Lewis in cooperation with the automobile industry; both

SiC and Si₃N₄ are under active, continuing development. Two turbine rotors made of SiC are shown in figure 34. In figure 35, the turbine rotor appears in combination with the combustor liner and turbine housing, also of SiC, and with the regenerator disk and housing of glass-ceramic. Figure 36 presents closeup views of the regenerator disk along with material of construction and thermodynamic performance; regenerator effectivenesses are 0.89 and 0.92 at rated power for the two designs, rising to 0.94 and 0.97 at 40% of rated power.

The SiC and Si₃N₄ monolithic ceramics have potential operating temperatures of 1600 to 1900 K (2400° to 3000° F) but suffer from being brittle. For this reason, both toughness of these ceramics as well as designing for use of brittle materials are emphasized in the automotive program. However, the long-time creep data needed for space-power systems are not available, a suitable life for automotive gas turbines being only 3500 hr. The lithium-aluminum-silicate glass-ceramic has been shown suitable for use in regenerator disks at temperatures to 1350 K (2000° F).

Reinforcing such glass-ceramics with SiC fibers has several promising features. Such a ceramic-matrix composite has greater toughness than the monolithic ceramics, and the SiC fibers add to the strength of the glass-ceramic. If fully successful, this ceramic-matrix composite might permit increasing turbine-inlet temperature to 1900 K (3000° F).

Carbon-carbon composites are also under active investigation by DARPA and NASA Lewis and appear to have the potential for operation at 2500 - 2750 K (4000° to 4500° F). In addition, the absence of oxygen in space should make utilization of carbon-based materials simpler than in the terrestrial environments for which they are being evolved. Among the metals, tungsten has great potential for strength at high temperature, exceeding molybdenum or even tantalum alloys. The key is to evolve tungsten alloys that, while having high long-time creep strength, are still fabricable as well as ductile postwelding. If made of alloyed tungsten, Brayton-cycle turbines might reach 1900 K (3000° F).

Performance Potential at High Temperature

Under sponsorship and guidance by JPL, the Garrett Corporation conducted a design study of a reactor-Brayton space-power system to operate at a peak temperature of 1500 K (2240° F; fig. 31 and ref. 32). As specified by JPL, design lifetime for the powerplant was 120,000 hr (13.7 years) at full power, and the 400 kWe net output power was conditioned for direct supply to ion thrusters. Two 400 kWe power-generating systems were included in the powerplant, one being completely redundant.

As shown in figure 31, the required reactor power is 1650 kWt, the powerplant's overall efficiency being 0.24, including power conditioning. The reactor designs were produced by Los Alamos National Laboratory (LANL) and were, for comparability, blood relatives of the reactor for SP-100. Figure 31 shows schematically the heat pipes extending from reactor to heat-source heat exchanger (HSHX); the coldest heat pipes will see the 1114-K entering cold gas, and the hottest heat pipes the 1500-K exiting hot gas. The reactor-outlet temperatures might run 50 K above these values, that is, from 1164 to 1550 K. This range of reactor-outlet temperature for Brayton produces a condition seldom emphasized: For a 1500-K Brayton powerplant, average

reactor-outlet temperature (1357 K) is substantially below that for a comparable thermoelectric power-generating system.

In a thermoelectric concept in which the thermoelectric modules are directly heated by the reactor's heat pipes, reactor-outlet temperature might be (as for Brayton) 50 K (90° F) above the hot-side temperature. Consider, for comparability with Brayton, thermoelectric hot-side temperature of 1500 K; reactor-outlet temperature for each heat pipe would be 1550 K, and average reactor-outlet temperature 1550 K as well. Thus, for the same peak cycle temperature, the thermoelectric concept would impose on the reactor an average temperature about 200 K (360° F) higher than would the Brayton concept.

This assessment of Brayton versus thermoelectric power generation on the basis of the required average reactor temperature is particularly relevant to the SP-100 reactor: The fuel is all in one volume, namely, that space outside the heat pipes and inside the core vessel. Those alternate reactor concepts employing multiple fuel pins contrast with the SP-100 reactor because for those reactors the reactor designer would be concerned about that pin having the most severe combination of fuel burnup and fuel temperature. But because the current concept for the SP-100 reactor has only a single fuel volume, one is chiefly concerned with the average conditions imposed on the reactor.

In addition, the thermoelectric concept for SP-100 employs radiant heat transfer from the reactor's heat pipes to the thermoelectric module's hot side, this radiant heat transfer requiring a temperature drop of the order of 100 K (180° F) which depresses system performance. This radiant heat transfer avoids the need for an electrical insulator on the hot side of the thermoelectric generator, a design feature that both permits high-voltage power generation and avoids the insulator's technological limitations on the thermoelectric concept. Thus there are sound technical reasons for imposing on the thermoelectric concept a 100 K penalty in temperature that is unnecessary in the Brayton concept.

Brayton at 1500 K is thus at the same state of technology as thermoelectric at 1300 K. As cited herein under Materials, the current materials technologies for ASTAR-811C and TZM make them suitable for application in Brayton space-powerplants at 1500 K (2240° F). Similarly, the current technology for SiGe thermoelectric materials limits their application to 1300 K (1880° F). Of course, technology programs might later make practical the operation of thermoelectric concepts at higher temperature, but that is true of almost any concept, Brayton included.

In addition, thermoelectric power generation at 1300 K imposes on the SP-100 reactor an average fuel temperature 200 K (360° F) higher than Brayton power generation at 1500 K.

The weight breakdown for the Brayton powerplant in the JPL-Garrett study (ref. 32) is presented in table I, specific mass for the powerplant totalling 21 kg/kWe. The 39 kg/kWe for thermoelectric power generation (ref. 1) is almost twice as great.

The 100% redundancy imposed on Brayton power generation has only a modest effect on system mass, adding only about 10%. Still, inclusion of such an idle, redundant mass seems a bit extravagant. Consider, if you will, reli-

ability of each power generation module to be 0.95; there would then be only a 0.05 probability that the idle, redundant powerplant would be needed.

An alternate approach would include multiple powerplants and would run them all to generate their rated powers. Inasmuch as degradation of the thermoelectric generators will lead to a 25% loss of power over 10 years (see THERMOELECTRIC POWER GENERATION), a similar loss in power appears tolerable for the Brayton concept. That 25% loss could arise from failure of one powerplant out of four, or two out of eight, for example. If each individual powerplant's reliability is 0.95, then the probability that at least 3 powerplants survive out of 4 is 0.986; for survival of 6, 7, or 8 powerplants out of 8, overall reliability is 0.9942.

The performance potential of Brayton-cycle powerplants at higher temperatures is shown by figure 37; as before, the measures of merit are efficiency of power generation and radiator area per kilowatt. At 1500 K, efficiencies of 0.2 to 0.25 correspond to radiator areas of 0.25 to 0.30 m²/kWe. In comparison with thermoelectric power generation at 1300 K, these efficiencies are about four times those for the thermoelectric concept with radiator area per kilowatt less than half as large.

If future advances in materials permit raising Brayton's turbine-inlet temperature to 2000 to 2500 K, substantial reductions in radiator area are achievable, the area being in inverse proportion to T^4 . At power-generation efficiencies of 0.2 - .25, radiator area per kilowatt would drop to 0.08 to 0.095 m²/kWe at 2000 K and to 0.03 to 0.04 at 2500 K. At the highest operating temperatures, radiator area might become so small that greater emphasis could be placed on raising overall efficiency to the 0.3 to 0.4 range and accepting the concomitant modest penalty in radiator area.

Combination with Gas-Cooled Reactors

The combination of Brayton-cycle power generation with nuclear reactors cooled by an inert gas is a natural one. The High-Temperature Gas-Cooled Reactor (HTGR) already in commercial service at Fort St. Vrain, Colorado, provides a very substantial base of technology for nuclear space-power systems (ref. 33). In this reactor, the uranium and the fission products are contained within small beads coated with dense pyrocarbon and SiC (ref. 34), these beads being embedded in a graphite matrix that contains channels for flow of the helium coolant. For the HTGR-outlet temperature of 980 K (1300° F), the fuel beads themselves operate at approximately 1500 K (2240° F), the high temperature drop resulting from the high power density. For the long lives and lower power densities usually characteristic of reactors for power generation in space, reactor-outlet temperature could be much closer to the fuel temperature. Fuel particles of this design were tested at 1620 K (1350° C) to extremely high burnups, namely, 70% fission per initial metal atom (ref. 34); in spite of the severity of these operating conditions, the fuel particle's coating was still sound, and no interaction had occurred between fuel and coating. Reactor-outlet temperatures of 1400 to 1500 K thus appear achievable with a highly developed, well proven technology, these temperatures being consistent with the materials technology presented herein for Brayton-cycle power generation.

Although not yet employed in commercial application of HTGR, ZrC was extensively evaluated as a substitute for the SiC barrier in the coated-fuel beads (ref. 35). This evaluation of the ZrC coating showed that the fuel and fission products were retained within the fuel beads at temperatures up to 1800 K (2780° F), an achievement permitting reactor-outlet temperatures up to about 1700 K (2600° F) in long-lived applications.

Inasmuch as the fuel beads themselves are the basic modules for confining both the nuclear fuel and the fission products, the highly developed technology for HTGR is readily applied in design and construction of nuclear reactors for combination with Brayton-cycle space-power systems. The strong current basis of technology for both reactor fuel and Brayton-cycle materials shows that reactor-Brayton powerplants can be readily developed in the near term for use at operating temperatures of 1500 K (2240° F).

Considerable emphasis has recently also been placed on extension of this technology for HTGR to higher temperatures, perhaps to reactor-outlet temperatures ranging from 2000 to 2500 K (ref. 36). Such future reactor technology in combination with the advanced materials for Brayton powerplants (see Materials herein) may provide the future capability to achieve the performance potential of Brayton at 2000 to 2500 K (3100° to 4000° F) shown in figure 37.

RANKINE CYCLE

Many, many Rankine-cycle powerplants use water as the working fluid and operate successfully here on Earth. Mercury was also used in Rankine-cycle topping powerplants for about 40 years and, although marginal economically, was successful technically. In addition, mercury was explored for about 10 years for use in space in combination with the SNAP-8 U-ZrH reactor. Various organic fluids have been investigated on and off for nearly 25 years for use in organic-Rankine space-power systems.

None of these fluids is suitable for use with nuclear reactors having outlet temperatures of 1300 K (1880° F) and above, but the alkali metals are. Of these, the greatest amount of technology was evolved for potassium and substantially less for sodium, rubidium, and cesium.

The principal technology evolved was for the potassium-Rankine cycle shown schematically in figure 38 (ref. 37). Lithium at 1420 K (2100° F) is pumped through the reactor and heated to 1480 K (2200° F). This stream of lithium counterflows with a stream of potassium in a boiler. Having boiled at 1365 to 1395 K (2000° to 2050° F), the potassium is superheated to 1420 to 1450 K (2100° to 2150° F) and then expanded through the turbine. Separate cooling loops and radiators cool the condenser and the major electrical components, namely, the alternator and the pumps.

Although a powerplant of this class was never built and tested, a great deal of the enabling technology has already been evolved. (See references 38-40, for example.)

Materials

The tantalum and molybdenum alloys described under BRAYTON CYCLE are compatible with lithium and potassium at temperatures up to 1500 K (2240° F). For

example, the tantalum alloy T-111 (Ta - 8W - 2Hf) was evaluated in two coupled loops roughly simulating the powerplant in figure 38. Lithium was heated from 1365 to 1480 K (2000° to 2200° F) and used to boil potassium at 1395 K (2050° F), which was then superheated to 1450 K (2150° F). After being throttled to about one atmosphere, the potassium was condensed at 1045 K (1420° F) and then pumped once more into the boiling region. This operation continued for 10,000 hr, and a post-test evaluation revealed that both loops were still in excellent condition. Of course, oxygen was carefully controlled, the oxygen levels of the two liquid metals being reduced to only a few parts per million before the test began. In addition, the test was conducted in vacuum.

The properties of materials alternate to ASTAR-811C and T-111 are described in references 41 to 46.

A selection of high-temperature electrical materials was also evaluated for use at hotspot temperatures up to 980 K (1300° F) (refs. 47 and 48). For example, not only were the magnetic characteristics of Nivco (Co-23Ni-1.1 Zr-1.8Ti) assessed at high temperature but its creep strength was measured as well (fig. 39). Several electrical components (solenoid, transformer, and stator for a 3-phase alternator or motor) were built of these materials and demonstrated at 980 K (1300° F) for 10,000 hr. This demonstration is a critical technology for the dynamic powerplants (Brayton, Rankine or Stirling cycles) because it so relieves the problems (i.e., radiator area) of cooling the electrical components of those systems.

Of the other high-temperature materials described under BRAYTON CYCLE, tungsten alloys will likely tolerate the alkali metals at high temperatures, but the other materials would all require protection from them.

Component Technologies

The principal components of a Rankine system are boiler, turbine, generator, condenser, and pumps. For the generator, the technology is that described above. The technologies for the other components will be described in turn.

Boiler - The boiler must, of course, discharge dry vapor, for slugs of liquid would likely damage the turbine or at least shorten its life. Here on Earth gravity separates the liquid and vapor phases so that dry vapor is produced, but that mechanism is not available in a spacecraft not being accelerated. This led to the once-through concept of boiler design, in which liquid enters the cold end of a cluster of boiler tubes, heat is added by the counter-flowing, hotter lithium, and dry, superheated vapor exists from the tube cluster (fig. 40).

Two problems in such a boiler are flow stability and the achievable heat-transfer coefficients. The selected geometry should also permit construction by assembly of elements that are not only easy to manufacture individually but that also permit simple assembly into a complete boiler. This combination of problems is solved by means of an insert in each boiler tube such as that in figure 41. The center plug restricts the flow channel available to the liquid, thereby increasing liquid velocity as well as introducing a pressure drop that provides the desired flow stability. The coiled wire (a spring) rotates the now rapidly moving liquid, creating centrifugal acceleration of

10's of g's and holding the liquid onto the tube wall; heat transfer from the tube wall is high as long as the liquid remains in contact with the wall. Vapor generated by the boiling liquid is separated from the liquid by the artificial-g field. Eventually fine liquid droplets are entrained by the fast-moving vapor, and the heat transfer declines to that characteristic of flowing gas instead of boiling liquid. Figure 42 compares the heat-transfer coefficients achieved with and without an insert that was a helical strip attached to a central rod, the insert providing much higher heat-transfer coefficients at either high exit quality or superheat

The technology for such boilers is summarized in references 49 to 60. Similar technology for a mercury boiler heated by counterflowing NaK is described by references 61 and 62.

Turbine - For a given turbine-inlet temperature, the structural problems of a Rankine-cycle turbine are much like those of the Brayton-cycle turbines already discussed, but the possibility of blade erosion is an additional problem. Just as with steam turbines here on Earth, expansion of potassium vapor produces liquid droplets through partial condensation of the vapor within the turbine. Under some circumstances, this can lead to erosion of the turbine blades and a limited life (ref. 63). Analysis of this condensation and possible erosion (ref. 64) based on the experience described in reference 63 sought to define conditions for which erosion would and would not be a problem. References 65 and 66 explored condensation of and erosion by steam and inferred tolerable operating conditions for other working fluids.

For assessment of the impact of this possible erosion on life of potassium turbines, two such turbines were built and tested with saturated potassium vapor at about 1100 K (1500° F). Although this turbine-inlet temperature is far below the desired value for an advanced space-power system, the temperatures and the vapor qualities throughout the test turbines are representative of those in the final stages of such a space-power turbine, and it is those final stages that might encounter blade erosion. For the test turbines, the first had two stages, and 0.92 was the quality of the vapor (i.e., 8% liquid) at the turbine exit. Operation of the turbine for 5000 hr produced negligible erosion of the blades. A subsequent 3-stage turbine further expanded the potassium vapor; vapor quality was 0.93 entering the third stage and 0.87 leaving. After 5000 hr of operation (fig. 43), the tips of rotor blades of Rene 77 in the third stage had the microscopic beginnings of erosion (refs. 67 to 69). Molybdenum-alloy blades (both TZM and TZC) in the third stage suffered substantially greater loss of blade metal, but this was attributed to chemical attack rather than erosion; the chemical attack on these blades apparently resulted from oxygen contamination of the potassium loop. This judgment concerning corrosive attack on moly blades in the third stage is supported by what occurred to moly blades in the second stage; their weight loss was 30 to 40 times what was observed for second-stage rotor blades in the preceding test of the two-stage turbine. The modest erosion of the Rene 77 rotor blades was judged to be representative of moly-alloy blades without chemical attack.

Although the 5000-hr duration of this test was too short to definitely establish that erosion would be a significant problem in missions of long duration (10 years or more), the three-stage turbine was then modified for liquid extraction (ref. 70). The effectiveness of this condensate removal in the

turbine was comparable with that for the steam turbines in terrestrial nuclear powerplants.

On the bases of these tests as well as the technology on electrical materials, two preliminary designs were evolved for potassium-vapor turboalternators (refs. 71 and 72); figure 44 is representative of the two designs. After partial expansion in a high-pressure turbine, the vapor is reheated; this reheating not only keeps turbine-exit quality within the range found acceptable in the test program but also raises powerplant efficiency. In both designs, the turboalternator rotor is supported by bearings lubricated by liquid potassium. Figures 45 and 46 illustrate the classes of bearing explored for this application (refs. 73 to 75).

Condenser - In the 1970 time frame (ref. 37), the condenser concept was a multitube heat exchanger (fig. 47) with potassium vapor entering the tubes and pumped NaK counterflowing in the heat-exchanger shell. That pumped NaK could be cooled in the radiator by thermal radiation from either metallic fins or from "vapor chambers," i.e., distorted and adapted heat pipes (ref. 76). Alternatively, heat pipes could replace the pumped NaK in both cooling the condenser and transporting the heat of condensation to the radiating surface.

References 77 to 80 describe the measured performance of such condensers, and figure 48 shows some of the measured data. In such a condenser, the vapor entered one end of the cooled tube, flowed longitudinally through the tube, condensed along the way and exited as a liquid from the other end of the tube. Inasmuch as the condensation of mercury under 0-, 1-, 1.5- and 2-g environments (refs. 81 to 87) was so extensively investigated, a similar study for potassium may not be needed. In any event, this past work on mercury shows what would likely occur for condensing potassium. Liquid condensed on the walls is disturbed by the rapidly flowing vapor, and waves form. Waves coalesce into slowly moving droplets. Under the action of hydrodynamic forces, the droplets are lifted off the wall and accelerated by the rapidly flowing vapor. The inertia of these accelerated droplets carries them through a final region of slowly moving vapor, and they impact the vapor-liquid interface near the tube outlet. Inasmuch as hydrodynamics rather than gravity provide the principal forces for moving the condensed liquid, the observed effect of gravity was very small for the range from 0- to 2-g's.

Pumps - The state of technology for electromagnetic (EM) pumps is reflected in references 88 to 93. Figures 49 and 50 show a pump built from this technology as a boiler-feed pump in a potassium-Rankine power system. The design conditions were as follows (ref. 92):

Potassium temperature, 810 K (1000° F)
Coolant temperature, 700 K (800° F)
Pressure rise, 1.65 MPa (240 psi)

NaK was chosen as the coolant.

The required pressure rise is very high for an EM pump. The necessarily long potassium duct was made compact by using a helical shape. The input power was 60-Hz ac at 135 V, thereby avoiding the power-processing losses associated with the low-voltage dc of a Faraday EM pump.

The high temperatures required use of advanced materials. The electrical materials were those in references 47 and 48, appropriate to hotspot temperatures up to 980 K (1300° F); up to 280 K (500° F) was thus available as the temperature drop within the pump for cooling. The magnetic material was Hiperco 27, the conductors nickel-clad silver and the insulation inorganic. The potassium-containing ducts were T-111 (Ta-8W-2Hf). On the other hand, the exterior of the pump was type-321 stainless steel so that the pump could be (and was) tested in air. The transition from T-111 to stainless steel was as follows: Nb-1 Zr was welded to T-111 and brazed to the stainless steel. This pump illustrates the existing experience in manufacturing complex components from tantalum alloys.

Measured performance was as follows: At the design temperatures for the potassium (810 K) and NaK (700 K), the design value of pressure rise (1.65 MPa) was achieved at the design potassium flow (1.47 kg/s) with an efficiency of 16.3%. The pump was tested with potassium temperature up to 1035 K (1400° F) and NaK-inlet temperature up to 755 K (900° F); the corresponding NaK-outlet temperature was 785 K (950° F). The pump performed successfully at its design conditions for 10,000 hr.

System Performance

Just as for Brayton, the tantalum alloy ASTAR-811C provides the enabling materials technology for operation at turbine-inlet temperatures up to 1500 K (2240° F). Of the candidate materials for higher temperatures (SiC, Si₃N₄, fiber-reinforced glass ceramics, carbon-carbon composites and alloyed tungsten), alloyed tungsten is probably the only one compatible with potassium. Conceivably some of these materials might be protected from potassium. For example, carbon-carbon might be coated with a film of chemically-vapor-deposited tungsten, leaving only the tungsten to face the potassium. On the other hand, even a pinhole in the tungsten coating would give the potassium access to the substrate, and progressive failure might result. Thus, substantial increase in operating temperature is more difficult for the potassium-Rankine concept than for the Brayton.

The potential performance of Rankine powerplants is thus shown in figure 51 for only 1500 K. Minimum radiator area is about 40% of that for Brayton at about 1500 K and only 15% of that for thermoelectrics at 1300 K, each at the current limit of its technology. This smaller radiator area has a substantial impact on system weight (table II).

The system study in reference 37 was used herein as the basis for weight estimation, but some adjustments were made for comparability with other concepts. In particular, the nuclear reactor and shield were replaced by interpolation among the various studies by Los Alamos National Laboratory. In addition, pumps for cooling the reactor and for cooling the condenser, the generator and the electronics were all replaced by heat pipes. In turn, the output power increased from 375 to 404 kWe because pump power was no longer required; powerplant efficiency rose accordingly from 0.19 to 0.205. Turbine-inlet temperature was increased from 1420 K (2100° F) to 1500 K (2240° F), and radiator area and mass were reduced a corresponding amount. Mass of structure was, as before, taken as 10%.

The 404 kWe powerplant then totals 5799 kg, specific mass then being only 14 kg/kWe. A Brayton powerplant at the same peak temperature weighs 50% more than this, chiefly because of its larger radiator. The SP-100 thermoelectric powerplant weighs almost three times as much.

The potassium-Rankine-cycle powerplant thus has potentially substantial advantages in both radiator area and specific mass in comparison with either Brayton-cycle or thermoelectric powerplants.

STIRLING CYCLE

The Stirling cycle is a concept for a closed-cycle reciprocating engine whose working fluids are high-pressure gases, either helium or hydrogen.

Historically, N. V. Philips Company of the Netherlands evolved the basic, enabling technology for the modern Stirling engine over about a 40-year period. Application of this technology to automotive engines was assessed in references 94 and 95, for example. Philips licensed their technology to General Motors Corp., to the Ford Motor Co., and to United Stirling AB & Co. in Sweden. References 96 and 97 summarize some of the early work in the USA on Stirling engines, and two publications (refs. 98 and 99) summarize the state of Stirling-engine technology in the mid-1970's.

Research supported by the Department of Energy on a Stirling engine for producing power in space from heat generated by Pu-238 is summarized in reference 100. Although that research engine had a great amount of mechanical trouble (chiefly from oil leaks), engine performance was measured. For peak temperature of 750° C and coolant temperature of 45° C, heat input was 4500 W and generator output 1000 W; accordingly, overall efficiency was 0.20, 95 W (ref. 100) being allotted to engine auxiliaries. Losses from the electric heater were estimated at 880 W; accordingly, estimated efficiency was raised in reference 100 to 0.25.

The largest research program on Stirling engines is supported by the Department of Energy in their automotive program, NASA's Lewis Research Center being the leading government laboratory. The principal contract is with Mechanical Technology, Inc., the principal subcontractors being United Stirling AB & Co. and A.M. General Corp. References 101 and 102 summarize the current technology for automotive applications, and reference 103 presents a means for analyzing and predicting performance of such Stirling engines. The current engines operate with peak metal temperature of 720° C and peak gas temperature about 50° C lower. Current redesign will raise those temperatures by 100° C. Thus, current technology for Stirling engines restricts them to peak temperatures substantially below the 1500 K goal for current concepts for nuclear reactors for space power.

Recent engine tests are reported in reference 102. This "basic Stirling engine" lacked both a control system and other auxiliaries. As an automobile engine, its output was also shaft power rather than electric power. Maximum power output was 73 horsepower. Maximum efficiency was 0.37 with a coolant temperature of 50° C and while producing 38 horsepower. For this "basic" engine, this efficiency was thus 55% of Carnot-cycle efficiency. Deduction of losses for auxiliaries and for power generation results in power output of 23-25 kilowatts-electric (kWe) at the efficiency peak of 0.3 to 0.33. For the

very low coolant temperature of 50° C and for a near-Earth-orbit sink temperature of 250 K, required radiator area is roughly 6 m²/kWe, about 10 times the radiator area required by even thermoelectric power generation. For effective use of Stirling engines in reactor space power systems, the existing concepts for engine design thus require substantial modification, extension, and adaptation.

Two life-limiting features of the automotive engine should also be changed for space application, as follows: (1) The so-called "kinematic" engine includes a seal on its piston rod that will likely continue to be life-limiting. That limiting component could be avoided entirely by use of a free piston and hermetic sealing of the entire engine. In concept, the free piston would drive either a linear alternator or a hydraulic pump such as that pursued for Stirling engines in the program on artificial hearts. (2) Helium would substitute for hydrogen as the working fluid within the engine. Although this change would require modest increases in size for the engine's heat exchangers, the propensities of hydrogen to diffuse through and to embrittle metals at high temperature would be completely avoided.

MHD

The large number of possible types of MHD powerplants can be classified according to the working fluid within the generator itself, namely, liquid metal or gas. The liquid-metal MHD powerplants use either one fluid or two.

The one-fluid liquid-metal concept operates in the following manner: Potassium, for example, could be boiled just as for the potassium-Rankine turbogenerating concept. Expansion of this vapor in a converging-diverging nozzle would produce a high-velocity stream of potassium vapor. Cold liquid potassium injected into this stream would condense the vapor and accelerate the liquid. This liquid potassium is highly conductive, and its useful energy could be extracted during its flow through an MHD generator.

In this concept, large losses occur in condensing the vapor and accelerating the liquid. The MHD generator has an efficiency (perhaps 0.65) that is substantially below that of a Rankine turbogenerator (perhaps 0.8). In addition, materials impose on the MHD concept the same temperature limits as on the Rankine-turbogenerator concept. For these reasons, this MHD concept has both lower efficiency and larger radiator area than the Rankine-turbogenerator system.

One concept for the two-fluid liquid-metal powerplant operates in the following way: into a heat stream of liquid lithium, liquid cesium is injected, the cesium boiling and producing a frothy mixture of cesium bubbles in liquid lithium. If the froth is passed through an MHD generator (ref. 104), the useful energy is provided by the expanding cesium vapor, and power extraction by the MHD generator depends on electrical conduction by the liquid lithium. Alternatively, the froth could be accelerated by expansion in a nozzle. The resulting high-speed stream of cesium gas and liquid lithium could be separated into its liquid and gaseous components by impingement onto, say, the apex of a cone. The cesium vapor is condensed for reuse, the liquid lithium passing through the MHD generator. Again, the MHD generator suffers in efficiency in comparison with a turbogenerator. Impingement of the high-speed

stream of liquid lithium onto the solid surface of the separator also imposes a considerable loss.

The MHD concepts that employ liquid metal in the MHD generator all suffer from low generator efficiency in comparison with a turbogenerator system at the same peak cycle temperature. In the future, however, the temperature limits imposed by reactor fuels and by compatibility with the alkali metals may exceed that imposed by strength required of the turbine rotor. If that situation comes to pass, then the competition between the liquid-metal MHD and turbogenerator concepts should be reexamined in order to determine if the higher temperature of the liquid-metal MHD concepts is sufficient to offset their performance deficit.

The gaseous MHD concepts operate on the Brayton cycle of the Brayton-turbogenerator concept but at higher temperature. Inasmuch as a conventional compressor is required for the Brayton-MHD powerplant, the accepted approach (refs. 105 and 106) uses a turbine to drive the compressor, as shown in figure 52. In that concept, peak cycle temperature was taken as 2500 K, and 2000 K is a representative generator-outlet temperature. For the assumed limit of 1500 K at the turbine inlet, the generator exhaust is cooled from 2000 to 1500 K in a turbine reheater in combination with a high-temperature recuperator. Heat is also recovered from the turbine exhaust in a low-temperature recuperator, and the cycle also includes two compressor intercoolers.

As shown by reference 105, the lightest turbo-MHD system has the same specific mass as the all-MHD concept but twice the powerplant efficiency; on the other hand, radiator area is one-third higher. In addition, the enthalpy drop required of the MHD generator is reduced from 37 to 18% (ref. 107), a change making the MHD generator significantly easier to develop.

Not only does addition of the turbine improve the gaseous MHD system but addition of MHD also improves the turbine concept. As shown by reference 105, addition of 2500 K MHD atop a 1500 K turbine powerplant raises powerplant efficiency from 0.21 to 0.40, reduces specific mass by 60% and decreases radiator area by 60%. The combination of turbine and gaseous MHD is thus especially attractive. The logical first step toward realizing this concept is development of a turbine powerplant for operation at 1500 K. The enabling technology for the inert-gas MHD concept should be explored concurrently but at a lower level of effort.

THERMIONIC POWER GENERATION

Reactor-thermionic power systems are of two basic types, namely, in-core and out-of-core. Locating the thermionic converters outside the reactor core avoids problems with the converters from fuel swelling and from neutron damage to electrical insulators but accepts the system complication of heat transport from the reactor to the thermionic converters. Performance of the thermionic converters depends almost entirely on temperature of the thermionic emitter (ref. 108) and is nearly independent of the heat source. However, the problems to be solved in bringing to fruition either the in-core or out-of-core concept are peculiar to that concept. The performance potential discussed herein relies on this temperature dependence and is considered applicable to either the in-core or out-of-core concepts.

Information in reference 109 formed the basis for figure 53. Performance of thermionic converters is very sensitive to emitter temperature, both power density and efficiency rising as emitter temperature is raised. Until 1973, the thermionic programs focussed on emitter temperatures of 1800 to 2200 K (ref. 110). More recently, interpretation of the data on reactor fuels has limited reactor-outlet temperature to perhaps 1500 K (ref. 2). This low ceiling on temperature has so limited thermionic performance that it is not competitive with that of concepts easier to develop (ref. 110).

The long-time performances of both reactor fuels and thermionic converters at higher temperatures are assessed in references 34 and 111. The impact of the higher temperatures on the performance of thermionic powerplants is shown by figure 54. Although efficiency and radiator area are both inferior to those for a Brayton system having the same peak temperature, the radiator area for a thermionic system at 2000 K is only 40% of that for a Brayton system at 1500 K and equal to that for a Rankine system at 1500 K (figs. 51 and 54).

The results of long tests of thermionic converters are summarized in table III (ref. 111). The first three converters were electrically heated, LC-9 performing stably at 1970 K for 46,647 hr (64 months). On the other hand, LC-3 shorted out in 10,406 hr. The other converters listed were all fueled, were tested in a nuclear reactor and received their heat from fissioning their nuclear fuel. The test articles 6F-2 and 6F-3 each contained six converters in a cylindrical stack much like a battery of cells in a 6-cell flashlight. Among the fueled converters, two of the seven devices operated successfully for periods up to 11,084 hr (15 months), the others failing for various reasons after periods ranging from 7685 to 12,534 hr (10-1/2 to 17 months).

Fuel temperature and converter lifetime thus remain outstanding problems in thermionic power generation.

POWER CONDITIONING

Power conditioning can have substantial impact on design of the power-generating system, and vice versa. In turn, power conditioning can markedly affect the evaluation and selection of competitive concepts for power generation. This stems from the inherent characteristics of the semiconducting silicon components of the power conditioners; specifically, such semiconducting switches have a forward voltage drop of 1 to 2 V. The product of this voltage drop and the current flowing is, of course, the power loss in the switch. For power generators producing low voltage, such power loss might range from 5 to 20% of the power generated. To this must be added other losses in power conditioning that might range from 2 to 5%.

The power loss itself is not the only penalty, for that loss must, of course, also be disposed of in space. The temperature limits on the semiconducting silicon components require a cold-plate temperature of about 100° C for effective cooling. At such a low temperature, the heat rejected per unit area of radiator is very low. In turn, solar radiation on that area can be a sizeable fraction of the heat to be rejected. In low orbits about the Earth, the Earth's albedo and thermal radiation from the Earth also interfere with thermal radiation at such a low temperature. An equivalent sink temperature of 250 K approximates these added thermal inputs to the radiator (ref. 112,

p. 4). Heat rejection for cooling the power conditioner can become a serious problem in its own right.

Reference 113 illustrates the problems that arise if attention in design is focussed on power generation and power conditioning is largely ignored. In that study of nuclear electric propulsion to the outer planets, the propulsion system was divided into power and thrust subsystems. The power subsystem comprised a nuclear reactor and its associated thermionic power generator. Power conditioning and the ion thrusters composed the thrust subsystem. In order that problems with a high-temperature electrical insulator operating at thermionic-emitter temperatures might be avoided, the 444 kWe output of the thermionic converters was at only 54 V dc. After deduction of voltage drops in the busbars, voltage at the input to the power conditioner was 45 V dc. Although this design approach reduced the problems in design of the power generator, the problems it produced in power conditioning should soon be evident.

The busbars leading from the thermionic converters to the power conditioner constitute an interface between the power generator and the power conditioner and could reasonably be aggregated with either subsystem. Although these busbars were lumped with the power subsystem in reference 113, I chose the more conventional aggregation of power transmission (busbars) and power conditioning.

Sans busbars, powerplant mass was 7322 kg and, corresponding to its net output of 444 kWe, its specific mass 16 kg/kWe (table IV). The mass of busbars was 1104 kg, and their power loss was 55 kW. For the power conditioner itself, its efficiency was taken as 0.88 in reference 113; accordingly, its loss was 46 kW. The complete powerplant, comprising the nuclear reactor, the reactor's shield, the power generator, the busbars, and the 1200 kg power conditioner, totalled 9626 kg. After subtraction of losses in the busbars and power conditioner, net powerplant output was 343 kWe. Accordingly, specific mass was 28 kg/kWe, 70% above the specific mass of the 16 kg/kWe for the incomplete powerplant.

Reference 113 acknowledges that specific mass of power conditioning for ion thrusters is currently about 10 kg/kWe of input power. This specific mass was arbitrarily reduced in reference 113 to 3 kg/kWe. Had the higher value been used, mass of the complete powerplant in table IV would have been 12,426 kg and its specific mass 36 kg/kWe, 220% of the specific mass of the bare powerplant. Power conditioning can, therefore, not only reduce powerplant output by a substantial amount but can also have a major impact on powerplant specific mass.

Radiator area is also affected by the need to reject the losses involved in power conditioning, the principal area being required to maintain the power conditioner at 100° C. For reference 113's 45 kW loss in the power conditioner, radiator emissivity of 0.9 and sink temperature of 250 K, ideal radiator area is 58 square meters. If real radiator area is 20% above this ideal, then 70 square meters are required for cooling the power conditioner, or 0.20 m²/kWe for the net output of 343 kWe.

To this 46 kW heat rejection must be added any heat flowing from the busbars into the power conditioner. The power loss in the busbars is 55 kW, and addi-

tional heat flows from the thermionic converters into the busbars, the busbars, of course, being connected (thermally as well as electrically) to both the thermionic emitters and the electron collectors. In reference 113, the emitter and collector temperatures are about 1650 and 1000 K, respectively, values having substantial thermal potential above the 373 K of the power conditioner. The total quantity of heat to be rejected is thus substantially greater than the 55 kW electrical loss in the busbars. In reference 113, some of this busbar heat is rejected by exposing the hot, bare busbars directly to space. But, by virtue of the fact that the busbars must be hot in order to radiate a substantial amount of heat to space, the busbars are also hot enough to conduct a substantial amount of heat into the power conditioner. Accordingly, reference 113 allots 100 square meters to the radiator area for cooling the power conditioner, about 40% above the area required for rejecting losses within the power conditioner itself. For the net power output of 343 kWe, this 100 m² radiator area corresponds to 0.29 m²/kWe for power conditioning alone. That area is so large that it equals the main radiator of a thermionic powerplant having thermionic emitter temperature of 1600 to 1650 K. (See fig. 54 herein as well as fig. 9 in ref. 113.)

This discussion reveals that the complete power system (including power conditioning) was not optimized in reference 113. Instead attention was focussed on the power generator alone, and that was a mistake. Powerplant design should thus always encompass the delivery of conditioned and regulated power. In turn, evaluation of competitive concepts for power generation should also include the power conditioning.

AIRCRAFT NUCLEAR PROPULSION

The technology required for aircraft nuclear propulsion is very different from that for nuclear power generation in space, for several reasons. First, the ubiquity of oxygen in the atmosphere excludes many high-temperature materials suitable for use in space. Metals to be considered are the oxygen-resistant superalloys much like those in current aircraft engines.

The aircraft needing nuclear propulsion will be large and will, therefore, require propulsive powers that are high in comparison with nuclear powerplants considered for use in space. For example, an airplane such as the Boeing 747 might require of the order of 200 to 300 MW of propulsive power for takeoff. A nuclear airplane would probably be heavier than a 747 and accordingly require even more power. In turn, reactor power might be of the order of 1 to 2 GW, approaching the power levels of reactors for stationary power. The power for cruising would depend greatly on the flight conditions selected but might be only 25% of that for takeoff. Then average power would be substantially below the peak. On the other hand, a military aircraft required to fly at, say, 0.9 Mach number at low altitude could require the same power for cruising as for takeoff. Reactor refueling might be required after perhaps 5000 to 10,000 hr of flight.

The reactor shield required would be heavier than for the usual space power systems because of the high reactor power and for several other reasons as well. Not only would people be fairly near the reactor but radiation scattering is also a significant problem. The sheer mass of the reactor and shield would constrain the airplane's center of mass to be fairly near the reactor. Thus, radiation scattering on massive portions of the airplane's structure

(such as the wing's) will result, and scattering from the surrounding air as well. During takeoff, landing, loading and servicing the aircraft, radiation will also be reflected from the surface of the airport.

Three basic, alternative approaches to extracting heat from the reactor might be considered, namely, air, alkali metal and inert gas. Passing high-pressure air from the engines through the reactor is a direct approach to heat removal, but the air ducts are large, and streaming of nuclear radiation through these ducts and past the reactor shield would be a problem.

The liquefied alkali metals are such effective reactor coolants that their coolant passages through the reactor's shield can be very small, thereby avoiding a critical problem of air cooling. From the viewpoint of heat transfer, lithium is the most effective of the alkali metals, but the lithium would freeze if its temperature ever dropped to 186°C (367°F), a problem requiring special consideration in design in order that the reactor and engines could always be restarted. NaK (the eutectic mixture of sodium and potassium) eases this problem by its low freezing point (-11°C , or 12°F), but the coolant passing through the reactor becomes very radioactive, a problem in radiation shielding. The alkali metals are also very sensitive to contamination by oxygen. For this reason, extreme cleanliness of the alkali-metal loop will be a critical problem in maintaining and servicing the aircraft. In addition, any fire involving the alkali metals would be unusually intense and difficult to stop.

High-pressure helium is, among the gases, an especially effective reactor coolant. The ducts carrying the reactor's coolant through the shield would be much smaller than the ducts for air-cooling, and the problem of radiation streaming past the shield would be greatly relieved accordingly. Many of the problems of the alkali metals would also be greatly diminished or eliminated, namely, the freezing, the requirement for extreme cleanliness, the neutron activation and the fire hazard.

The inert gases (a blend of helium and argon, for example) could be merely the reactor coolant or they might permit a completely different approach to design of the aircraft's nuclear powerplant. This high-pressure gas could be not only the reactor's coolant but also the working fluid for a closed Brayton-cycle powerplant. The turbines could drive either propellers or fans such as those in current turbofan engines. A waste-heat exchanger would be needed for rejecting heat from the powerplant to the aircraft's propulsive stream of air. The absence of oxygen in the working fluid would permit use of materials of construction not practical in the alternate nuclear aircraft engines, and thereby operating temperatures might be raised by several hundred degrees. Much of the technology from HTGR would be directly applicable to this class of high-temperature inert-gas aircraft powerplant.

Nuclear safety is always a critical issue in any consideration of aircraft nuclear propulsion, in part because of the conditions under which aircraft generally operate. Specifically, our aircraft fly near and even directly over population centers. Aircraft flight at high speeds and high altitudes also provides the energy for very energetic accidental impact, a critical problem for nuclear safety.

This raises the question whether or not conventional aircraft would be the most appropriate first application of nuclear propulsion to flight vehicles (refs. 114 and 115); the air-cushion vehicle (ACV) is an alternative with several advantages. In contrast with conventional aircraft, ACV's fly both low (3 to 6 m or 10 to 20 ft) and slow (100-160 km/hr, or 60-100 mph) while supported on a cushion of low-pressure air that slowly escapes past a flexible skirt surrounding the vehicle's periphery. In concept, such vehicles would transport cargo along selected corridors away from population centers. The low speeds, low altitudes and selected flight paths would all ease the basic question of safety of nuclear-propelled aircraft. In addition, these ACV's might have gross weights of 10,000 - 20,000 tons, over 20 times the mass of the largest current aircraft. The large size of these vehicles would readily tolerate the mass of reactor shielding required as well as design features to protect the reactor against accidental vehicle impact. Because of the broad relief of problems in design and nuclear safety provided by ACV's, nuclear propulsion in this airborne application might be the logical antecedent of full-blown aircraft nuclear propulsion.

CONCLUDING REMARKS

This survey paper focusses on power generation in nuclear powerplants for use in space. In particular, the states of technology of the principal competitive concepts for power generation were assessed. The possible impact of power conditioning on power generation was also discussed. For aircraft nuclear propulsion, the suitability of various technologies was cursorily assessed in the context of the service conditions imposed by flight in the Earth's atmosphere; a program path was suggested to ease the conditions of first use of aircraft nuclear propulsion. The following observations and conclusions were drawn:

1. The intended lives of space nuclear powerplants (5 to 10 years and more) are so long that a life demonstration before committing to actual use in space is impractical. Despite that, a life test for a substantial period (1 to 2 years) is necessary before committing the concept to actual use. A successful life test could then be continued at only modest cost.
2. In the absence of a complete life demonstration, a firm base of constituent and component technologies must be relied on for life projection. Although this base of enabling technology is a sine qua non for nuclear space power, current programs look forward to simple demonstration without adequate substantiating technologies.
3. Several alternate concepts for power generation compete for use, each concept having outstanding problems. For concepts having a history of single-point failures, tolerance of failure is claimed with yet inadequate consideration of the consequences of failure.
4. Because of the unsettled state of the technologies, the assessments of the various concepts for nuclear power generation vary widely, depending on personal judgment instead of firmly based technologies.
5. If a nuclear-power concept selected for demonstration has basic problems yet unresolved, these problems might not become obvious for 6 to 8 years into the program.

6. The cure for such future programmatic calamities is a firm base of constituent and component technologies. These enabling technologies can provide both increased assurance of success and design approaches alternate to that selected. An adequate precursor base of technology will both shorten the period of demonstration prior to mission use and save money in the long run.

7. Once successfully developed, a nuclear powerplant will have a wide range of application in space, in part for its low cost but primarily for the mission flexibility it confers.

8. The mission manager's continuing selection of non-nuclear space powerplants is only partly a result of the currently unsettled state of technology for nuclear power. Principally, successful mission managers have a congenital distrust of their dependence on the arcane. A nuclear-power program that successfully achieves its goals for performance, cost and schedule would contribute substantially toward dispelling this distrust. The necessity of this programmatic success places special emphasis on items (5) and (6) above.

9. In assessing competing concepts for power generation, efficiency of power generation is an important measure of merit because of its effect on reactor life. Inasmuch as each reactor is limited in the energy it can produce during its lifetime, high efficiency of power generation results in production of a large amount of useful product, namely, kilowatt-hours of electric energy. If efficiency is doubled, this could be exploited to either double the power generated or to double reactor life.

10. Research on high-temperature materials can likely expand the capabilities of future nuclear power systems. Both ceramics and tungsten alloys promise performance gains, but carbon-carbon composites in combination with high-temperature gas-cooled reactors offer the greatest potential.

11. Because of design interactions between power generation and power conditioning, evaluation of concepts for power generation must include the required power conditioning. Ignoring power conditioning can double the radiator area required and double specific mass of the powerplant.

12. For nuclear propulsion of aircraft, the problems presented by atmospheric oxygen, by the power levels and by nuclear safety require technologies markedly differing from those for nuclear space power. Air-cushion vehicles might be the first class of aircraft to which nuclear propulsion can be usefully applied.

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TABLE I. - WEIGHT BREAKDOWN FOR
GARRETT-JPL BRAYTON CONCEPT

[Power output, 400 kWe; redundancy of power
generation, 100%; turbine inlet, 1500 K.]

Component mass, kg -	
Reactor	850
Reactor shield	800
Heat-source heat exchanger (2)	420
Rotating machinery (2)	430
Alternators' radiator	150
Recuperators (2)	730
Waste heat exchangers (2) and radiator	4040
Ducting and miscellaneous	400
Power processing	150
Structure	<u>300</u>
Total	8270
Specific mass, kg/kWe	21

TABLE II. - WEIGHT BREAKDOWN
FOR RANKINE CYCLE

Component mass, kg -	
"LANL" reactor	957
"LANL" shield	818
Primary loop and boiler	281
Power generation	1270
Turbogenerator	590
Condenser	156
Feed pump	136
Other	388
Main radiator loop	951
Alternator cooling	181
Turbine cooling	50
Electrical equipment	546
Electronics cooling	218
Structure (at 10%)	<u>527</u>
Total	5799
Power output, kWe	404
Specific mass, kg/kWe	14

TABLE III. - SUMMARY OF THERMIONIC TESTS

[Fueled-converter success rate, 2/7.]

Device	In pile or out	Fuel	Temperature, K	Power, W/cm	Duration, hr	Result
LC-9	Out	None	1970	8.0	46,647	Operable
LC-11	Out	UC-ZrC	1870	6.5	18,569	Operable
LC-3	Out	W-UO ₂	2000	8.9	10,406	Short
I-4	In	UC-ZrC	1840	6.0	8,754	Leak
C11	↓	UO ₂	1840	3.4	7,881	Crack
2E-1		UO ₂	1820	4.0	12,534	Swelling
2E-2		UO ₂	1820	5.6	11,084	Operable
1F-1		UC-ZrC	1780	4.0	8,560	Swelling
6F-2		UO ₂	1780-1950	2.3	7,685	Cracks
6F-3		UO ₂	1740-1820	3.0	8,062	Operable

TABLE IV. - EFFECT OF LOW GENERATED
VOLTAGE ON POWER CONDITIONING

[JPL study: 444 kWe at 54 V dc.]

	Mass, kg	Power, kW	kg/kWe
Bare power system	7322	444	16
Busbars	1104	55	
Power conditioning	1200	46	
Complete system	9626	343	28

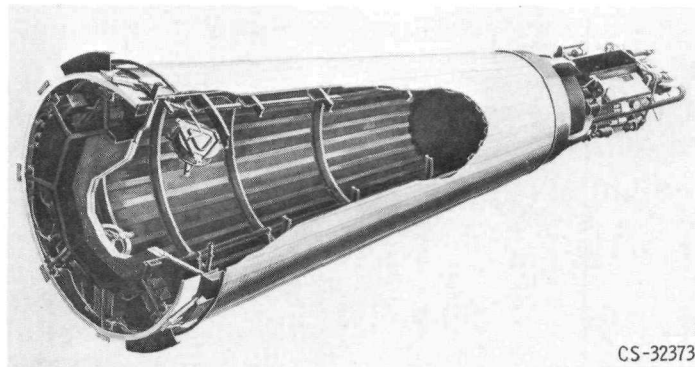


Figure 1. - SNAP 10A system.

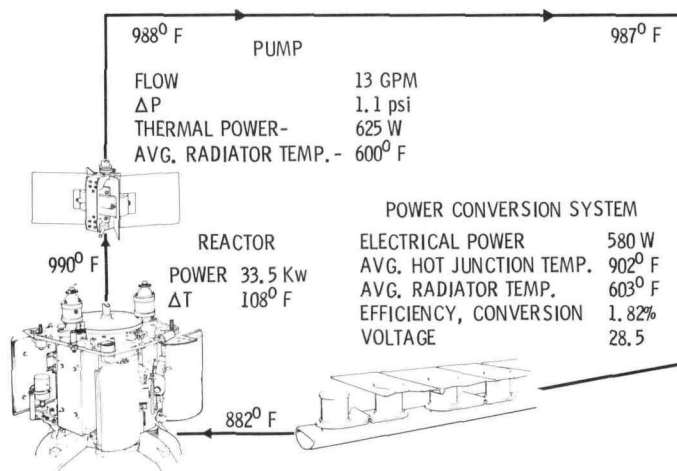


Figure 2. - SNAP 10A thermodynamic cycle.

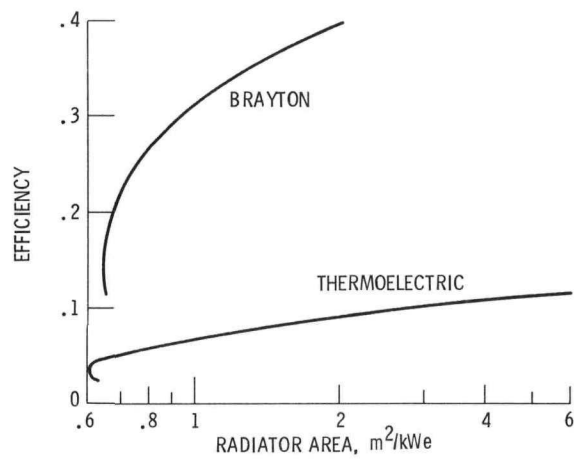


Figure 6. - Brayton cycle at 1300 K.

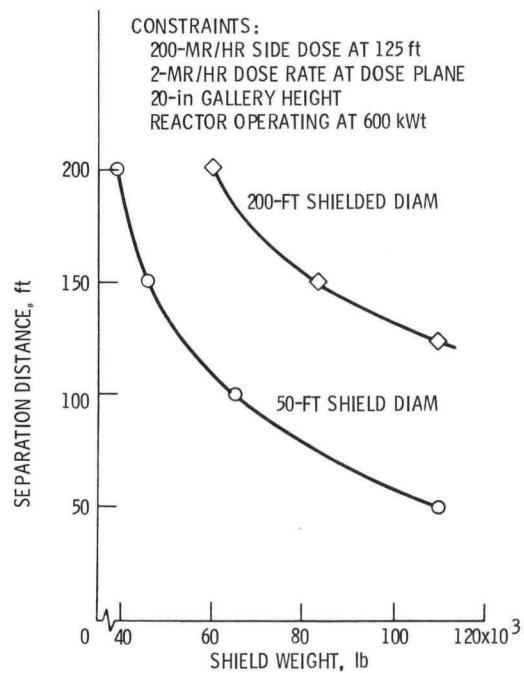


Figure 7. - Reactor shield weight.

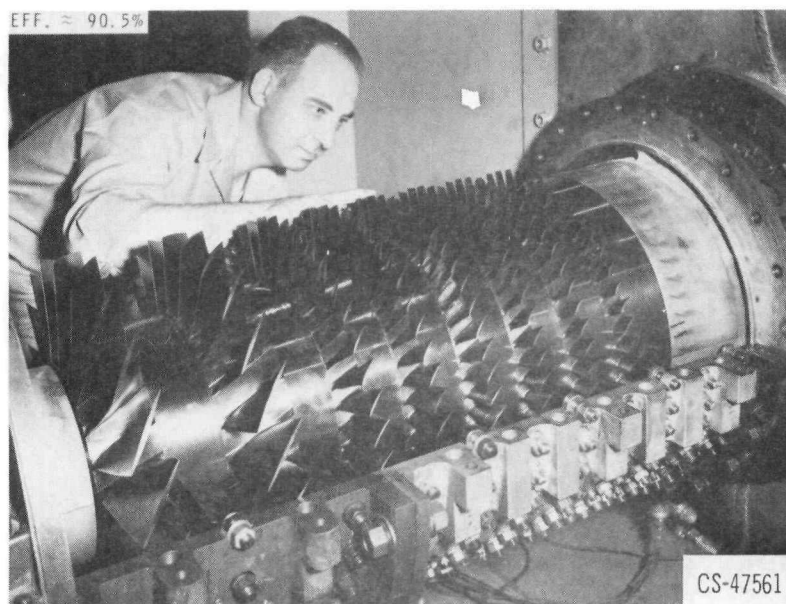


Figure 8. - Research compressor.

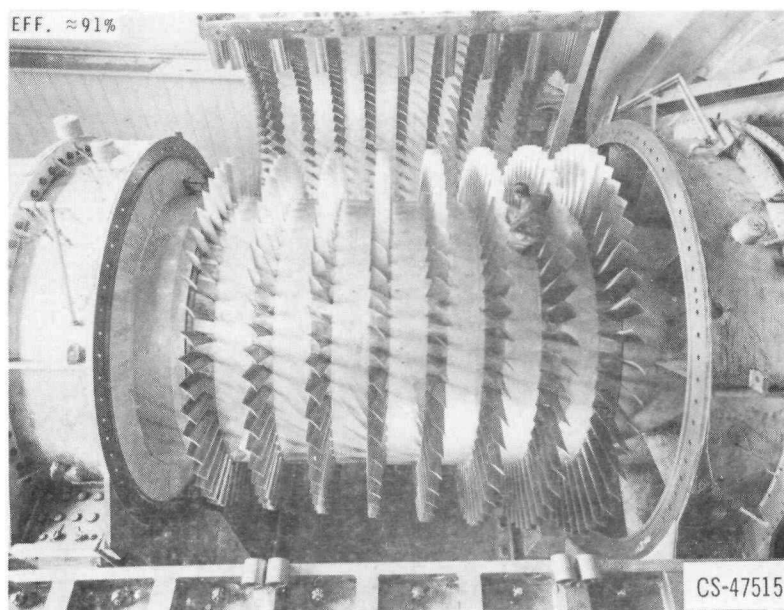
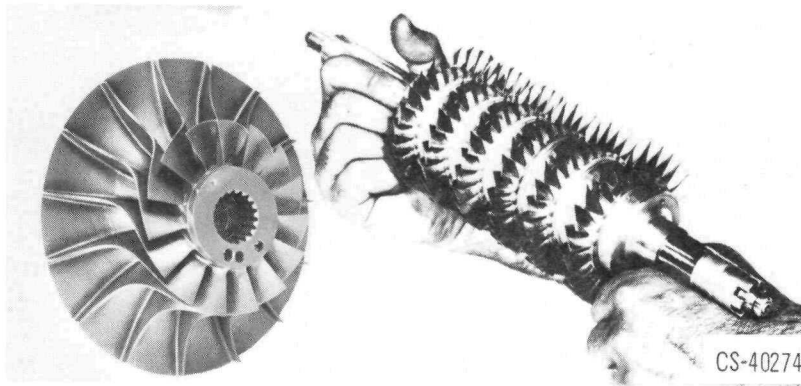


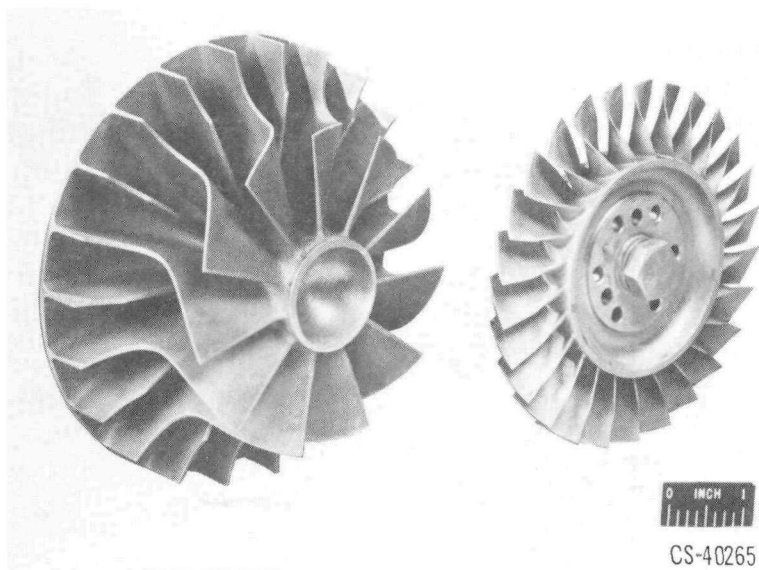
Figure 9. - 20 ft. diameter compressor.



(a) Centrifugal.

(b) Axial.

Figure 10. - Brayton-cycle compressors.



(a) Radial.

(b) Axial.

Figure 11. - Brayton-cycle turbines.

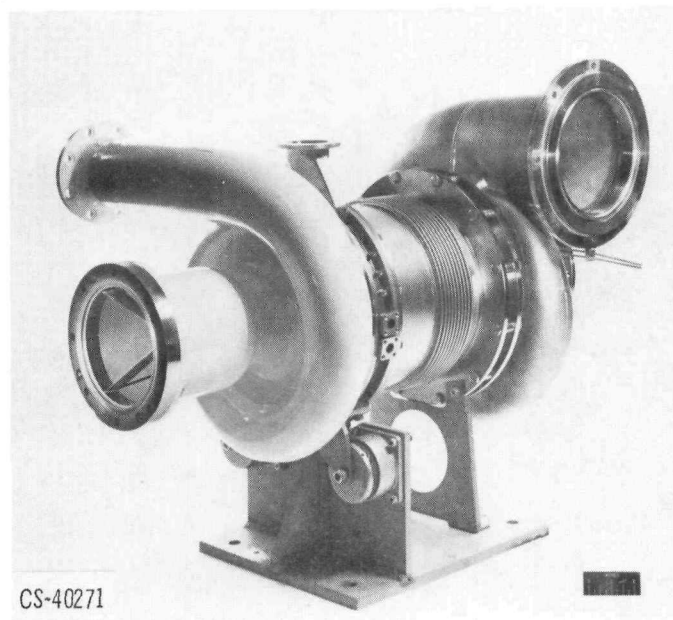


Figure 12. - Research turbocompressor.

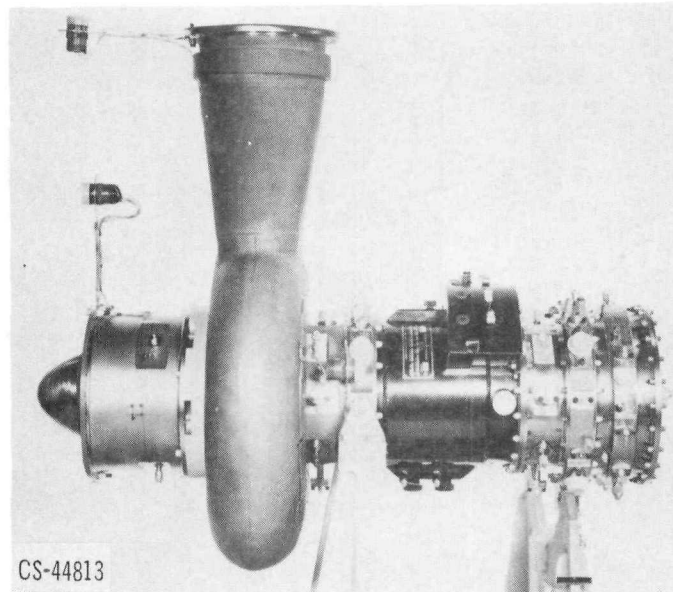
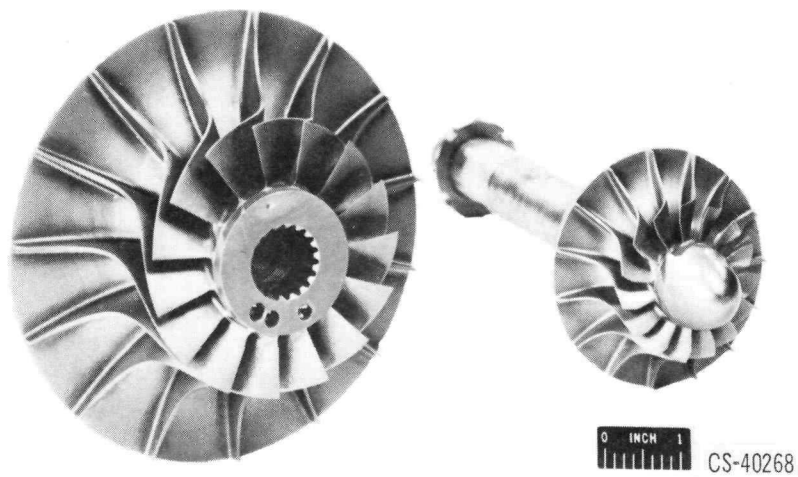


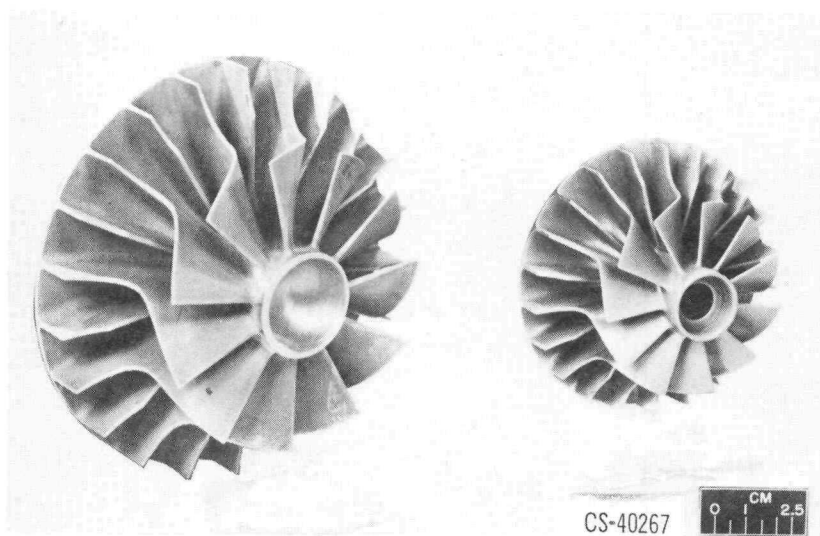
Figure 13. - Turboalternator.



(a) 6.0-Inch diameter.

(b) 3.2-Inch diameter.

Figure 14. - Compressors for size effect study.



(a) 6.0-Inch diameter.

(b) 3.2-Inch diameter.

Figure 15. - Turbines for size effect study.

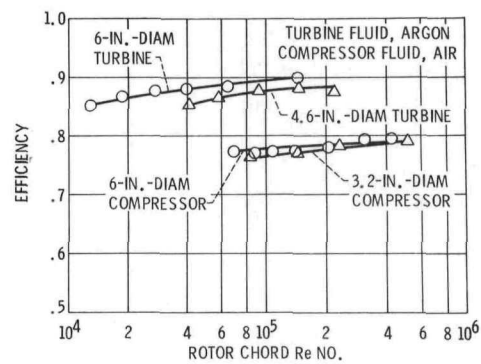


Figure 16. - Effects of Reynolds number and size on efficiency.

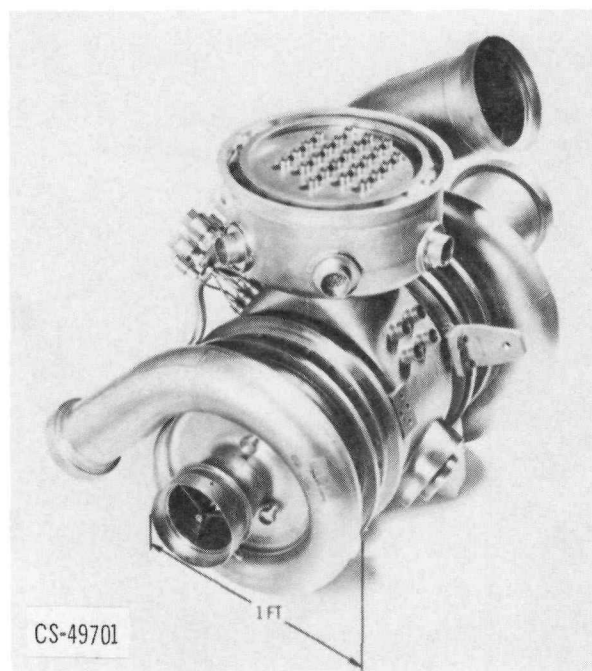


Figure 17. - Brayton rotating unit.

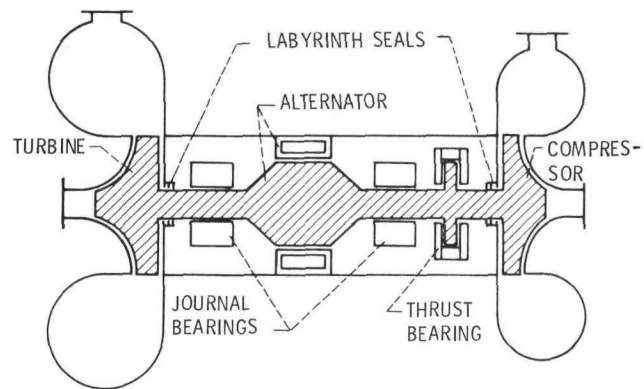


Figure 18. - BRU schematic.

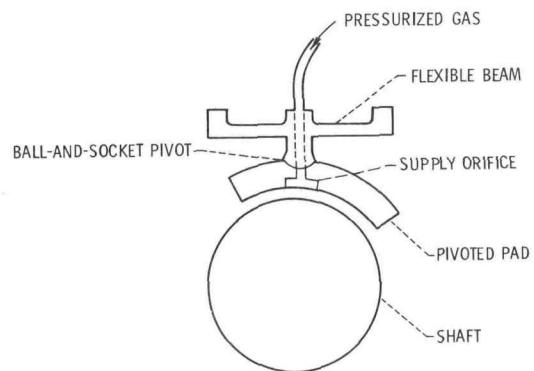


Figure 19. - Pivoted-pad gas bearing.

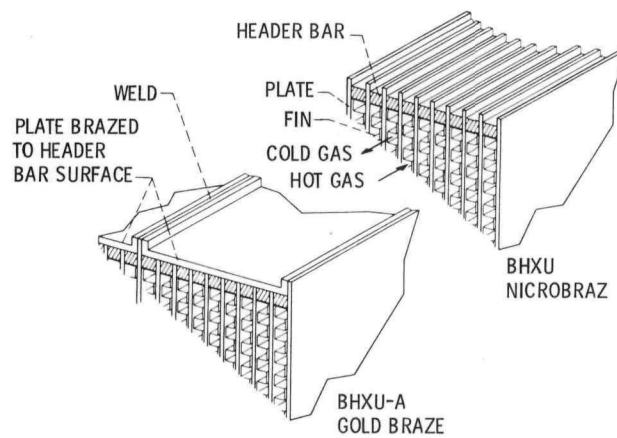


Figure 20. - Recuperator construction.

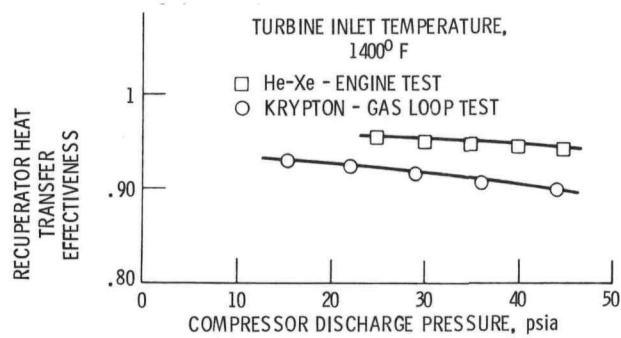
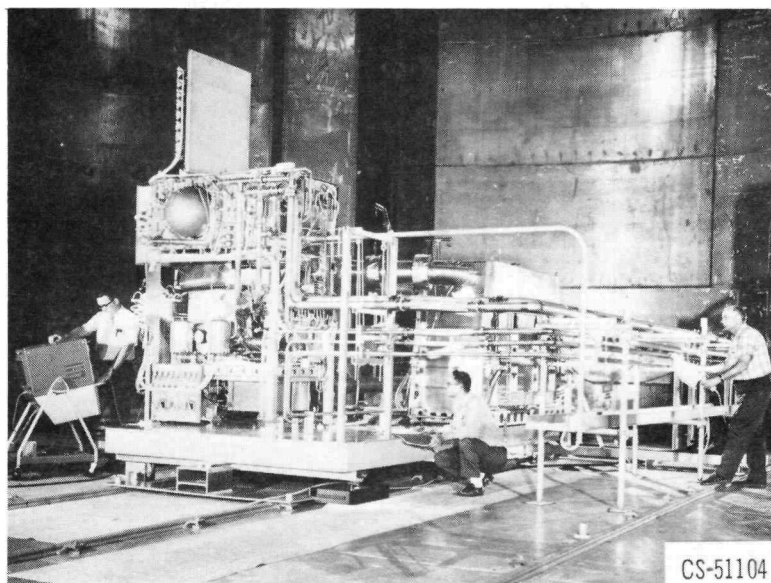


Figure 21. - Effect of working fluid.



CS-51104

Figure 22. - Brayton system in SPF.

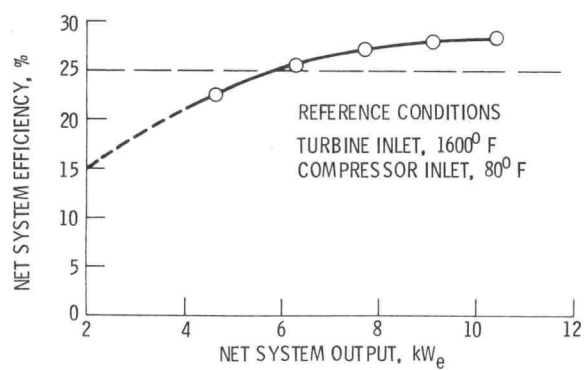


Figure 23. - Brayton engine performance.

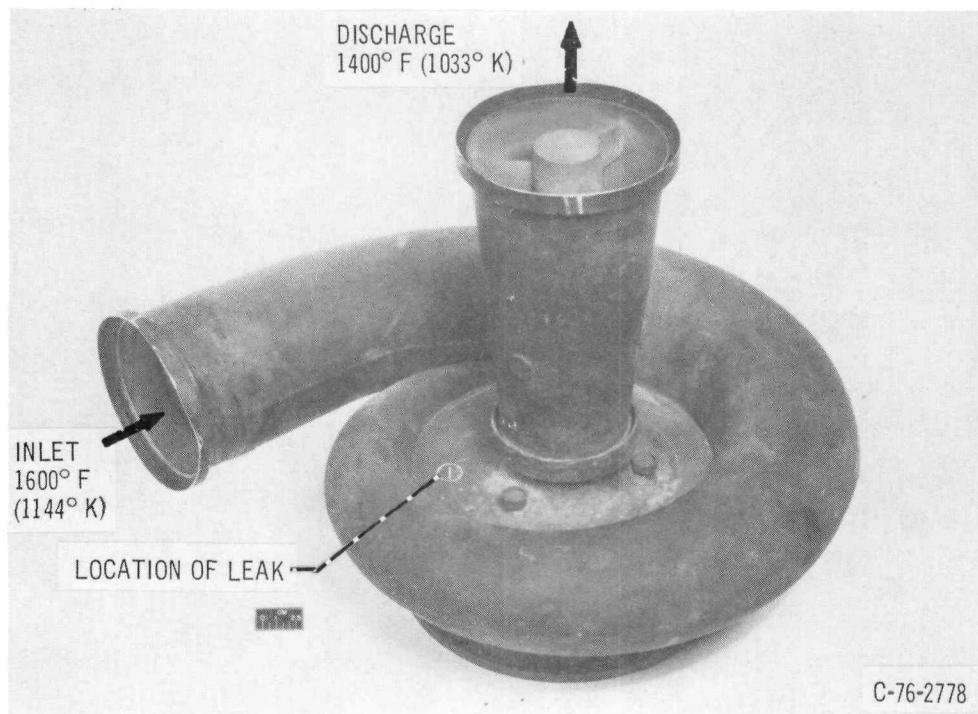


Figure 24. - Turbine scroll Brayton rotating unit No. 2.

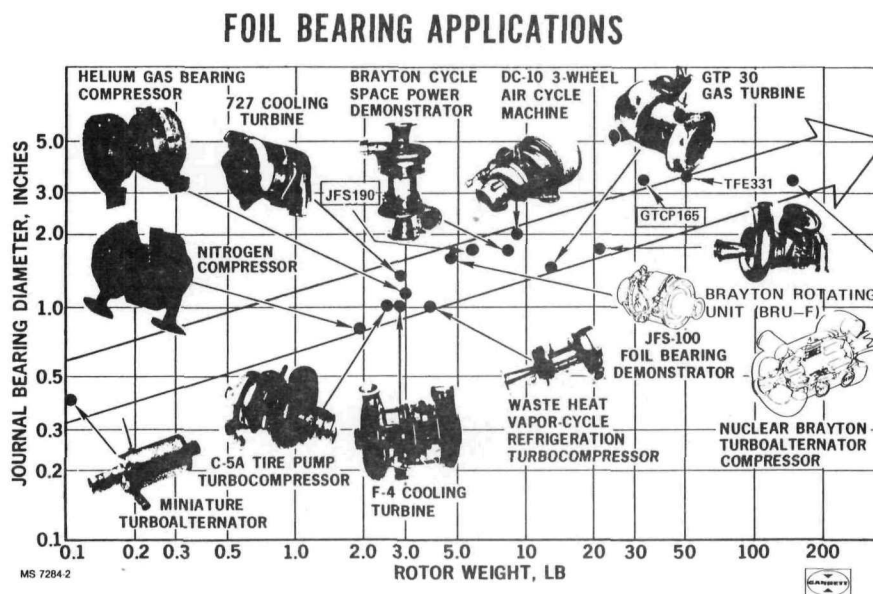


Figure 25. - Foil bearing applications.

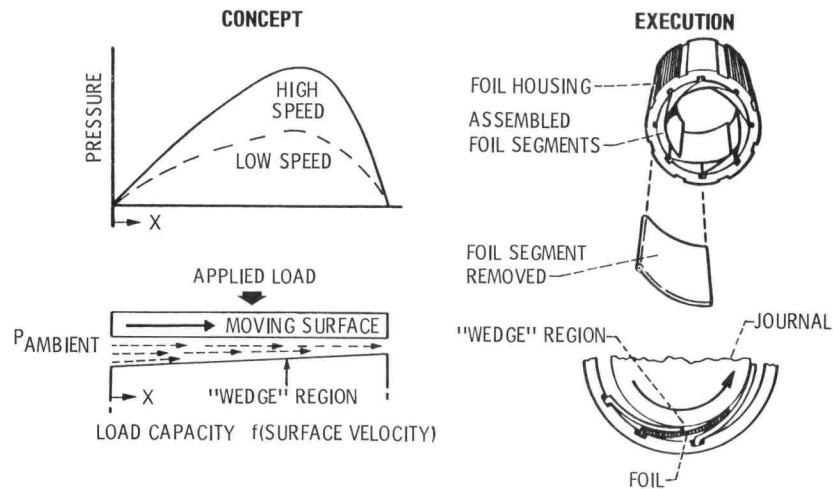
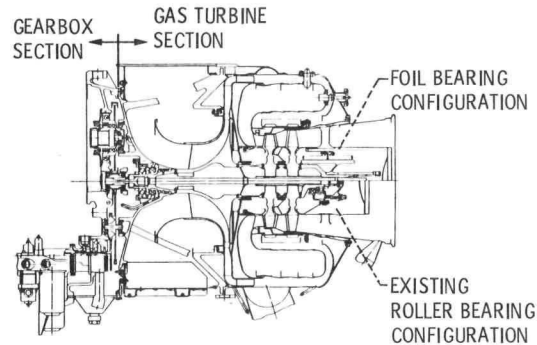


Figure 26. - Foil bearings are self-acting.



BEARING DESIGN

- JOURNAL-DIAMETER X LENGTH = 3.5 x 4.2 in.
- BEARING DN = 3.45×10^6
- NUMBER FOILS = 8
- FOIL COATING = TITANIUM CARBIDE
- JOURNAL COATING = KAMAN SCA*

BEARING REQUIREMENTS

- ROTOR SPEED = 38 800 rpm
- ROTOR WEIGHT = 32.3 lb
- TURBINE-END LOAD = 22 lb
- MAX BEARING TEMPERATURE = 790° F

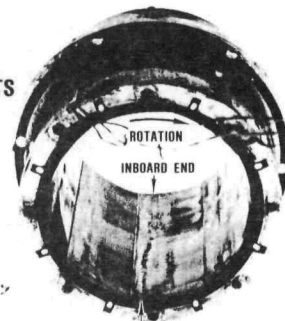
*CHEMICALLY ADHERENT OXIDE COMPOSITE CERAMIC COATING CONTAINING Al_2O_3 , SiO_2 , AND Cr_2O_3 .
APPLIED BY KAMAN SCIENCES CORP., COLORADO SPRINGS, COLORADO.

Figure 27. - Garrett GTCP 165-series gas turbine engine with turbine - end foil bearing.

200-HOUR ENDURANCE TEST OF FOIL BEARING-EQUIPPED GTCP 165

OUTSTANDING SUCCESS

- BEARING IN EXCELLENT CONDITION AFTER TEST
- 187 TOTAL STARTS ACCOMPLISHED DURING TEST
- SAME BEARING USED IN SUBSEQUENT ENGINE TESTS
- ALL ENGINE AND BEARING PERFORMANCE PARAMETERS CONSTANT THROUGHOUT TEST
- FOIL BEARING SURVIVED TEMPORARY LOSS OF COOLING AIR SUPPLY [LINE BROKE]
- FOIL BEARING SURVIVED LONG PERIODS OF HIGH ENGINE VIBRATION [GEARBOX INDUCED]
- FOIL BEARING UNSCATHED WHEN ENGINE COMPRESSOR END BEARING FAILED



OUTBOARD END
VIEW OF BOTTOM SECTOR OF BEARING
POST-TEST CONDITION
OF BEARING



MS 7284-11

Figure 28. - 200-hour endurance test of foil bearing-equipped GTCP 165.

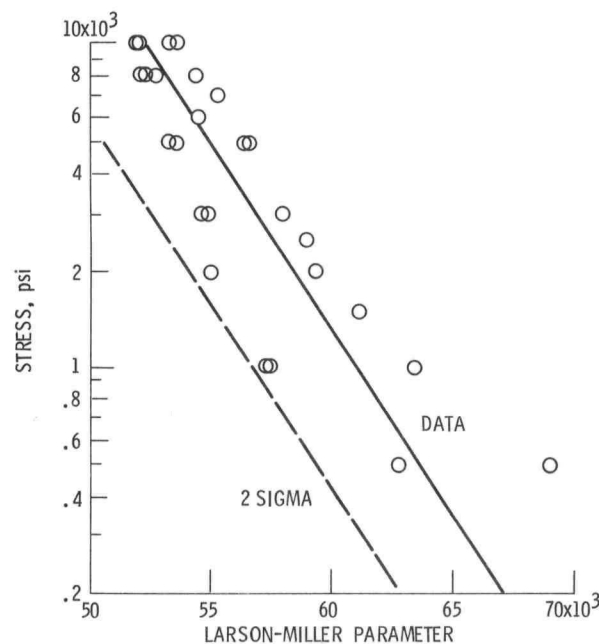


Figure 29. - One-percent creep of ASTAR-811C, log plot.

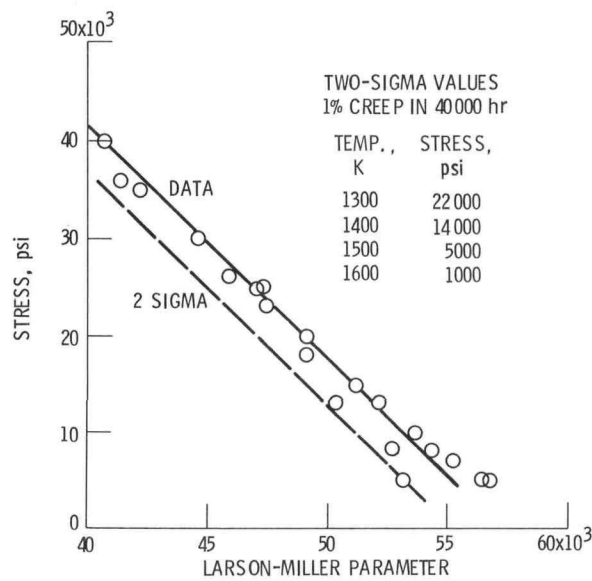


Figure 30. - One-percent creep of ASTAR-811C, linear plot.

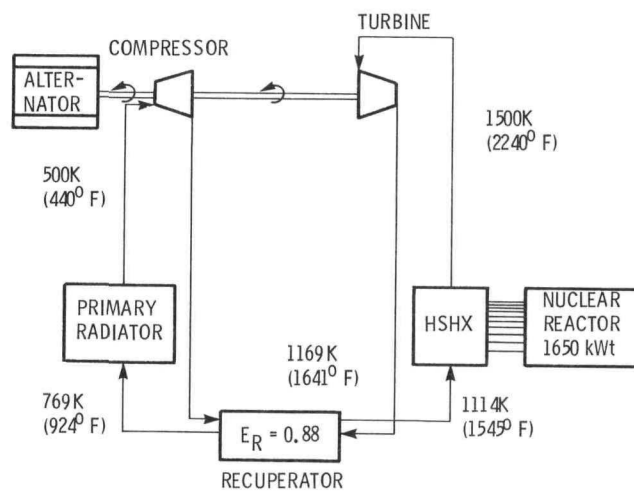


Figure 31. - Garrett-JPL Brayton concept for 400 kWe.

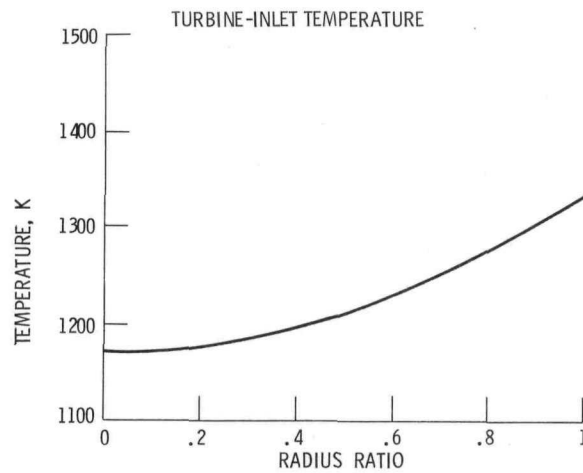
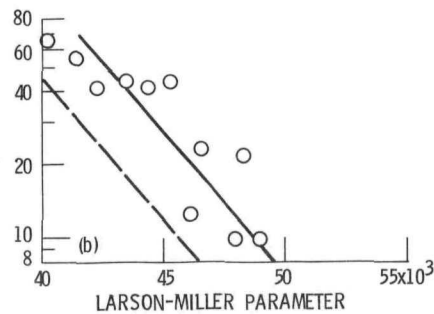
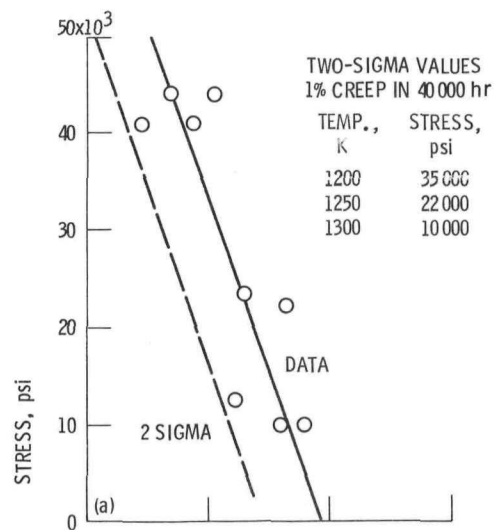


Figure 32. - Brayton-rotor stagnation temperature.



(a) Linear correlation.

(b) Logarithmic correlation.

Figure 33. - One-percent creep of moly alloy TZM. Eleven tests, 94140 hr.



Figure 34. - Ceramic turbine rotor for automotive gas turbines.

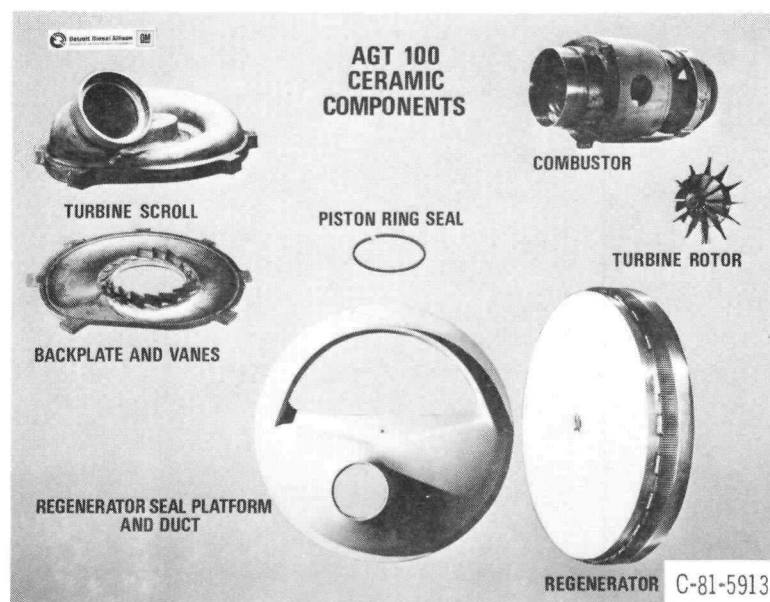


Figure 35. - Ceramic parts for automotive gas turbine.

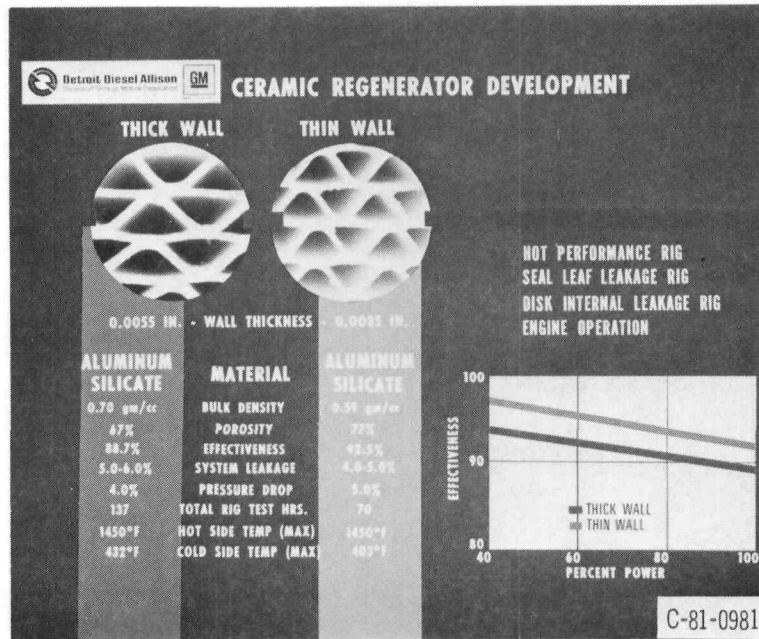


Figure 36. - Glass-ceramic regenerator for automotive gas turbine.

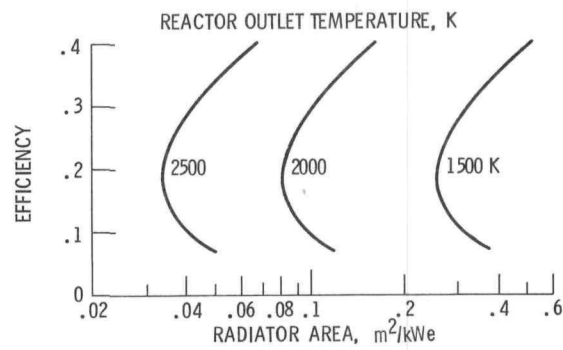


Figure 37. - Brayton-cycle performance at high temperature.

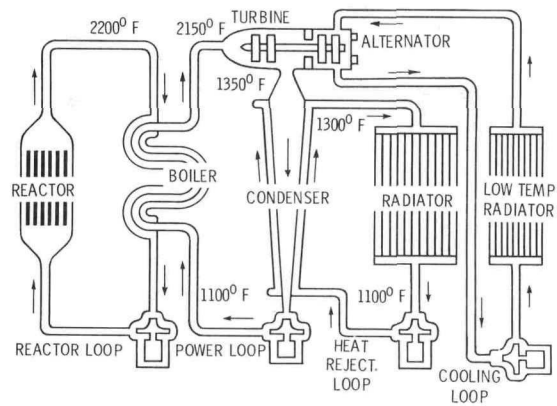


Figure 38. - Potassium - Rankine system.

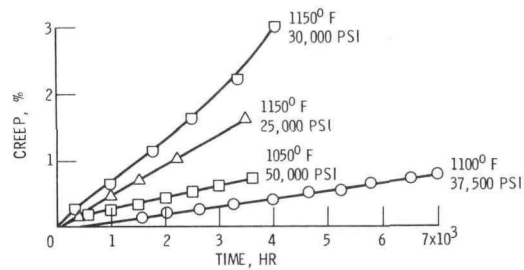


Figure 39. - Creep of Nivco.

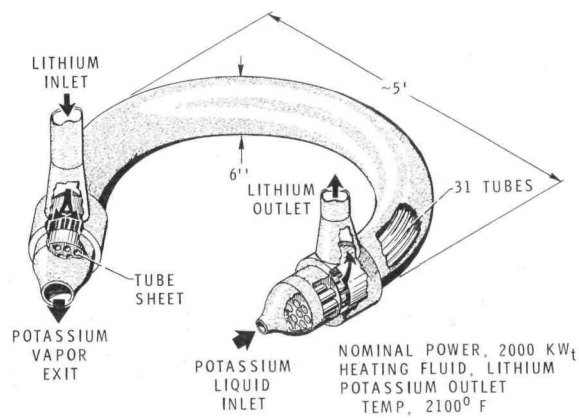


Figure 40. - Potassium boiler.

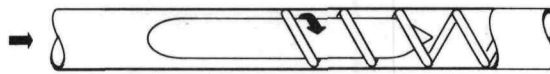


Figure 41. - Center plug and spring boiler-inlet insert.

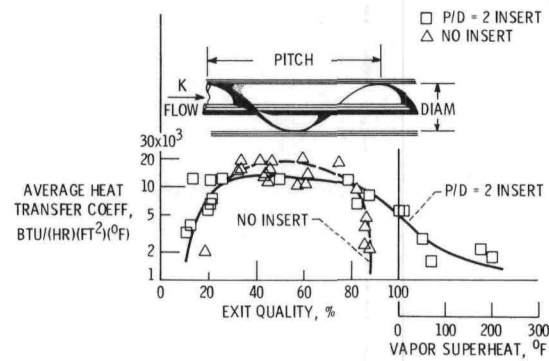


Figure 42. - Boiling-K heat transfer.

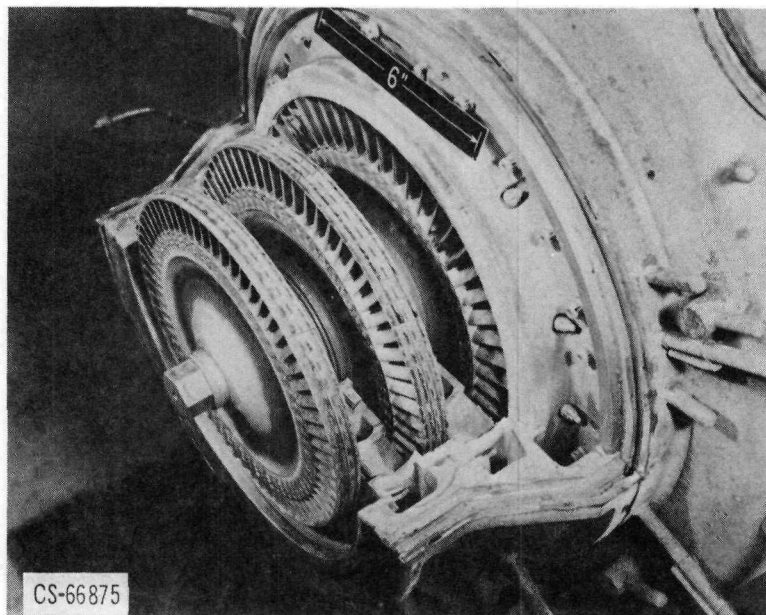


Figure 43. - Three stage potassium turbine after 5000 hours operation.

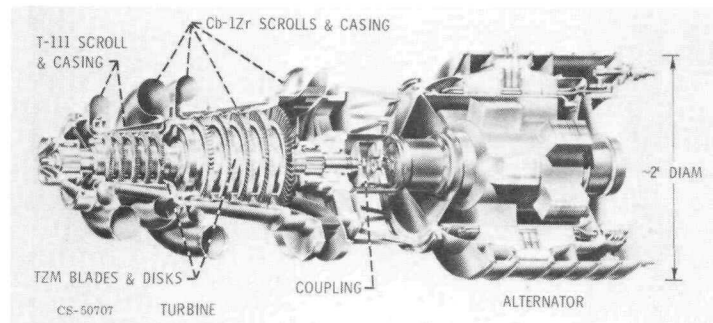


Figure 44. - Potassium turbolternator (KTA), preliminary design.

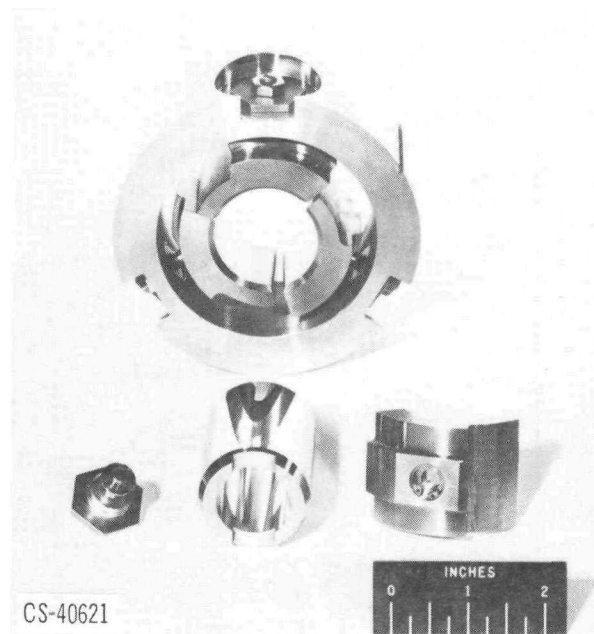


Figure 45. - Pivoted pad bearing assembly.

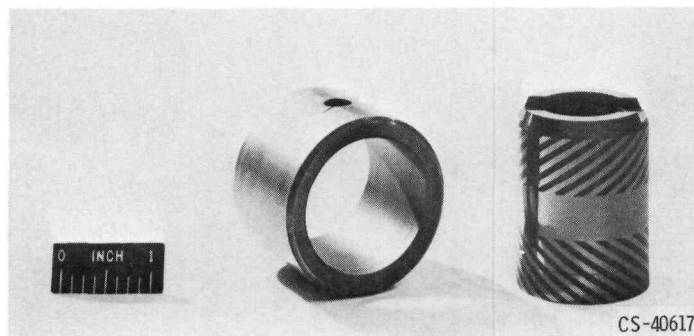


Figure 46. - Herringbone groove bearing.

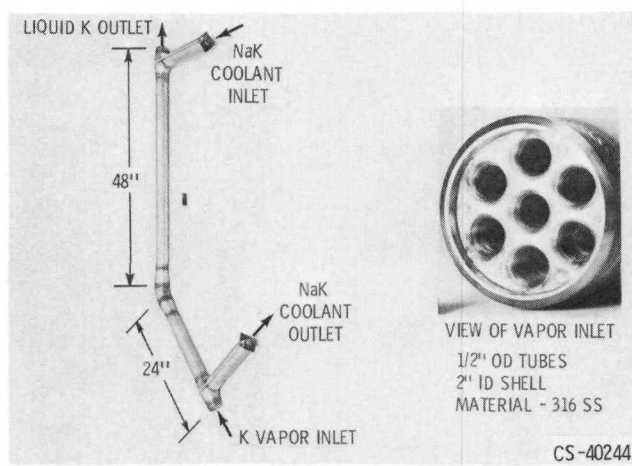


Figure 47. - Seven-tube K condenser.

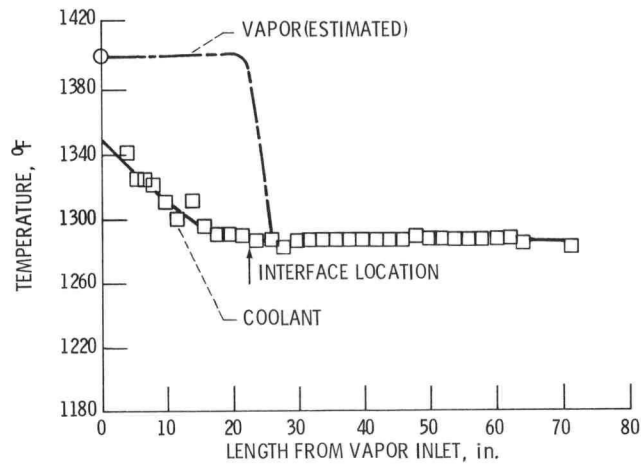


Figure 48. - Condenser temperature profiles.

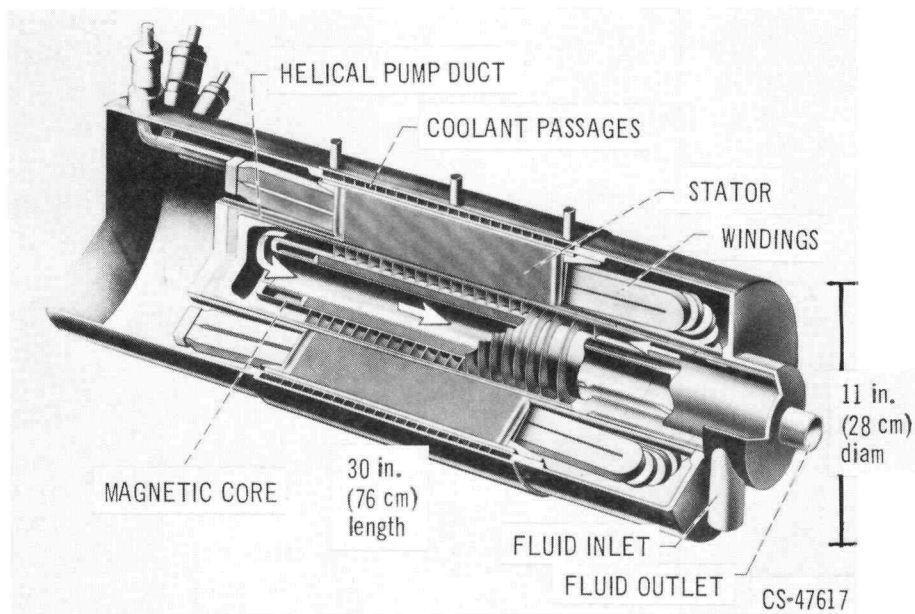


Figure 49. - Electromagnetic helical induction pump - boiler feed.

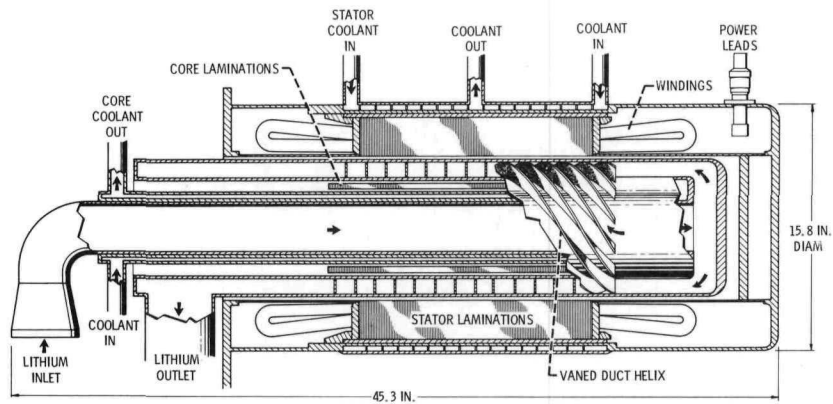


Figure 50. - Electromagnetic boiler - feed pump details.

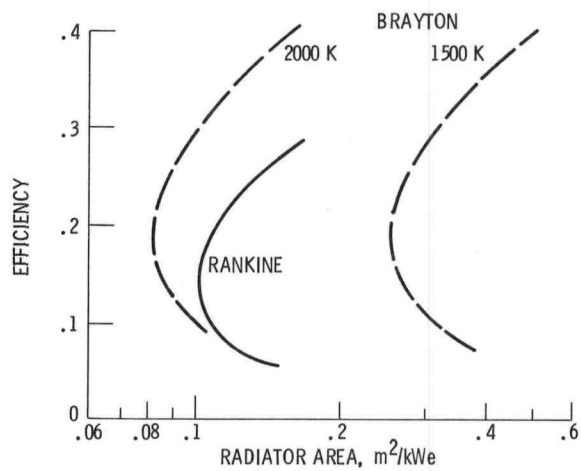


Figure 51. - Rankine cycle at 1500 K.

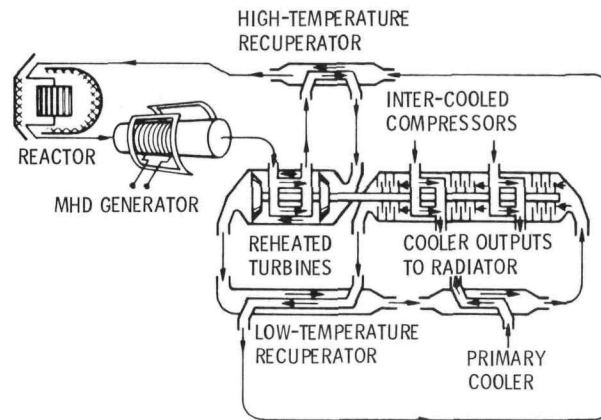


Figure 52. - Turbo-MHD power system.

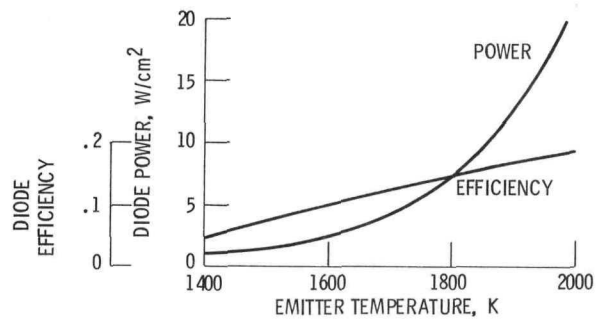


Figure 53. - Effect of temperature on thermionic performance. $\phi_c = 1.5$ eV, $V_d = 0.4$ V, $F_A = 1.75$.

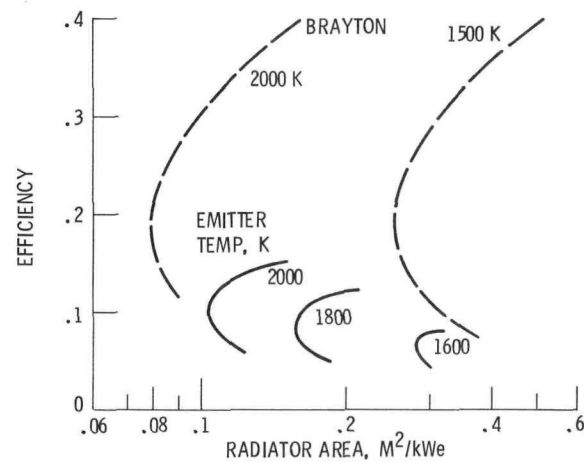


Figure 54. - Performance of thermionic concept. $\phi_c = 1.5$ eV, $V_d = 0.4$ V, $F_A = 1.75$.

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16. Abstract This survey paper focuses on power generation in nuclear powerplants in space. In particular, the states of technology of the principal competitive concepts for power generation are assessed. The possible impact of power conditioning on power generation is also discussed. For aircraft nuclear propulsion, the suitability of various technologies is cursorily assessed for flight in the Earth's atmosphere; a program path is suggested to ease the conditions of first use of aircraft nuclear propulsion.			
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