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SNAP PROGRAMS
SUMMARY REPORT

AEC Research and Development Report

Atomics International Division
Rockwell International

P.O. Box 309
Canoga Park, California 91304

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**SNAP PROGRAMS
SUMMARY REPORT**

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ABSTRACT

This report is a summary of the overall SNAP Reactor Programs, covering the period from 1951 through 1973. The following programs are reviewed: SNAP 2, SNAP 4, SNAP 8, SNAP 10A, 5-kwe Program, and the SNAP Reactor Aerospace Safety Program.

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I. INTRODUCTION AND HISTORICAL SUMMARY

A. SNAP FOR SPACE

1. SNAP Reactor Historical Background

The Systems for Nuclear Auxiliary Power (SNAP) compact hydride U^{235} reactor concept was conceived at AI in 1951, as the result of contact with RAND Corporation, which indicated a military requirement for a reactor heat source for earth satellite applications. In April 1951, RAND Corporation published Report R-217, "Utility of a Satellite Vehicle for Reconnaissance," which concluded that either nuclear or solar energy sources were required as a power source. Their preference for radioisotopes over a nuclear reactor was based on what they considered the strategic nature of U^{235} and what they also considered the inherent difficulty of controlling a reactor at the high temperatures (in the order of $1000^{\circ}F$) deemed necessary for an efficient heat cycle.

Following this report, AI initiated a company-funded review of possible reactor systems for satellite applications. These studies were completed in 1953, and indicated that a thermal reactor, moderated with zirconium hydride, fueled with fully enriched uranium, and reflected with beryllium, had promise as a compact, high-temperature heat source.⁽¹⁻³⁾

This concept differed in two ways from that considered in the RAND report. First, the use of the metal hydride of zirconium as the moderating material resulted in a very compact reactor core. Also, the zirconium hydride and NaK coolant could be operated at high temperature with a low core pressure, resulting in a low-weight system. Secondly, these early calculations showed the system to have a negative temperature coefficient of reactivity, which could permit relatively easy control.

The first suggestions for the use of metal hydrides for nuclear reactor applications came from the Nuclear Energy for Propulsion of Aircraft (NEPA) Program at Oak Ridge in 1948. The entire emphasis was placed on the use of these materials for shielding. The use of hydrides as a moderator was mentioned but discarded, due to the poor thermal and mechanical properties of the pressed powder bodies.⁽⁴⁾

In the time period from 1950 to 1953, Sylvania Electric Co. made a few cold-pressed and sintered, and hot-pressed zirconium hydride bodies for AI and for the Aircraft Nuclear Propulsion (ANP) materials research program. AI started research on hydrides in early 1952, as part of the AEC-sponsored work on materials which might have applications in nuclear reactors. AI continued this work through 1956, when the SNAP Program was initiated.^(5,6)

By August 1953, AI, utilizing the available information on the moderating properties of zirconium hydride, had started the preliminary design study of a zirconium hydride - uranium oxide powder compacted reactor fuel for a satellite power source. This work, which was done with company support, was published in 1954.⁽⁷⁾ The patent granted for this fuel is No. 2,929,707.

Interest in nuclear auxiliary power systems concepts continued to grow within the Department of Defense and the AEC; and, in 1955, a formal request for proposals was issued jointly by the AEC and the Air Force - Wright Air Development Center (WADC) (Dayton, Ohio). This request was for a study of possible sources of electrical energy (1 to 10 kwe range) with minimum weight and the capability of continuous unattended operation for 1 year in space. The request was specifically concerned with an auxiliary power supply system for an Air Force project.

AI proposed⁽⁸⁾ a homogeneous zirconium-yttrium-uranium hydride fuel composition, using sodium as the coolant, and calcium hydride (CaH_2) as the reflector. The core comprised a plurality of concentric fuel-moderator regions with nonuniform fuel distribution to achieve minimum weight. The sodium coolant entered the reactor at 700° F, was discharged at 1200° F, and transferred its heat to mercury, the working fluid in the power conversion system operating on the Rankine cycle. This reactor was the first design proposing the use of massively hydrided zirconium-uranium alloy fuel elements, coupled with liquid metal coolant for efficient heat transfer, to achieve a small, lightweight reactor. The SNAP 2, SNAP 8, and SNAP 10A systems were developed from this design.

The first SNAP critical assembly (SCA) was designed and operated in 1957.⁽⁹⁾ In 1958, AI was the first to be successful in utilizing a ceramic-coated cladding to retain hydrogen in metal hydrides at high temperature. In November 1959, the SNAP Experimental Reactor (SER) was operated at full power with an outlet

temperature of 1200°F. The reactor operated for 1 year, and proved to be the highest-temperature long-lived reactor in the free world.^(10,11) Results from the SER Program led to the design and fabrication of the SNAP 2 Development Reactor (S2DR). Operation of the S2DR started in 1961, and continued through the end of 1962.⁽¹²⁾

2. SNAP Reactor Thermoelectric Concept History

In April 1958, the Air Force, in a letter to the AEC Division of Reactor Development, expressed interest in a reactor using thermoelectric conversion with an electrical power capability of 200 w. At the fifth meeting of the Joint Air Force Ballistics Missile Division (AFBMD) - AEC SNAP Coordinating Committee in October 1958, the Air Force representative outlined the characteristics of a small nuclear power supply which would be of interest to AFBMD.⁽¹³⁾ An electrical output of several hundred watts or more was desired. It was indicated that, at this power level, the heat could be rejected directly to space from the outer surface without circulating fluids, the outer radiating surface forming the cold junction of a static thermoelectric or thermionic converter. Orbital startup would obviate all nuclear radiation problems during ground handling, and would permit unrestricted personnel access to the vehicle. Operation during the difficult ascent conditions would be circumvented. Startup would be delayed until it was known that a successful orbit had been achieved, eliminating the hazard of abort radioactive contamination. If a 300-w output were achieved with a 5% overall efficiency, the 1-year burnup of fuel would be only 2 gm. Operation for many years would thus be possible. The reactor unit would have a longer life than comparable radioisotope units, and would offer clear advantages during ground operation.

The AEC representative requested that AI study this concept, and report their findings at the next meeting.⁽¹³⁾ At the sixth meeting of the SNAP Coordinating Committee,⁽¹⁴⁾ in January 1959, AI presented the results of a feasibility study for a reactor thermoelectric system. The study was based on a 200-w net electrical output from a completely static system, without the use of heat transfer fluids or pumps. It was indicated that active reactivity control would not be necessary, and the negative temperature coefficient of reactivity alone would permit operation at constant temperature within acceptable limits. The estimated system weight was 400 lb. Orbital startup was considered feasible, and

a lifetime of several years was considered likely. This concept was tentatively identified as SNAP - IIb, and the power requirement established as 250 w at 28 v dc.⁽¹⁵⁾

In March 1959, the AEC formally established the reactor thermoelectric program as SNAP 10, with a power requirements of 250 w and a weight objective of 1 w/lb, excluding the shield. The "hot strap" temperature of the thermoelectric element was tentatively established at 1200° F, with an overall efficiency of 12% Carnot. These values subsequently turned out to be quite optimistic. The early work on this concept is contained in References 15 through 24, and is summarized in Reference 25.

In September 1959, a specification for a bid for the thermoelectric converter was prepared,⁽²⁶⁾ and a contract was placed with Westinghouse Electric Corporation in March 1960. This contract was based on a power of 250 w, a hot strap temperature of 1170° F, and an overall efficiency of 3.66%.

The original SNAP 10 concept consisted of a conduction-cooled reactor with a thermoelectric converter mounted on its surface, as shown in Figure 1. The initial design calculations indicated that, with placement of thermoelectric converters around the radial surface of the reactor, the surface area requirements were greater than the surface available. Fins were therefore used to extend the surface available for radiation. To prevent scattering of the nuclear radiation from the reactor by the finned radiator, it was necessary to increase the size

Figure 1.
Original SNAP 10
Concept

of the radiation shield. This conduction-cooled concept met all the then-existing requirements for the SNAP 10 System.

After this pure conduction-cooled reactor-converter-radiator concept was selected, the SER, a NaK-cooled SNAP 2 concept, was operated for >125,000 kwh of thermal output with excellent results. This integrated output was in excess of the power required for the SNAP 10 System during the 1-year life. It thus appeared possible to reduce the complexity of the overall development effort for the SNAP 10 Program by utilizing a standardized SNAP 2 reactor in the SNAP 10 System, rather than developing a reactor specifically tailored to the low-power system.

In view of this demonstrated capability, a study was made of a system using forced-convection cooling to meet the SNAP 10 requirements. This system would use the SNAP 2 reactor in conjunction with a forced-convection liquid metal loop to transfer the heat to an integral thermoelectric converter-radiator. Reduction in heat transfer limitations imposed in the conduction system would permit system development objectives to be achieved with a minimum of advanced development. Also, the total system weight would remain the same. The forced-convection concept also had the advantage of growth up to several kilowatts; and the radiator could be located below the radiation shield, permitting a reduction in shield size and weight over that required for the pure conduction concept. In addition to reducing the reactor development problems, this concept would improve heat transfer efficiencies and radiator requirements, and thus permit lower thermoelectric element hot strap temperatures. It was estimated that these lower hot strap temperatures would extend converter life by a factor of 10, by reducing the converter degradation rate. Also, since the reactor temperatures were lower than those used on the SNAP 2 System, such reductions would enhance the reliability, hydrogen retention, and reactivity stability already demonstrated in the SER.

On June 15, 1960, AI requested permission from the AEC to proceed with the forced-convection-cooled SNAP 10 concept.⁽²⁷⁾ Authorization to proceed with the development of an electromagnetic pump for this concept was given by the AEC on November 30, 1960.⁽²⁸⁾ On December 27, 1960, the Department of Defense, in a letter from Herbert F. York, Director of Defense Research and Engineering, to John A. McCone, Chairman of the Atomic Energy Commission,

formally defined a reactor thermoelectric space power system at 300 w as a nuclear reactor program which would meet Air Force requirements.⁽²⁹⁾ In the meantime, as a result of discussions between the AEC Division of Reactor Development (DRD) and Atomics International with the Air Force Ballistic Missile Division, it was decided to push ahead with the forced-convection-cooled design; in recognition of the power growth capability of this concept, the power level was raised from 300 to 500 w. At the 500-w level, the SNAP 10 unit would meet the requirements for a ballistic missile defense satellite which was under review at that time by AFBMD.

The AEC, on December 30 1960,⁽²⁸⁾ formally directed AI to terminate the conduction-cooled SNAP 10 Program, with exception of the effort necessary to complete testing of the SNAP 10 critical assembly and the Westinghouse thermoelectric converter.

AI was authorized to redirect effort to the convection-cooled system, utilizing a SNAP 2 reactor with minimum modifications, liquid metal heat transfer, a dc electromagnetic pump, and thermoelectric conversion with the cold junction rejecting heat to space. This system was designated SNAP 10A, with a design objective of 500 w electrical at a nominal 28 v at the end of 1-year life, and with an unshielded system weight not greater than 500 lb. The shielded system weight was not to exceed 750 lb. The thermoelectric converter design was to be developed, such that the power output would be capable of extension to higher levels. The SNAP 10A System was to be compatible with the launch vehicle and its environment, capable of orbital startup, and designed to eliminate the need for active control during the required 1-year system life.

3. SNAPSHOT Flight Test History and Planning

By early 1960, the state of development of the SNAP reactor thermoelectric and reactor mercury-Rankine system had reached the point where the Department of Defense (DOD) and the AEC considered early flight tests of these concepts desirable. The SNAPSHOT Program, as a joint USAF and AEC program, was established in 1960, with the objectives of two flight tests of the SNAP 10A reactor thermoelectric system in September 1962 and February 1963, and two flight tests of the SNAP 2 reactor mercury-Rankine systems in September 1963 and January 1964.

In a letter dated March 25, 1960, from Headquarters USAF, the Air Research and Development Command was directed to prepare a development plan for flight testing of nuclear auxiliary power systems. This action resulted in the Space System Abbreviated Development Plan for Nuclear Auxiliary Power Orbital Test (SNAPSHOT) Program⁽³⁰⁾ dated May 27, 1960. This plan was approved at the levels of Headquarters USAF, Secretary of the Air Force for Research and Development (SAFRD), and Director of Defense Research and Engineering (DDR&E), and forwarded to the AEC through the Military Liaison Committee on December 27, 1960. AEC concurrence with the plan, citing subsequent schedule and specification changes on SNAP 10 at 300 w to SNAP 10A at 500 w, which had been worked out between the AEC and USAF project offices concerned, was transmitted by letter on March 7, 1961.⁽³¹⁾

On April 17, 1961, Headquarters Space Systems Division, Air Force Systems Command issued the "Proposed Advanced Technology Program, Nuclear Auxiliary Power Orbital Tests (SNAPSHOT)" plan.⁽³¹⁾

The flight dates were revised by the USAF, in June 1963, to July and November of 1964; and the delivery of the SNAP 10A units to VAFB was rescheduled to June 1, 1964 and October 1, 1964. In September 1963, the USAF formally slipped the schedule an additional 6 months, with flight dates of January 1965 and May 1965.

In December 1963, the USAF-funded development and manufacturing effort on the Agena vehicle for the SNAP 10A SNAPSHOT flight was terminated. Subsequently, all Air Force funding for the SNAP 10A flight test was withdrawn, for lack of a minimum requirement for space nuclear power systems.

As a result of the lack of firm USAF mission requirements, the Bureau of the Budget took the position that specific mission requirements should govern the pace of the AEC's space reactor programs. These Fiscal 1965 budget cut-backs resulted in a termination of SNAP 2 as a system, cancellation of the SNAP 10A flight testing, and shutdown of the SNAP 50 Program. SNAP 10A was reprogrammed for ground testing only.

In February 1964, the Joint Congressional Committee on Atomic Energy proposed that the AEC proceed with the flight testing of SNAP 10A, using AEC funds. As a result, the AEC Fiscal 1965 authorization bill was revised to include

a provision for SNAP 10A flight testing, including the procurement of the Atlas booster and Agena satellite vehicle.

The SNAP 10A flight test program was reinstated in March 1964, with a flight date of March 1965. Procurement of an Atlas booster and a second flight Agena was also programmed for a second flight ~6 months later, in the event of a failure of the first flight.

After this date, the program remained on schedule, and liftoff occurred at 13:24 PST on April 3, 1965. Full power was achieved ~12 hr after launch. On May 16, 1965, after 43 days of continuous operation, premature shutdown of the SNAP 10A System occurred, as the result of an electrical malfunction in the Agena.

The flight test program was considered a success, and a second flight was not scheduled.

The SNAP 10A Program was formally concluded on June 30, 1966.

4. High-Power SNAP Systems — SNAP 8 and SNAP 8 Derivative Systems

Mission power requirements, anticipated to require increased power requirements and increased booster payload capabilities, were projected, resulting in development of higher-powered SNAP systems. The SNAP 8 System, using a zirconium-uranium-hydride core and mercury-Rankine cycle power conversion system, was initiated in early 1960 to develop a system to supply the 30 to 60 kwe power range of anticipated requirements. The system selected required outlet temperatures nominally of 1300° F, the highest proposed for SNAP systems. The nuclear system was contracted to AI by the AEC, while the power conversion system effort and system integration was conducted by Aerojet General Corp., under a NASA contract. The details of the AGC effort are summarized in Reference 32.

The first test reactor, SNAP 8 Experimental Reactor (S8ER), was tested in Building 010 at the Santa Susana Field Laboratory without the power conversion system. Critical experiments were run in the fall of 1962, and initial significant power operation began on November 22, 1963. The reactor test was terminated on April 15, 1965, after 12,080 hr of operation, 5014 hr of which were at 450 kw or greater and 1200° F outlet or greater.

Post-test disassembly disclosed that $\sim 80\%$ of the 211 fuel elements had cladding cracks. A few failures had been anticipated from surge tank fission gas activity, but lack of accurate data on fission product release characteristics of the fuel had prevented anticipation of the degree of failure actually observed.

A special investigation was undertaken by the AEC with members of the national laboratories, to determine the cause of the cladding cracks. It was determined that fuel swelling induced strains in the cladding beyond the capability of Hastelloy N, which undergoes severe ductility decrease under irradiation. The fuel swelling was in excess of the limited design fuel-to-cladding gap which was selected prior to significant fuel swelling data development. The fuel swelling was attributed to excessive growth at the $>1400^{\circ}\text{F}$ fuel temperatures experienced in the core as a result of the hydraulics of the core design.

A developmental reactor, S8DR, was in the process of design. It was modified to increase the fuel-to-cladding gap somewhat, adopt a new ceramic hydrogen barrier, and hydraulic baffling was modified (in an attempt to moderate core temperatures).

A non-nuclear developmental mockup, S8DRM-1, was assembled for an environmental test of the reactor and flight-oriented hardware associated with the primary heat transfer system. Successful qualification of the non-nuclear hardware was achieved by April 1965.

The second test reactor, the SNAP 8 Developmental Reactor (S8DR), achieved dry criticality on June 22, 1968; and five automatic startups were performed in December 1968. Power operation was initiated in January 1969, initially at 600 kwt, 1300°F outlet. During the initial 1000-hr operation, power and temperature anomalous fluctuations were observed, but these ceased prior to the shift to a 500-hr, 1-Mwt, 1150°F outlet operation.

Increased fission product activity in the cover gas and coolant, and increased hydrogen loss, was noticed in the subsequent 600-kwt, 1300°F operation. The outlet temperature was dropped to 1200°F following a scram in July 1969, and the reactivity loss rate (corrected for temperature difference) was noted to increase following this and subsequent scrams. The fission product activity continued to rise. An administrative decision was made to shut down the S8DR on December 15, 1969, after 7000 hr of operation, to determine the cause of the apparent fuel cladding failures.

Post-test examination revealed all external components in excellent condition. A total of 72 fuel elements were found to have cracks in their cladding. Failure, as in S8ER, was attributed to unanticipated fuel growth in the hot, mid-radius zone of the core, where it was determined, in subsequent hydraulic tests, that vortexing of the coolant had occurred, and excessive gas had accumulated from that entrained in the flow-through surge tank of the system. The operational power and temperature anomalies were deduced to have occurred from a hydrogen redistribution feedback on cluster-decluster of fuel elements, which eventually settled into a tight-packed cluster in the high-temperature mid-radius rings. The fuel growth was again ascribed to core thermal-hydraulic design-related excessive fuel temperatures.

Three derivative reactor designs were taken through conceptual design, following the S8DR test, in the 1969-1971 time period. They were the Zirconium Hydride Reference Reactor, the Reactor Core Test, and the Space Power Facility Reactor.

The Reference Zirconium Hydride reactor was a 600-kwt, 1250° F outlet, 20,000-hr life design, to meet the requirements for a 25-kwe thermoelectric power conversion system. It was designed for 4 π shielding.

The Reactor Core Test, a 600-kwt, 1200° F, 20,000-hr concept, was designed to resolve the fuel element problems still outstanding after the S8DR and S8ER post-mortem findings. It used a finned element design to promote coolant cross flow.

The Space Power Facility Reactor, a 300-kwt, 1200° F outlet, 5-year life concept, was designed for operation at the NASA Plumbrook Space Power Facility. This concept was the last design effort in the SNAP 8 family of reactors.

B. SNAP FOR LAND AND SEA

Applications studies and surveys for DOD and potential commercial applications indicated that a power plant was needed that would serve the power range from 400 to 3000 kwe. These applications included:

- 1) Large remote radar bases
- 2) Truck- and rail-mounted power for relocatable base operations and energy depots

- 3) Hardened missile bases
- 4) Barge- or ship-mounted emergency power and service
- 5) Remote island bases for power and water (desalination)
- 6) Antarctic bases, and deep underground (underocean) bases
- 7) Deep ocean-bottom surveillance system power
- 8) Hardened strategic weapons silos
- 9) Hardened strategic command centers
- 10) Advanced military base power (portable)
- 11) Advanced remote radar bases (transportable)
- 12) Remote mining power (relocatable)
- 13) Deep underground mines
- 14) Small island industrialization and resort hotel power — water and refrigeration
- 15) Off-shore oil (deep bottom separation and processing)
- 16) Power for development of special remote resources in under-developed countries
- 17) Long-range fishing vessel propulsion and power.

Other small nuclear power applications which were examined, but found to be marginally feasible, in descending order, are:

- 1) Submarine tug — submerged barge sea transportation
- 2) Nuclear-powered locomotives
- 3) Large nuclear-powered ocean-going blimps, for remote airborne radar picket early warning system.

These systems were judged technically feasible, but met safety and economic obstacles that would require a great deal of experience from operations of plants listed among the primary applications list, before final judgment could be made. However, nuclear-powered land-mobile combat vehicles did not appear to be economically or operationally (safety) feasible.

SNAP 4 was a new reactor concept that was specifically derived to overcome the engineering problems the AEC-Army water reactor program encountered, and to provide the Navy with an alternative small power supply for deep-ocean and anti-submarine warfare systems. It was further recognized that, if these objectives could be achieved, a potential commercial market existed for remote mining operations and remote island industrialization. From the beginning, SNAP 4 was derived as a power "system." It was recognized that many of the problems the Army effort encountered in pursuing its objectives were outside of the reactor core, and even outside the reactor itself.

In order to meet the transportability and economic objectives, it was necessary to reduce maintenance and operating personnel requirements, and their attendant logistics support. It was further recognized that economics dictated a multipurpose "standardized" design, in order to write development and engineering off over many units, and to reduce production and erection costs per unit.

When put in production, the power system could be competitive in size and energy cost with large diesels in remote areas, underwater areas, or areas more than a few hundred miles from seaports or refineries.

II. SNAP 10A

The difficulty of supplying electric power for space application, and the unique energy-density advantages of nuclear fission, were recognized in the early 1950's. By 1955, studies had resulted in the identification of a reactor concept for the 0.5- to 10-kwe power range. A U^{235} -fueled and zirconium-hydride-moderated reactor with liquid metal coolant was chosen, on the basis of minimum reactor weight, applicable operating temperature, and a reasonable extrapolation of reactor and materials state of the art. These studies formed the basis for the establishment of the SNAP Program by the Atomic Energy Commission in May 1956. Reactor concept verification by means of a critical assembly was achieved in October 1957.

Interest in thermoelectric power conversion for space systems was initiated in 1958. Notable technical achievements in the SNAP Program brought the thermoelectric direct-conversion nuclear-power systems to an advanced development stage by 1960, requiring further definition of plans for their proof testing in an earth orbital flight. Consequently, a joint AEC - Air Force flight demonstration program, termed SNAPSHOT, was established.

A. PROGRAM OBJECTIVES

The objectives of the SNAP 10A-SNAPSHOT Program was to:

- 1) Demonstrate, proof test, and flight qualify SNAP 10A for subsequent operational use
- 2) Obtain public acceptance of nuclear reactors in space, by launching, under unquestionably nonhazardous conditions, simple forerunners to larger, more complex systems

B. PROGRAM HISTORY

As noted previously, the SNAP compact hydride U^{235} reactor concept was conceived at AI in 1951, as the result of conversations with RAND Corporation personnel, who indicated a requirement for a reactor heat source for earth satellite applications. From that time on, numerous studies of reactor thermoelectric system designs and spacecraft requirements were conducted. On December 30, 1960, the AEC formally directed AI to proceed with the design of a

convection-cooled system, utilizing a SNAP 2 reactor with minimum modifications, liquid-metal heat transfer, a dc electromagnetic pump, and thermoelectric conversion with the cold junction rejecting heat to space. This system was designated SNAP 10A, with a design objective of 500 w electrical at a nominal 28 v at the end of 1 year of life. The thermoelectric converter design was to be developed, such that the power output would be capable of extension to higher levels. The SNAP 10A System was to be compatible with the launch vehicle and its environment, capable of orbital startup, and designed to eliminate the need for active control during the required 1-year system life. Table 1 is a summary of the power, weight, and flight schedule changes for the SNAP 10A reactor thermoelectric concept from 1960 through the flight test in April 1965.

TABLE 1
SNAP 10-10A POWER, WEIGHT, AND FLIGHT DATE HISTORY

Date	Power (w)	Unshielded Weight (lb)	Shielded Weight (lb)	Flight Date	
				1st	2nd
December 1960	500	500	750	June 1963	-
May 1961	-	505	825	January 1963	April 1963
December 1961	-	-	-	March 1963	-
February 1962	-	-	-	April 1963	-
May 1962	-	-	-	September 1963	-
July 1962	-	650	950	-	-
April 1963	-	-	-	June 1964	September 1964
June 1963	-	-	-	July 1964	November 1964
September 1963	-	-	-	January 1965	May 1965
March 1964	500	650	950	March 1965	-

In May 1961, following the issuance of the USAF "Proposed Advanced Technology Program for Nuclear Auxiliary Power Orbital Tests (SNAPSHOT)," the SNAP 10A Program had been formalized to include the following test and flight systems:

- 1) PSM-1, a prototype structural test system
- 2) PSM-2, an electrical analogue simulator of the thermoelectric converter, to be delivered to the Agena contractor in February 1962 for combined SNAP 10A - Agena testing.

- 3) PSM-3, a prototype thermal and hydraulic test system
- 4) FSM-1, an electrically heated, non-nuclear, qualification test system
- 5) FSM-2, an electrically heated qualification test system, to be delivered to the Agena contractor in September 1972
- 6) FSM-3, a second electrically heated qualification test system, to be delivered to the Agena contractor in January 1963
- 7) FS-1, a nuclear qualification test system, scheduled for criticality tests in January 1962
- 8) FS-2, the first flight system, scheduled for delivery in December 1962
- 9) FS-3, the second flight system, scheduled for delivery in June 1963
- 10) FS-4, a spare flight system, scheduled for delivery in January 1964.

The weight limit was re-established at 875 lb, which included a 50-lb contingency. The power objective remained at 500 w at the end of 1 year. However, it was recognized that a 90-day orbital test was a more reasonable objective, corresponding with the expected status of component qualification by the planned launch date of March 13, 1963.

The FSM-2 test system was originally intended to be mated with the Agena in a static firing test by the vehicle contractor, and the FSM-3 test system was to be operated with the Agena in a combined thermal vacuum test. These testing requirements were deleted from the program in December 1961, and FSM-2 and FSM-3 were designated non-power-producing electric mockups, to be used in combined Agena - SNAP 10A electrical and mechanical compatibility testing. FSM-2 was a prototype model, while FSM-3 was to include the flight hardware. These systems were later designated FSEM-2 and FSEM-3, to indicate their primary function as electrical mockups.

In December 1961, the silicon-germanium (SiGe) direct radiating thermoelectric concept was selected for the SNAP 10A System, and a production contract placed with Radio Corporation of America (RCA).

In April 1962, the USAF revised the Agena development and fabrication schedule to a September 1963 launch, and delivery of FSM-2 was delayed 6 months. This schedule delay in the Agena Program was primarily the result of the need to radiation harden and requalify certain critical Agena components.

In June 1962, the revised schedule for the delivery of the flight systems to the Air Force at VAFB was July 1, 1963 for FS-2, October 1, 1963 for FS-3, and August 15, 1963 for the space flight system, FS-4. FSEM-2 was to be delivered to the Agena contractor on October 15, 1962, and FSEM-3 on April 1, 1963.

In June 1962, AI recommended to the AEC that a second non-nuclear qualification test system, FSM-4, be added to the SNAP 10A Program. Scheduling of an additional non-nuclear qualification test at this time permitted the use of fully qualified components of the flight design, and the use of SNAP 10A - Agena compatibility test data, in updating the system design and the testing program.

The FSM-4 test system was approved in July 1962, and the FS-2 system, then starting assembly, was redesigned as FSM-4. A new flight system, FS-5, was scheduled as a replacement for FS-2.

The flight dates were revised by the USAF in June 1963, to July and November of 1964, and the delivery of the SNAP 10A units to VAFB was rescheduled to June 1, 1964 and October 1, 1964. In September 1963, the USAF formally slipped the schedule an additional 6 months, with flight dates of January 1965 and May 1965.

In December 1963, the USAF-funded development and manufacturing effort on the Agena vehicle for the SNAP 10A SNAPSHOT flight was terminated. Subsequently, all Air Force funding for the SNAP 10A flight test was withdrawn, for lack of a minimum requirement for space nuclear power systems.

As a result of the lack of firm USAF mission requirements, the Bureau of the Budget took the position that specific mission requirements should govern the pace of the AEC's space reactor programs. These Fiscal 1965 budget cut-backs resulted in a termination of SNAP 2 as a system, cancellation of the SNAP 10A flight testing, and shutdown of the SNAP 50 program. SNAP 10A was reprogrammed for ground testing only.

In February 1964, the Joint Congressional Committee on Atomic Energy proposed that the AEC proceed with the flight testing of SNAP 10A, using AEC funds. As a result, the AEC Fiscal 1965 authorization bill was revised to include a provision for SNAP 10A flight testing, including the procurement of the Atlas booster and Agena satellite vehicle.

The SNAP 10A flight test program was reinstated in March 1964, with a flight date of March 1965. Procurement of an Atlas booster and a second-flight Agena was also programmed for a second flight, ~6 months later, in the event of a failure of the first flight.

After this date, the program remained on schedule, and liftoff occurred at 13:24 PST on April 3, 1965. Full power was achieved ~12 hr after launch. On May 16, 1965, after 43 days of continuous operation, premature shutdown of the SNAP 10A System occurred, as the result of an electrical malfunction in the Agena.

The flight test program was considered a success, and a second flight was not scheduled.

The SNAP 10A Program was formally concluded on June 30, 1966.

C. SYSTEM DESCRIPTION

This section defines the system requirements, and describes component and system characteristics.

1. Design Requirements

The SNAPSHOT Program established the following general system requirements:

a. Power Output

The electrical power output of not less than 500 w, associated with a 1-year operating lifetime, with excess capability or other features as required to compensate for performance degradation over that period of time.

b. Lifetime

A 1-year mean-time-to-failure as a design goal for the SNAP 10A power unit. This 1-year lifetime objective was utilized in selecting materials, shielding, operating temperatures, and other design parameters.

Determination of the actual lifetime achieved in orbit, and observation of performance degradation with time, were SNAPSHOT objectives. Lifetimes of 1 year were not considered likely on these experimental flights; nevertheless, the tests were designed and planned with a 1-year life objective.

A nuclear ground-test demonstration of 90 days integrated operation at full power by one flight prototype unit was the acceptance criterion for flight.

c. Weight and Length

The SNAP 10A unit could not exceed 124 in. in length, nor a total weight of 825 lb, plus up to 50 lb for diagnostic and test instrumentation. This instrumentation would not be required in subsequent operational systems. This weight did not include the nose cone or shroud used to protect the SNAP 10A System during ascent.

d. Shielding

Shielding within the SNAP 10A structural package limited the integrated 1-year nuclear radiation flux across a 60-in. diameter dose plane, 17.5 ft from the bottom of the reactor, to 10^7 r and 10^{12} n/cm². The location and configuration of additional shielding within the Agena vehicle, if required, was to be an Air Force responsibility. This did not preclude additional shielding within the SNAP 10A structural unit when its placement there was jointly determined to be advantageous.

e. Integration Test Hardware

The AEC was to furnish the following test items to the Air Force for the SNAP 10A flights:

- 1) Structural Mockup — The structural mockup was to reproduce the configuration and mass distribution of the flight package, including fluids. It was to be used to check physical mating and mechanical interactions during static test firings.
- 2) Electrical Prototype — Suitably placed electrical heaters were to be substituted for the nuclear heat source. Otherwise, this prototype was to duplicate the flight package, and generate full thermal and electrical power. It was to be used in environmental tests affecting final Agena vehicle design. In subsequent integration checks to demonstrate thermal and electrical compatibility, it was to furnish power to the Agena vehicle.
- 3) GSE Simulators — This unit was to duplicate those features of the payload package necessary to rehearse launch pad procedures. It was to be used to establish compatibility of special handling.

f. Ground Support Equipment

- 1) Prelaunch — The AEC was to provide ground support equipment (GSE) and services for checkout of the SNAP payload package in prelaunch operations, before and after mating to the Agena vehicle. Equipment and services incident to the launching vehicle was to be furnished by the Air Force.
- 2) Post-launch — Tracking, command, and data acquisition were to be furnished by the Air Force, using existing facilities. Data reduction was also to be furnished by the Air Force.

The specific design objectives for SNAP 10A at the time of the flight test were:

- 1) Produce a minimum of 500 w at 28.5 v dc for a minimum of 1 year in a space environment
- 2) The system configuration to be compatible with, and capable of mounting to, the forward end of the Agena satellite vehicle
- 3) Total system weight not to exceed 950 lb, including the nuclear radiation shield and the diagnostic instrumentation used to obtain flight test performance data
- 4) The system to be capable of remote automatic startup in orbit, via radio command
- 5) Following the initial startup period, the system to be self-regulating, without the need for an automatic control system
- 6) Following the initial startup period, the system to be static, involving no moving mechanical parts, with the single exception of the NaK coolant flow
- 7) The system to include a radiation shield capable of attenuating the reactor neutron and gamma flux to 5×10^{12} nvt/year and 4×10^7 r/year at the Agena - SNAP 10A mating plane
- 8) The system to be designed to survive the launch environment, using the Atlas-Agena booster combination.

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Figure 2. SNAP 10A System

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Figure 3. SNAP 10A Thermodynamic Cycle

2. System Description

The SNAP 10A System utilized a compact thermal reactor employing zirconium hydride fully enriched uranium fuel, and a direct radiating SiGe thermoelectric converter. The heat was transported from the reactor to a thermoelectric power conversion unit by a closed NaK heat transfer system, utilizing a thermoelectric-powered dc conduction pump. Heat was rejected to space by means of a 62.5 ft^2 radiating surface, made up of individual radiators on each thermoelectric couple. Lithium hydride was used for the nuclear radiation shield. An illustration of the SNAP 10A system, indicating the major system components, is shown in Figure 2. Figure 3 is a schematic of the thermodynamic cycle.

To integrate the reactor heat source and the thermoelectric power conversion into a complete useful space power system, it was necessary to consider spacecraft integration and operational criteria. The most important of these were: (1) thermal isolation from the spacecraft payload, (2) operation in the space environment of vacuum and micrometeorites, (3) remote startup in orbit, via radio command, (4) capability to operate without subjecting the spacecraft to excessive disturbing torques, (5) design and installation to permit efficient, low-weight, shadow-shielding of payloads, (6) minimum interference and interaction with basic booster and payload subsystems, (7) configuration within vehicle-structural and flight-stability criteria, (8) capability to withstand the severe shocks, vibrations, gravity, pressure, and temperature transients during vehicle launch, (9) prelaunch installation and checkout with maximum safety, and minimum vehicle and facility interference, and (10) capability of low-cost quantity production with maximum reliability.

The basic thermodynamic coupling of the reactor heat source and the thermoelectric converter was accomplished by a pumped liquid-metal loop of NaK-78. The objective of no moving parts was maintained by the application of a dc conduction pump.

The system configuration was virtually dictated by minimum-weight shielding requirements. The shielding specification for SNAP 10A is to limit the radiation level at the power unit - spacecraft interface to $\sim 5 \times 10^{12}$ nvt fast neutrons, and 4×10^7 r gammas in 1 year. Beyond this level, semiconductor payload

TABLE 2
SNAP 10A SYSTEM DESIGN PARAMETERS

System Characteristics	Original Concept (1961)	Flight Design (1963)	Flight System (1965)
Electrical power, minimum (w)	500	500	533
Reactor thermal power, nominal (kw)	30	33.5	43.8
Reactor type			
Fuel	ZrH-U ²³⁵	ZrH-U ²³⁵	ZrH-U ²³⁵
Reflector	Be	Be	Be
System weight, shielded (lb)	875	935	969
Design life (years)	1	1	1
Radiator area (ft ²)	62	62.5	62.5
Overall package length (in.)	-	137	137
Mounting base diameter (in.)	50	50	50
Coolant	NaK-78	NaK-78	NaK-78
Coolant flow rate (gpm)	12	12.5	13.3
Reactor outlet temperature (°F)	936	990	1030
Reactor inlet temperature (°F)	828	882	962
Power conversion system			
Material	PbTe	SiGe	SiGe
Voltage	28.5	28.5	30.3

PHYSICAL PROPERTIES

BASE ALLOY	Zr - 10 wt% U
N _H	6 - 6.5 atoms H/cc fuel (x 10 ²²)
DENSITY	5.6 gm/cc
THERMAL COND.	12 Btu/sec-in.-°F
HEAT CAPACITY	0.18 Btu/lb.-°F
STRENGTH	10,000 - 20,000 psi

Figure 4. SNAP 10A Fuel Element

hardening becomes difficult. The critical shielding problem was the elimination of neutron scattering from miscellaneous components and structural members. The most efficient shield resulted when the required payload shadow shield served the additional function of eliminating unwanted scattering. This was accomplished by locating all components and structural members within the conical shadow of the basic payload shield. The result was a conical configuration, with a base diameter determined by the payload. This configuration was compatible with the large radiator surface area required for heat rejection to space. The complete SNAP 10A System had an overall height of 137 in., and a mounting base diameter of 50 in. This configuration was dictated by minimum-weight shield requirements.

The more important parameters of the SNAP 10A system are summarized in Table 2. These parameters are tabulated for the original design concept, as developed in 1961, the flight design of 1963, and for the SNAPSHOT FS-4 flight test in 1965.

3. Reactor Description

The concept developed for the SNAP 10A reactor consisted of a core containing 37 uranium-zirconium fuel elements enriched with U^{235} (Figure 4). The fuel (1.25 in. diameter) elements were hydrided to an N_H of 6.5 (6.5×10^{22} atoms H/cc-fuel), which is approximately the same hydrogen concentration as that of cold water. Each element was clad in Hastelloy N tubing, with a ceramic hydrogen barrier applied to its inner surface. The fuel elements were positioned in the 8.9-in. diameter by 16.125-in. long core by upper and lower gridplates. The gridplates were held by a fixed ring at the bottom and holddown springs at the top. Beryllium reflectors were used in the void space between the close-packed hexagonal fuel-element array and the circular reactor vessel. The top reactor-vessel head acted both as a vessel closure and restraining structure for the core holddown springs.

The reactor vessel was surrounded by a beryllium structure which functions as a reflector, as shown in Figures 5 and 6. Four control drums, which formed a part of the reflector assembly, provide both the fine and coarse control required for SNAP 10A reactor control. As shown in Figure 7, a lithium hydride (LiH) shield was positioned directly below the reactor vessel, and was contained

Figure 5. SNAP 10A Reactor

Figure 6. SNAP 10A Control and Reflector Assembly

Figure 7. Exploded View of SNAP 10A Reactor and Shield

AI-AEC-13067

in a stainless-steel casing. In the SNAP 10A System, a flexible mount was provided to support the LiH within the casing, allowing thermal cycling of the system. Actuating devices and instrumentation were attached directly to the reflector assembly. Control elements, such as the reactor controller, were located in the instrumentation compartment.

The SNAP 10A reactor subsystem consisted of the following five subassemblies and their components:

- 1) Reactor structure
 - a) Reactor vessel
 - b) Reactor vessel top head
 - c) Gridplate support ring
 - d) Reactor vessel support structure
 - e) Attachment brackets for pumps, reflectors, etc.
- 2) Reflector assembly (Figure 6)
 - a) Main reflector block
 - b) Control drums
 - c) Drum drives (gears, bearings, etc.)
 - d) Actuator motors
 - e) Position indicators
 - f) Reflector ejection system (retaining band, squib actuators, springs, actuating devices, etc.)
 - g) Wiring harness
- 3) Reactor Core
 - a) Gridplates
 - b) Fuel elements
 - c) Core positioning mechanism
- 4) Radiation Shield (Figure 7)
 - a) Shield casing
 - b) Shielding material
 - c) Shield support assembly
- 5) Control Equipment and Instrumentation
 - a) Controller
 - b) Temperature sensors
 - c) Reflector ejection timer
 - d) Position sensor demodulator

4. Heat Transfer System and Pump

Heat was transported from the reactor to the power conversion system by NaK-78 coolant, which was chosen for its low melting point of $\sim 12^{\circ}\text{F}$ and low vapor pressure at reactor operating temperatures. Heated NaK, exiting from the core at 1010°F , was circulated at a rate of 13.1 gpm by means of a liquid-metal dc conduction-type pump composed of a permanent magnet and an integral thermoelectric power supply. This thermoelectric power supply provided high current and low voltage, and consisted of a thermocouple which utilized the 300°F ΔT established by the hot NaK temperature and a cold-junction integral heat-rejection radiator. The hot NaK was then pumped to the thermoelectric power conversion system which converted a portion of the thermal energy directly into electrical energy through the Seebeck effect. The pump is shown in Figure 8.

From the outlet of the power conversion system, the NaK coolant was collected and returned to the reactor core. Attached to the return leg were

Figure 8. SNAP 10A Thermoelectric Pump 7561-5469

bellows-type expansion compensators which pressurized the system in orbit to 5 psia, and could accommodate the full thermal expansion of the NaK during heatup. The entire NaK piping circuitry was arranged to take maximum advantage of the existing structure and components for meteoroid protection.

5. Thermoelectrics

The principle of direct conversion of heat to electricity was discovered by Thomas Seebeck in 1822. For over 100 years, the approach was considered to be of no practical significance, because of the low efficiency of known materials. The efficiency of a thermoelectric power conversion device is a function of the material's figure of merit, Z , which is defined as $s^2/\rho k$ where s is the Seebeck coefficient ($v/^\circ C$), ρ is the electrical resistivity ($\Omega\text{-cm}$), and k is the thermal conductivity ($w/cm\text{-}^\circ C$). The development of semiconductor materials in the late 1950's resulted in the identification of a number of intermetallic compounds and alloys of low thermal conductivity that could be doped to provide a carrier concentration leading to a maximum Z . For a semiconductor material with a Z of $2 \times 10^{-3} \text{ } ^\circ C^{-1}$, the ideal efficiency is $\sim 10\%$ at a temperature difference of $200^\circ C$. This is still a low efficiency; however, the large energy content of a reactor minimizes the significance of this inefficiency, and the disadvantage was further offset by the inherent reliability of direct conversion with no moving parts. In practical large space power units, the high heat-rejection temperature required for minimum radiator area and weight results in converter net efficiencies of 2 to 3%.

Of practical importance were the identification of a converter design capable of capturing a maximum fraction of the available theoretical materials efficiency, and the degree to which the physical and metallurgical properties of a given thermoelectric material control the design. In the design of a practical thermoelectric converter: (1) the individual thermocouples must be electrically isolated to allow a series connection to provide a useful output voltage, (2) the transfer of heat from source to radiator must have minimum thermal impedance in series with the thermoelectric material, because loss of available temperature difference directly diminishes the converter efficiency, (3) the electrical impedance of interconnections and contact junctions in series with the thermoelectric material must be minimal, because extraneous electrical losses directly diminish the converter efficiency, and (4) shunt heat transfer from source to radiator must be minimal.

The physical and metallurgical properties of the thermoelectric material can greatly influence the converter design and efficiency. For example: (1) the extraneous converter resistance is strongly influenced by the ability to make metallurgically stable, minimum-resistance electrical contacts to the thermoelectric material, (2) any requirement for a coating or barrier on the thermoelectric material to suppress sublimation provides a shunt heat path, and (3) the physical strength of the thermoelectric material can determine structural or thermal stress limitations.

The SNAP 10A converter had to operate for 1 year in space, at $\sim 1000^{\circ}\text{F}$ (558°C), with an efficiency degradation of $<10\%$, and survive the missile launch into space which imposed 10-g acceleration, 50-g shock, and 5-g vibration in the range of a few to several thousand cycles per second.

In the case of the SNAP 10A converter the contending thermoelectric materials were PbTe and SiGe alloy. In the final selection, SiGe alloy was chosen, despite its lower figure of merit, because:

- 1) SiGe is stable to above 1800°F (980°C); PbTe sublimates at temperatures above 800°F (426°C), and would require an encapsulant. The potential of high-temperature operation of SiGe provided future performance growth.
- 2) Stable low-resistance electrical contacts to SiGe can be made by metallurgical bonding. Experience with PbTe indicates unstable higher resistance mechanical contacting only.
- 3) The mechanical properties of SiGe alloy were more uniform, and less restrictive.

The power conversion system consisted of n- and p-doped SiGe thermoelectric material, thermally coupled with, but electrically isolated from, the NaK heat transfer medium. These SiGe pellets were bonded to 40 parallel stainless steel tubes, arranged in a conical configuration about the structure. The NaK flow from the reactor was equally divided among the 40 tubes. Along any one tube, there were 72 SiGe thermocouple elements, or a total of 2880 individual power-producing elements in the system. These pellets were electrically isolated from the hot NaK tube by thin alumina discs. The heat transport path was

A) ORIGINAL PRODUCTION MODULE CONSTRUCTION AND PROCESSING

B) FINAL PRODUCTION MODULE CONSTRUCTION AND PRODUCTION

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Figure 9. Construction of SNAP 10A Thermoelectric Converter

7580 1032C

Figure 10. SNAP 10A Thermoelectric Converter Module

from the NaK tube through the insulator, through the SiGe pellet, to an individual aluminum radiator. This radiator, which used the space vacuum to electrically insulate the cold junction, was sized to dissipate sufficient heat to maintain a ΔT of $\sim 310^\circ\text{F}$ between the hot and cold SiGe junctions. The total effective radiator area was 62.5 ft^2 , and had an average power dissipation capability of $\sim 0.5\text{ kw/ft}^2$ at a mean temperature of 604°F . All materials which comprised the thermoelectric modules were brazed or otherwise metallurgically bonded to each other, to ensure a sound structure and a thermally conductive stack as shown in Figure 9.

The electrical current flowed through n- and p-doped SiGe pellets, connected alternately at their hot junctions by a copper hot strap and at the cold junction through the aluminum radiators. All 72 pellets on a tube were connected electrically in series. Alternate converter legs were connected in parallel by jumpers between each of the 36 pairs of radiators. A drawing of the thermoelectric couples is shown in Figure 10.

6. Lithium Hydride Shield

The SNAP 10A shadow shield was located directly below the reactor, and utilized a cold-pressed lithium hydride shielding material, reinforced with stainless steel honeycomb and contained in a Type 316 stainless steel casing. It was designed to limit the total reactor integrated nuclear radiation to $<4 \times 10^7$ r of gamma radiation and $5 \times 10^{12}\text{ n/cm}^2$ for neutron energies of 1.0 Mev at any point on the dose plane for normal reactor operation. An average thermal power of 34 kw for 1 year was considered normal reactor operation.

The 225-lb shield was designed to be able to withstand 10 full thermal transients simulating reactor startup conditions. The shield also had to withstand reactor operating conditions, with a maximum temperature of 823°F at the attachment to the radiator converter and a minimum temperature of 583°F near the surface exposed to space.

One of the significant advances in the SNAP 10A Program was the selection of LiH for the radiation shield. The basic work on LiH was completed under the Aircraft Nuclear Power (ANP) Project. Additional work on LiH fabrication, handling, and processing was done at Oak Ridge. Lithium hydride has a very

Figure 11. SNAP 10A Conical Structure

high and stable hydrogen density, which makes it attractive as a fast neutron shield; it produces a minimum amount of secondary radiation, is almost an optimum-weight shield, is stable, and does not melt until temperatures of over 1200° F are reached. These characteristics give LiH a significant advantage over other shielding materials, such as water and hydrocarbon plastics.

7. Support Structure

The conical configuration of the power conversion system, support for the reactor and radiation shield, and mating to the Agena boost vehicle were achieved with a corrugated, thin-walled, sheet-metal structure. This structure consisted of 18-mil thick titanium, fabricated into square corrugations and formed into the conical shape. Internal stiffening was provided by conventional rings and gussets. A picture of the conical structure is shown in Figure 11. The top and bottom of the cone were further stiffened by a heavy box-like structure, designed to distribute the load evenly and ensure local torsional stability.

The entire conical structure was mated to the Agena vehicle through 4-1/2 in. long, Y-shaped titanium support legs which bolted to an Agena mating ring at eight discrete points. The legs provided load-carrying capability, freedom for differential thermal expansion, and thermal isolation from the Agena.

An instrument compartment was mounted immediately below the conical power conversion support structure. All control, instrumentation, and electrical signal conditioning components were located in the relatively low-temperature and low-nuclear-radiation environment of the instrument compartment.

An ejectable heat shield surrounded the thermoelectric converter, to prevent the liquid metal coolant from reaching the plugging temperature prior to startup. The heat shield was automatically ejected when the coolant temperature reached 275° F.

8. Control and Instrumentation

The function of the control and instrumentation system was to provide for automatic startup in orbit via ground command, and to monitor the performance of the system before and during startup and during full-power operation. These systems also provided for safety monitoring of the systems on the launch pad,

and for manual or automatic shutdown of the reactor by reflector ejection, both on the launch pad and in orbit. A system for recording critical system parameter data, in the event of a system malfunction or failure, was also provided.

Once the startup sequence was initiated by ground command, the control system provided all the logic, timing, and power switching required to bring the system to its design initial temperature level, and to maintain the electrical power capability above the design point for a fixed period of time. The control system was de-energized by ground command after the reactor system had attained a steady-state condition. Long-term reactivity changes, due to fuel burnup, fission product accumulation, and hydrogen diffusion, were compensated for by a burnable poison in the fuel.

III. SNAP 2

A. SYSTEM CONCEPT

1. Origins and History

The SNAP 2 Program started as an outgrowth of Rand Corp. studies in 1951 concerning the feasibility of nuclear reactor power supplies for satellite vehicle auxiliary power requirements. Interest in nuclear auxiliary power system concepts had increased steadily; and, in 1955, a formal request for proposal studies was issued jointly by the Department of Reactor Development (DRD) of the Atomic Energy Commission (AEC) and the Air Force (AF) Wright Air Development Center (WADC). Based on an earlier 9-month feasibility study, Atomics International (AI) and AiResearch, a Division of the Garrett Corp., proposed⁽⁸⁾ a homogeneous zirconium-uranium hydride (Zr-H-U) fueled reactor concept, utilizing sodium as a coolant and coupled to a mercury Rankine (MR) cycle power conversion subsystem (PCS). The early experimental work was then separately performed by two teams, Lockheed Missiles and Space Division (LMSD) in conjunction with Thompson Ramo Wooldridge (TRW), and AI in conjunction with AiResearch. Funding for early developmental work was jointly sponsored by the AEC (reactor subsystem) and the AF (power conversion subsystem).

In January of 1957, AI and AiResearch submitted the winning proposal for a 3-kwe powerplant,⁽³³⁾ utilizing a 1200°F zirconium hydride moderated, enriched uranium fueled reactor, coupled to a hermetically sealed MR PCS. In April of 1957, SNAP 2 became an authorized program, with the AEC assuming funding control over all developmental activities, and AI designated as the prime contractor. At AEC direction, a technical competition was held between AiResearch and TRW, relative to development of the PCS.^(34, 35) Both companies were evaluated as technically equal;⁽⁶⁾ however, a potential cost savings could be realized with the TRW plan, due to the fact that TRW was concurrently developing a similar 500-w turbomachinery PCS for the AEC SNAP 1 Program. Accordingly, the initial SNAP 2 PCS subcontract was awarded to TRW in January 1958. The AF-funded portion of the program subcontracted to AiResearch was phased out in March of 1958.⁽³⁶⁾

Figure 12. Basic SNAP 2 – Mercury Rankine System Schematic

By May of 1960, development had proceeded favorably, and the flight test plans had been specified with the establishment of the AEC-AF SNAPSHOT Program.⁽³⁷⁾ Four flights were scheduled: two SNAP 10 launches (a 300-w reactor thermoelectric power plant), and two SNAP 2 launches. The first flights of SNAP 10 and SNAP 2 were scheduled for September 1962, and September 1963, respectively. Launchings were to be from the Atlantic Missile Range (Cape Kennedy), utilizing Atlas-Agena boosters. In April of 1961, the SNAP 10A (now an uprated 500-w reactor thermoelectric powerplant) and SNAP 2 flights were rescheduled for Pacific Missile Range launches in the CY 1963-1964 period.⁽³⁸⁾ During the next 3 years, several SNAP 2 flight test re-schedules were incurred, due to delays in the SNAP 10A Program, launch vehicle availability, and turbo-machinery development problems.

In mid-FY 1964, the SNAP 2 flight test requirement was cancelled, and the program was reoriented as the Mercury Rankine Technology Program (MRP). The prime objective of the MRP was to develop the component and system technology (exclusive of the heat source, whether reactor or isotope) for a nuclear turboelectric space powerplant capability. The initial objective was to develop and prequalify, for 10,000 hr, the prototype modular components of a 3- to 5-kwe PCS module. These modular components would be utilized in multiple PCS installations to satisfy specified power and/or reliability requirements. During the period of FY 1965-1966, the program objectives were steadily modified as a result of funding reductions. On October 1, 1966 phase out of the MRP was initiated, except for the continuation at TRW of two 10,000-hr CRU-V endurance runs.

2. Description of Concept

The MR cycle had originally been selected because its temperature-pressure characteristics closely matched the hydride reactor capabilities and temperature limits. The cycle provided a reasonably high system efficiency, coupled with a small radiator area requirement.

A system schematic is shown in Figure 12. In the power cycle of the basic MR system, the compact SNAP reactor was used to heat NaK-78 fluid recirculating in a sealed loop. The hot NaK was circulated through a boiler to evaporate and superheat mercury. The mercury vapor was then expanded within a turbine

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Figure 13. Multiple Combined Rotating Unit Mercury Rankine Power Conversion System Installation

which drove a permanent magnet alternator and a mercury boiler feed pump; these three components were all mounted on a single, 36,000-rpm rotating shaft, and were supported by mercury-lubricated bearings. This assembly was called the Combined Rotating Unit (CRU). The alternator generated ac electrical power at 1800 Hz. The exhaust mercury vapor from the turbine was condensed and subcooled within the radiator-condenser (RC), and was pumped back to the boiler by means of a high-pressure boiler feed pump. The NaK primary loop working fluid was pumped by a thermoelectric (TE) pump (a dc conduction pump, in which the electrical current is supplied by thermocouple elements). The hot NaK provided heat to the thermocouple hot junction, and the relatively cold mercury returned from the RC and cooled the thermocouple cold junction. Figure 13 depicts a typical reactor system installation utilizing multiple CRU machinery. Many variations of the installation are possible, depending on the defined performance and reliability objectives.

An isotope heat source MR system would operate similarly. The heat source would consist of isotopic fuel capsules radiating to a surrounding forced-convection mercury boiler-superheater assembly. The boiling temperature would likely be selected somewhat higher than in the reactor system, in order to achieve a greater overall cycle efficiency. Most probably, a supplementary NaK coolant loop would be utilized to reject the (initially excessive) isotope heat source decay thermal energy. Figure 14 shows a typical isotope system installation.

The pertinent features of the MR concept include:

- 1) A high heat-rejection temperature, favoring small radiator area and low weight
- 2) A hermetic system, with no operating seals between working fluids, or between the system and the space environment
- 3) Reasonable simplicity, with a minimum of operating auxiliaries
- 4) A reasonably high turbine inlet temperature, yielding good cycle efficiencies, but not so high as to incur serious reliability penalties
- 5) A simple, single-shaft, rotating-machine concept, yielding high reliability.

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Figure 14. Multiple Combined Rotating Unit Mercury Rankine Power Conversion System With Isotope Heat Source

Advantages of the nuclear MR space power system include:

- 1) Long life (10,000 hr, and potentially greater), with no maintenance in the space environment
- 2) Minimum orientation requirements
- 3) Insensitivity to space radiation levels
- 4) No deployment requirements
- 5) Extreme flexibility in power output and configuration
- 6) Low orbital drag
- 7) Insensitivity to sun-shade variations; hence, no energy-storage requirement
- 8) Growth potential.

The major disadvantage of a MR system is associated with the corrosiveness of the mercury working fluid. Special materials must be utilized throughout; the corrosion products generated (crud) must be either filtered or separated out, to prevent possible deposition within critical flow areas. A unique difficulty is associated with protecting the alternator stator windings from liquid mercury. Also, the lubricating properties of mercury are rather poor; this deficiency has to be compensated for, so that the bearings will have adequate load-carrying capability. Cavitation-erosion damage within the pump and bearings is of major concern, if long-life requirements are imposed. Additionally, at the power levels considered, the low volumetric flow rates and high shaft speed requirements reduce machine efficiencies, and introduce critical clearance design criteria.

Problems associated with the reactor heat source would generally be found to be fewer and less difficult. They included, principally, fuel fabrication, hydrogen retention, and coolant pumping. Other major system components, such as the RC, were to present only straightforward materials and engineering development problems.

Altogether, the conceptual system presented a very great potential relative to the anticipated engineering difficulty, and truly merited an intensive development effort.

TABLE 3
SNAP 2 MERCURY RANKINE PROGRAM MILESTONE SUMMARY

AEC-AF Study Contract	May 1956
Feasibility Report	January 1957
AEC Development Contract	April 1957
PCS Competitive Proposals	August 1957
TRW PCS Subcontract	January 1958
First Experimental Attainment of Criticality with Reactor Mockup	October 1957
SER Critical	June 1959
Power Operation of the SER	October 1959
Initial Operation of CRU Proving Feasibility	September 1959
484-hr CRU-I Test	January 1960
Official Authorization of SNAPSHOT Flight Test Program	May 1960
SNAP 2 Flight Design Defined	June 1960
Redirection of CRU Development Program	December 1960
S2DR Reactor Test	April 1961
Flight Design Modified to Incorporate TE NaK Pump	January 1963
Phase I RC Test of PSM-3	February 1963
PSM-2 Structural Tests	August 1963
Performance Test of Prototype CRU-V Unit	September 1963
SNAP 2 Flight Test Cancelled, MRP Initiated	November 1963
Reoriented MRP Initiated	January 1964
Integrated System Operation of PSM-1	January 1964
Integrated System (0-g Simulation) Operation of PSM-3	September 1964
4868-hr CRU-V Endurance Run	August 1965
MRP Phaseout at AI	October 1966

B. TECHNICAL DEVELOPMENT

1. Scope and Major Accomplishments

The SNAP 2 - Mercury Rankine (MR) System development program extended over a period of ~ 10 years. The major accomplishments of the overall SNAP 2 - MRP included:

- 1) Development of the zirconium hydride reactor concept to the status of an operational nuclear heat source
- 2) Demonstration of the performance characteristics of a MR PCS
- 3) Verification of the feasibility of a high-speed, single-shaft, working-fluid-lubricated and hermetically sealed turboalternator assembly
- 4) Definition, by ground prototype and simulated zero-gravity tests, of the system startup, steady-state, and shutdown characteristics of a Rankine cycle powerplant for potential use in space.

Direct outgrowths of the SNAP 2 Program are the SNAP 10-10A and the SNAP 8 system concepts. Table 3 summarizes the program milestones.

At the time of program initiation, there were several major technological questions concerning the feasibility of space operation of a nuclear MR system. Table 4 summarizes these specific feasibility questions, with their subsequent initial demonstrations or verification accomplishments. As can be noted, there was a rational analytical, experimental and/or fabrication demonstration for each of the basic technological questions. In particular, it was shown that the expected critical mercury corrosion rate was tolerable, and that the crud-generation effects could indeed be controlled.

Considerable operating experience with both components and systems was obtained. A total of $\sim 35,000$ hr of testing was logged with the critical CRU turbo-machinery equipment; $\sim 21,000$ hr of this total were achieved with the latest CRU-V prototype design machinery. All of the critical factors relative to CRU space operation (repeatable injection startup, shock and vibration, automatic control, etc.) were demonstrated, with the exception of actual zero-gravity operation. Additionally, endurance runs of up to 6600 hr (CRU-IVM) and 4700 hr (CRU-V) were accomplished without failure or serious degradation.

TABLE 4
SNAP 2 MERCURY RANKINE SYSTEM FEASIBILITY SUMMARY

Technological Problem Area	Primary Demonstration or Verification	CY
Criticality of ZrH Reactor	Zero Power Critical Experiment	1957
Fabricability of Enriched Uranium-Zirconium Hydride Fuel Elements	SER Fuel Elements	1958
Load Capability-Stability of Mercury Lubricated Bearings	Prototype Bearing Component Tests	1958
High-Speed Operability of Turbo-alternator Assembly	500-hr CRU-I Test	1959
1200° F Power Operation of Reactor	SER Power Demonstration	1959
Zero-Gravity Boiling and Condensing	Off-Base Flight Test	1962
Protection of Alternator from Mercury Attack	CRU-V with Evacuated Stator Operation	1964
Long-life Capability of Turbine, Pump, and Alternator	6600-hr CRU-IVM-3 Test	1965
Mercury Corrosion Endurance Effects	2500-hr CRU-V-1A Test (4700-hr CRU-V-5C Test)	1963 (1966)
Stable Operation of MR System	PSM-1 Test	1964
Automatic Orbital Startup of MR System	PSM-3 Test	1965

For the once-through NaK-to-mercury boiler, operational performance of the prototype compact design was demonstrated, through a total of 29,000 hr of testing. One area of concern was the incomplete understanding of the deconditioning, low-boiling, heat-transfer phenomenon. However, based on the investigations undertaken, it was concluded that eliminating oil and maintaining leak tightness would allow fully conditioned boiler operation.

In a period of only 3 years, the NaK thermoelectric pump was developed very satisfactorily, with a total of 19,000 hr of testing experienced. Similarly, the radiator-condenser also was developed to a favorable point, with both performance and structural capabilities demonstrated. The other associated components (expansion compensator, injection tank, pressure regulator, parasitic

control, valves, etc.), although not developed to the status of flight configuration, did not appear to require much further development effort.

Relative to the system experience, three developmental installations, PSM-1, PSM-2, and PSM-3, were fabricated and tested to study packaging, structural, and simulated zero-gravity operational characteristics. No difficult or unique problems were encountered with the various developmental system test sequences, while both stability of operation and reliability of startup were defined over a wide range of conditions. The major remaining development effort would have been to extend the demonstrated component lifetime capabilities to 10,000 hr and greater, and to demonstrate the endurance performance of an integrated MR system.

2. System Potential and Applications

A basic 5-kwe Mercury Rankine System, utilizing a reactor heat source, is illustrated in Figure 15. System studies have developed a multiple-PCS approach for greater power and reliability. The basic 3- to 5-kwe PCS would be used in multiple building blocks to cover a power range above 3 kwe and up to at least 25 kwe. Combinations of active and redundant PCS subsystems could be assembled to provide a variety of discrete power levels and reliabilities within that power range. Normally, each PCS would have its own boiler, TE pump, CRU module, start-component, and RC assembly; however, shared combinations of the components could be considered, relative to specific performance and reliability objectives.

The PCS modules, whether or not totally independent of each other with regard to the mentioned components, would share a common NaK loop in a reactor-heated system, and a combined heat-source boiler in the isotope-heated system. Figure 16 is a schematic of a multiple-PCS reactor system that is basically applicable also to an isotope system.

A weight tabulation for such a system is given in Table 5. How a multiple-PCS power source might be assembled realistically is illustrated in Figure 17, which shows four active and one redundant modules, mounted in a typical flight configuration on an extensible boom.

Studies such as this revealed that the MRP system is exceedingly adaptable to various design configurations, and that only small efficiency penalties are incurred by the use of multiple-PCS assemblies, up to at least 40 kwe. In the

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Figure 15. 5-kwe Mercury Rankine System

ESSENTIAL FEATURES OF MULTIPLE MODULE CONCEPT:

- INDIVIDUAL MODULES ALLOW "BUILDING BLOCK" APPROACH TO COVER A WIDE VARIETY OF APPLICATIONS
- ALL MODULES IDENTICAL - DEVELOPMENT COSTS REDUCED
- ALL MODULES ARE INDEPENDENT OF EACH OTHER - ENHANCING SYSTEM RELIABILITY
- WIDE POWER RANGE COVERED (3-50 kwe) WITH SAME MODULE SIZE
- REDUNDANT MODULES MAY BE ADDED TO ACHIEVE DESIRED RELIABILITY

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Figure 16. Mercury Rankine System Multiple Power Conversion System Application

TABLE 5
TYPICAL UNSHIELDED 10-kwe MERCURY RANKINE
SYSTEM WEIGHT TABULATION

System Components	Weight (lb)	
	2 Active Modules	Redundant Module
Reactor	600	-
NaK Loop	160	75
Boiler	60	30
CRU Module	180	90
Startup Equipment Mercury Piping	350	145
Electrical Control and Conditioning Equipment	200	120
Structure	50	20
Radiator (Shared Fin)	350	90
Miscellaneous	100	30
	<u>2050</u>	<u>600</u>

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Figure 17. Multiple Power Conversion
System Configuration

power range of 15 to 50 kwe, especially, the MRP reactor - multiple-PCS system was found to offer significantly superior system weights and radiator areas, when compared with other power conversion concepts.

Additionally, it was concluded that a life capability of at least 10,000 hr had been demonstrated with components developed on the program; and that a few minor developmental modifications (some of them already identified and investigated) should permit operational life times of 2 to 3 yr.

IV. SNAP 8

Anticipated increased mission power requirements, coupled with increased booster-system payload capability, prompted a second generation of SNAP nuclear power supply systems. SNAP 8 was such a system, designed to provide 30 to 60 kw of electrical power, with a 10,000-hr life, for space vehicles. It was based on the SNAP 2 concept of a NaK-cooled, uranium-zirconium hydride reactor, coupled to a mercury Rankine cycle power conversion system. The SNAP 8 system was a joint AEC-NASA-sponsored development program, starting in 1960.

A. PROGRAM OBJECTIVES

The objectives of the SNAP 8 development work were to: (1) develop the SNAP 8 reactor, including its associated controls, shield, and safety features, and (2) test the complete SNAP 8 System (i. e., reactor coupled to power conversion equipment). The power conversion equipment was under development by the Aerojet General Corporation, under contract to NASA.⁽³⁹⁾ The general objectives and operational constraints are shown in Table 6.

TABLE 6
SNAP 8 GENERAL OBJECTIVES

Planned application	Electrical power
Safety criteria	Subcritical during launch Orbital startup Orbital shutdown at end of power conversion system(PCS) life, backed up by reentry shutdown mechanism
Life	10,000 hr
Power conversion	Mercury Rankine cycle Turboalternator being developed by Aerojet General Corp., under contract to NASA
Electrical power output	30 kw (capable of extension to 60 kw)
Reactor thermal power	300 kw (capable of operation at 600 kw)
Control mode of operation	System operates at full power. Electrical power not used by payload is dissipated in parasitic load. The reactor control system maintains the reactor coolant outlet temperature around 1300°F by stepping reflector control elements in or out, as required.

TABLE 7
S8ER OPERATIONAL HIGHLIGHTS

Modification of Bldg 010 to accommodate S8ER initiated	August 16, 1961	Completed 6 months of continuous operation at 450 kw or greater with a core outlet temperature of 1300°F	March 18, 1965
Critical experiments	September to December 1962		
Water immersion experiments	January to March 1963	Completed 365 days of operation at 400 kw or greater with a nominal core outlet temperature of 1300°F	March 24, 1965
Dry criticality achieved in Bldg 010	May 19, 1963	Completed longest continuous power operation (4542 hours) at a core outlet temperature of 1300°F	
Wet criticality achieved	June 23, 1963	Power levels were 450 kw or greater during this operation	March 26, 1965
Initial significant power operation	November 22, 1963		
Initial operation at 600 kw/1300°F	December 11, 1963	Completed 10 000 hours of operation at 400 kw or greater	April 1, 1965
Exceeded the previous total 497,000 kwh of energy generated by SNAP reactors	February 26, 1964	Completed 12,000 hours of operations	April 11, 1965
Completed 60 days of operation at 450 kw or greater with a core outlet temperature of 1300°F	April 28, 1964	Completed 5,000 hours of continuous operation at 450 kw or greater with a core outlet temperature of 1200°F or greater	April 14, 1965
Completed 1000 hours of continuous operation at 600 kw with a core outlet temperature of 1300°F	September 2, 1964	Completed 5016.5 hours of continuous critical operations, all except 2.2 hours of which were at 450 kw with a core outlet temperature of 1200°F or greater	April 15, 1965
Exceeded longest continuous operation of any other SNAP reactor (1152 hr)	November 5, 1964	Completed 12,080.6 hours of operations	April 15, 1965
Exceeded the AI operating continuity record of 59 days previously held by Piqua Nuclear Power Facility (PNPF)	November 16, 1964	Released a total of 5,154,332 kwh of thermal energy	April 15, 1965
Completed 60 days of continuous operation at 450 kw with a core outlet temperature of 1300°F	November 17, 1964	Achieved a Time Operated Efficiency of 72.03% for the period from noon of the day permission was received from the AEC for operation, May 17, 1963, through final reactor shutdown at 0604 on April 15, 1965	April 15, 1965
Completed 90 days of continuous operation at 450 kw with a core outlet temperature of 1300°F	December 17, 1964		
Completed 10,000 hours of operations	January 18, 1965	Achieved a Time Operated Efficiency of 91.09% for the period from the start of the day of initial significant power operation, November 22, 1963 through final reactor shutdown at 0604 on April 15, 1965	April 15, 1965
Completed a total of 100 days of operation at 600 kw with a core outlet temperature of 1300°F	February 4, 1965		
Exceeded longest known continuous operation of any reactor (147 days set by Yankee Reactor)	February 12, 1965	Achieved a Time Operated Efficiency of 97.91% for the final year of operation from April 15, 1964 through final reactor shutdown at 0604 on April 15, 1965	April 15, 1965
Completed 365 days of critical operation at core outlet temperatures of 1200°F or greater	March 5, 1965		

B. PROGRAM HISTORY

The SNAP 8 reactor design evolved through a series of test reactors, pointing toward a flight system design. Changing program requirements led to variations in the design point that resulted in three conceptual designs in the latter phases. These systems included the Reference Zirconium Hydride Reactor, the Reactor Core Test, and the Space Power Facility (SPF) Reactor.

The SNAP 8 Program began early in 1960. The testing of the first SNAP 8 reactor, the SNAP 8 Experimental Reactor (S8ER), began in May 1963 and continued to April of 1965. A test of the SNAP 8 reactor with flight component design, the SNAP 8 Developmental Reactor (S8DR), was conducted from January 1969 to December 1969.

1. S8ER

The S8ER test was the first in a series of tests on the reference design SNAP 8 reactor only, with the purpose of verifying its capability of operating at 600 kw with a reactor coolant outlet temperature of 1300°F, for a period of 12,000 hr. A chronological listing of the test milestones is given in Table 7.

Post-test core disassembly and fuel element analysis indicated 80% of the 211 fuel elements had cladding cracks, even though the reactor had met or exceeded all test objectives. Detailed examination of selected fuel elements showed the cracks to have resulted primarily from over temperature swelling of the hydrided fuel, coupled with the low irradiated ductility of Hastelloy N cladding material.

2. S8DRM-1

This reactor system was an initial flight-configured environmental test (non-nuclear) of the reactor and flight-oriented hardware associated with the primary heat transfer system. Tests included thermal, thermal-shock, shock, and vibration under anticipated launch conditions. Successful qualification of all non-nuclear systems, including nine automatic startups, was achieved by April of 1965.

3. S8DR

The S8DR test employed an upgraded core design, based on the experience obtained in the S8ER test. In addition, the test of the reactor assembly was conducted under simulated space environmental conditions, with all components flight qualified and acceptance tested. As in the S8ER test, the reactor heat was transferred via primary and secondary circulating NaK systems, and ultimately rejected to the atmosphere. The modifications to the S8ER core design incorporated into the S8DR consisted of:

- 1) Change in fuel element hydrogen permeation barrier to a thermal-cycle resistance coating
- 2) Increased radial and axial fuel-rod-cladding clearances
- 3) Redesigned fuel element end caps, to alleviate stress cracking noted in S8ER elements
- 4) Modified coolant core flow profile.

Facility preparations were completed in January 1968. Acceptance testing of the reactor assembly was completed in May 1968. Dry criticality was achieved on June 22, 1968, with 169 elements. Non-nuclear checkout of the S8DR system uncovered a NaK leak in the primary loop EM pump throat, which was replaced. Physics and control drum calibration tests were performed. Installation of the fueled reactor and reflector in the vacuum vessel was completed in November 1968. Dry and wet criticality tests and five automatic startups were performed in December 1968. Power operation was initiated in January 1969.

The initial power operation of S8DR consisted of a 1000-hr, 600-kwt, 1300°F outlet run, with a subsequent adjustment to a planned 1-Mwt, 1150°F outlet run for 430 hr. Following the megawatt run, the power and temperature conditions were again set to 600-kwt, 1300°F outlet conditions.

During the initial 600-kwt run, there were apparent temperature and power fluctuations of small magnitude. These ceased shortly before the start of the megawatt run. About 500 hr after the end of the megawatt run, it was noted that the fission-products activity in the coolant began to increase, as indicated by cover-gas activity monitors in the primary system surge tank and confirmed

by radiochemical analyses of primary NaK coolant samples. At about the same time, the reactivity loss rate from the reactor also began to increase. Additionally, step increases in reactivity loss rate associated with reactor scrams were observed in the time period from early May to late July, 1969.

By early July, it had become apparent that a number of fuel element failures had occurred in the S8DR core. The coolant outlet temperature was reduced from the design value of 1300° F to 1200° F, and an extended 600-kwt power run continued. By late summer of 1969, a noticeable decrease in both the coolant activity and reactivity loss rate increases had occurred.

By late fall of 1969, an administrative decision was made to shut down the S8DR reactor on December 15, 1969, to determine the cause of fuel element ruptures. After 7,000 hr of power operation, the reactor was shut down, and allowed to cool for several months prior to removal of the unitized core assembly from the Bldg. 059 test facility, for shipment to AIHL in late March of 1970.

S8DR reactor operations can be summarized as follows:

- 1) Reactor critical for 7532 hr, with 1670 hr at 600 kwt and a reactor outlet NaK temperature 1280° F, and 6688 hr at 600 kwt and outlet temperatures >1000° F.
- 2) Total energy released was 4.3×10^6 kwh.

Post-test visual examination of core externals indicated components were in excellent condition. A total of 72 fuel elements were found to have cladding cracks typical of stress-rupture phenomena. The fuel element failures were attributed to excessive cladding strain due to fuel swelling resulting from temperatures in excess of design values. Probable sources of the high temperatures were discovered in a series of post-test hydraulic experiments, in which it was found that entrainment of surge-tank cover gas occurred. This was coupled with the observation of two standing stable vortices in the inlet plenum. The vortices tended to agglomerate small entrained gas bubbles and periodically release larger bubbles, which passed through the core in the region of maximum fuel element damage. In addition, stable clustering occurred, causing additional flow decreases, and hence, fuel temperature increases. This resulted in fuel temperatures estimated to be over 100° F hotter than the design value.

Figure 18. S8ER Reactor Assembly View

Figure 19. S8ER Core and Reflector Section and Plan View

C. SYSTEM DESCRIPTIONS

1. S8ER

The S8ER was a compact reactor, designed to provide a test of the SNAP 8 fuel under the reference conditions for the system. The core contained 211 uranium-zirconium hydride fuel elements and was reflected radially by beryllium, as shown in Figures 18 and 19. Cooling was provided by NaK (sodium-potassium eutectic) which was circulated through the core axially from bottom to top. An external heat transfer system removed the heat from the NaK. Reactor control was provided by six movable control drums which adjusted the neutron flux leakage from the reactor. The design data are summarized in Table 8. The characteristics of the major components are summarized in the following sections.

a. Fuel Elements

The reactor contained 211 fuel-moderator elements. Each element consisted of an individual fuel rod, a hydrogen diffusion barrier containing burnable poison, exterior cladding, end caps, and grid-plate indexing pins (see Figure 20).

The rods were composed of enriched (93.15% U^{235}) fuel and hydrogen moderator, in the form of a solid zirconium-uranium alloy (10% uranium), hydrided to a hydrogen density of 6×10^{22} atoms/cc. The rods were 0.532 in. in diameter, and 14.0 in. long. The hydrogen diffusion barrier was a 0.003-in. thick ceramic coating (AI-8763D), applied on the inside surface of the cladding, to prevent the loss of the hydrogen moderator. A burnable poison (3.34 mg Sm_2O_3 per linear inch of fuel element) was added to the ceramic coating, to compensate for reactivity loss due to buildup of fission-product poison. The cladding consisted of 0.540-in. ID Hastelloy N tubing with a 0.010-in. wall thickness. The cladding envelope was completed by Hastelloy end caps which were seal welded to the tubing. The manufacturing process ensured that a continuous diffusion barrier was provided on the inside surface of the cladding. The finished fuel element was 0.560 in. in diameter and 14.47 in. long (excluding the 0.140-in. diameter positioning pins located at the ends of each element). The 211 elements were arranged in a triangular pattern on 0.570 in. centers. The assembly resulted in a hexagonal right cylinder, with slightly chopped corners, 8 in. across flats and 14.47 in. long.

TABLE 8
S8ER DESIGN DATA SUMMARY

Design Parameters

Design power level	600 kwt
Life	10,000 hr
Primary coolant	NaK 78 eutectic
Secondary coolant	NaK 78 eutectic
Number of fuel elements	211
Fuel loading	6.56 kg of U enriched 93.15%
Reflector	3 in. nominal thickness Be
Control	6 rotating control and safety drums

Fuel Elements

Composition	Hydrided Zr-U alloy
Hydrogen concentration	6×10^{22} atoms/cc
Alloy	90 wt % Zr - 10 wt % U
Cladding (Hastelloy N)	0.010-in. walls and 0.250-in. end caps
Fuel rod diameter	0.532 in.
Fuel rod length	14.0 in.
H ₂ diffusion barrier	0.003-in.-thick ceramic coating on inside of cladding surface
Burnable poison	48 mg of Sm ₂ O ₃ per element
Maximum cladding temperature at design power level	1440°F
Maximum core temperature at design power level	1542°F

Reactor Core

Core vessel size	9.34 in. OD by 21 in. nominal height
Core vessel material	316 SS
Core vessel wall thickness	1/16 in.
Upper grid plate material	316 SS
Upper grid plate diameter	9-3/16 in.
Upper grid plate thickness	11/32 in.
Upper grid plate coolant passages	420 5/32-in. -diameter holes

TABLE 8 (Continued)
S8ER DESIGN DATA SUMMARY

Lower grid plate material	Hastelloy C
Lower grid plate diameter	9-3/16 in.
Lower grid plate thickness	5/16 in.
Lower grid plate coolant passages	420 1/8-in.-diameter holes
Internal reflector	18 BeO filler pieces
Internal reflector cladding	Hastelloy N
<u>Control and Safety Elements</u>	
Number of elements	6
Material	Beryllium
Length	14.5 in.
Drum radius of curvature	4.69 in.
Nominal thickness	3 in.
Weight of each control element (including support and shim)	26.5 lb
Radial bushing clearance (self aligning)	0.004 to 0.006 diametral
Friction torque	10 in. -lb
<u>Control Element Drives</u>	
Drive means	1 reversible a-c motor per control element (only one element can be inserted at one time)
Element rotation rate	0.055 rpm (2 ϕ /sec)
Element rotation range	105°
Time to rotate full range	319 sec
Coast rotation	0.033° (0.2 ϕ)
Rotation stops	Limit switches and mechanical stop
Overload protection	Slip clutch
Drive system backlash	None - taken out by scram spring
<u>Scram System</u>	
Scram power	Torsional spring
Scram torque	30 in. -lb total 20 in. -lb for acceleration
Scram activation	De-energizing of electromagnetic clutch

TABLE 8 (Continued)
S8ER DESIGN DATA SUMMARY

Primary Coolant

Flow rate	140 gpm
Reactor inlet temperature	1100°F
Reactor inlet pressure	38.7 psia
Reactor outlet temperature	1300°F
Reactor outlet pressure	36.8 psia
NaK inventory	315 lb

Secondary Coolant

Flow rate	140 gpm
Heat exchanger inlet temperature	1050°F
EM pump inlet pressure	30.4 psia
Heat exchanger outlet temperature	1250°F
EM pump outlet pressure	54.5 psia
NaK inventory	622 lb

Nuclear Characteristics

Lattice spacing	0.57 in. center-to-center
Mean prompt neutron lifetime	6.7 μ sec
Effective delayed neutron fraction	0.0077
Median fission energy	0.21 ev
Thermal flux	2×10^{12} n/cm ² /sec

b. Core Vessel and Assembly

The core and core-structure components, (i. e., fuel elements, grid plates, internal reflectors, and flow baffle plate) were integrated into a complete package by means of six vertical tie rods at the circumference of the core. The core assembly was fixed to the core vessel by the lower end of the tie rods, which screwed into brackets attached to the inner circumference of the lower plenum head of the core vessel. The core vessel was constructed of a cylinder of Type 316 stainless steel, 9-3/16 in. in ID and $\sim$ 15-1/2 in. in length, with a 1/16-in. thick wall.

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Figure 20. S8ER Fuel Element

Figure 21. S8ER Heat Transfer System P&I Diagram

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c. Reflector and Support

The core vessel was surrounded by 3 in. of beryllium (nominal thickness of equivalent annular cylinder) which formed the external radial reflector. The reflectors were ~14.5 in. long (approximately same as fuel element length), and were divided into six stationary and six movable sections. The portions of the reflector which formed the six movable sections (control drums) were segments of right circular cylinders with the axis outside the reflector assembly. Upper and lower bushings (shown on Figure 19) were located on the axis of each segment, permitting rotation of the control drums. Reactor control was accomplished by rotation of the control drums toward or away from the core, thereby changing the neutron economy and reactivity of the core. The control drums were also used as safety elements, to scram the reactor. Removable shim plates were provided on the control drums to permit adjustment of the control drum worth prior to reactor startup.

d. Control Drum Drives

The control drums were rotated by a motor-driven mechanism, located above the shutdown shield. This mechanism provided scram capability through a clutch drive which, when released, permitted the drums to be driven out to their shutdown position by springs. Snubbing action was provided at the end of the stroke. Position readout and limit-switch information were also included in this assembly.

e. Reactor Installation

The reactor was installed in a pressure vessel which provided a protective atmosphere of dry helium and containment for any accidental release of radioactive material. The vessel was located below grade level, to afford shielding for the reactor. The vessel also held a shutdown shield immediately above the reactor, followed by the control drive mechanisms, and a biological shield. The heat removal system, consisting of the primary and secondary NaK loops, was located in other vaults of the facility. The final heat rejection consisted of an air-blast heat exchanger for dumping heat to the atmosphere. The P&I diagram of this system is shown in Figure 21.

TABLE 9
S8DR CHARACTERISTICS

General		Control Drums	
Design Power Level	600 kw	Number of Drums	6
Reactor outlet temperature	1300°F	Material	Beryllium
Reactor inlet temperature	1100°F	Length	15.375 in (total reflector length - 18.5 in)
Average power density	37 kw/liter of core	Drum Radius of Curvature	4.5 in
Maximum Power Level	1 Mwt	Nominal Thickness	5.125 in
Reactor outlet temperature	1100°F	Drive Means	Electromagnetic stepping motor
Reactor inlet temperature	830°F	Rotation Range	0 to 109°
Design Life	12,000 hr	Full-In Stop	1.0 ± 1.0°
Primary Coolant	NaK	Full-Out Stop	135 ± 2.0
Secondary Coolant	NaK	Drive-System Backlash	±0.05°
Number of Fuel Elements	211	Scram Power	Torsional spring
Fuel Loading	8.2 kg U enriched to 93.15% U ²³⁵	Nuclear Characteristics	
Reflector	5.125 in nominal thickness	Mean Prompt Neutron Lifetime	8.4 μsec
Control	6 rotating Be reflector drums	Effective Delayed Neutron Fraction	0.0080
Reactor Core		Median Fission Energy	0.15 ev
Core Vessel Size	9.244 in ID by 32.2 in height to outlet line	Thermal Flux @ 600 kw (<1 ev)	3 × 10 ¹² n/cm ² sec
Material	316 stainless steel	Radial Peak to Average Power	1.29
Wall thickness	0.091 in	Axial Peak to Average Power	1.36
Grid Plates		Clean Wet Excess Reactivity	\$14.40
Coolant passages	420 (0.156 in diam)	NaK Reactivity Worth	\$0.25
Spacing of fuel element positioning holes	0.573 in (triangular lattice)	Total Isothermal Temperature Coeff of Reactivity	-0.18¢/F° at 70°F
Spacing of coolant holes	0.331 in (hexagonal lattice)	Power Coefficient of Reactivity	-0.20¢/F° at 1200°F
Upper grid plate		Rise to 600 kw Power and 1200°F Core Average Temperature	\$2.15
Material	316 stainless steel	Reactivity Inventory (12,000 hours)	
Diameter	9.217 in	Hot, end of life excess (600 kw)	\$5.45
Thickness	0.850 in	Samarium burnout	\$2.90
Lower grid plate		Fission product poisoning	
Material	Hastelloy C	Xc	-\$1.05
Diameter	9.217 in	Sm	\$1.02
Thickness	0.65 in	Other	\$1.17
Internal Reflectors		Axial hydrogen redistribution	\$0.65
Material	30 stainless-steel clad BeO and 12 solid stainless-steel pieces	Hydrogen loss	-\$5.15
Cladding thickness	0.030 in	U ²³⁵ burnup	\$0.66
Cladding material	316 stainless steel	Control Drum Worth	
Fuel Elements		Total - 6 drums	\$21.06
Fuel Rods		Single drum	\$3.90
Composition	Hydrided ZrU alloy	Maximum differential worth	5.75¢/deg
Fuel	10.5 wt % U	Thermal and Hydraulic Characteristics	
Enrichment	93.15 wt % U ²³⁵	Average Heat Flux (at 600 kw)	47,000 Btu/hr-ft ²
Hydrogen concentration	6.05 × 10 ²² atoms/cc	Fuel to Coolant Heat Transfer Area	43.6 ft ²
Diameter	0.529 in	Coolant Flow Area in Core	
Length	16.825 in	Cold	0.0614 ft ²
Axial hydrogen gap	0.240 in	Hot	0.0653 ft ²
Radial hydrogen gap	3.5 mils	Hydraulic Diameter (individual coolant channel)	0.00675 ft
H ₂ diffusion barrier	2-mil ceramic coating on inside of cladding surface	Mass Flow Through Core (600 kw and 200°F ΔT)	48,400 lb/hr
Burnable Poison	58 mg Sm ₂ O ₃ /element (in ceramic)	Average velocity through core, hot	4.6 ft/sec
Cladding (Hastelloy N)	0.540 in ID, 11.3 mils thick	Mass Flow Through Core (1000 kw)	60,000 lb/hr
Fuel Elements		Average velocity through core hot	5.6 ft/sec
Diameter	0.563 in		
Length	17.325 in (including positioning pins)		
Peak fuel temperature at design power level	1485°F		
Peak cladding temperature at design power level	1367°F		
Average fuel burnup (12,000 hr at 600 kw)	0.22 metal atom %		

All dimensions are at room temperature conditions unless otherwise specified

2. S8DR

The SNAP 8 Developmental Reactor was operated to provide a test of a revised fuel element design in a system built with flight-designed components. The fuel was longer, >2 in., as compared to S8ER, to provide more reactivity. Other changes were incorporated in the fuel design to accommodate fuel swelling, which had been encountered in S8ER. It was designed to operate at 600 kw power and 1300°F coolant outlet temperature, or alternatively at 1000 kw and 1100°F outlet temperature. It had a planned life of 12,000 hr, which was terminated prematurely by a programmatic decision. The test was originally designed to be run in conjunction with the power conversion system, as an integrated unit in a breadboard configuration, but this feature was deleted. A summary of the principal characteristics of the reactor are given in Table 9. An overall view of the reactor is given in Figure 22.

a. Fuel and Cladding

The fuel elements were designed to alleviate the element failure problem encountered in the S8ER. Also, because of the increased swelling rate which occurs with the presence of a β -phase zirconium hydride fuel material, the first appearance of a β -phase material was to be used as an end-of-life design criteria.

Design limitations were imposed to minimize cladding failure from the following possible failure mechanisms:

- 1) Creep-induced strain to the point of rupture
- 2) Creep collapse due to cladding-wall pressure differential
- 3) Fatigue due to thermal cycling during unscheduled reactor scrams and scheduled startups.

External pressure on the elements (reactor coolant pressure) was limited to 35 psia during 1300°F coolant temperature operation. For the 1100°F outlet temperature operation (1 Mw condition), or any lower temperature, the reactor pressure would be increased to 50 psia.

The S8DR core contained 211 fuel elements, cooled by NaK-78 flowing axially between them. Each fuel element consisted of a cylindrical rod of fuel-moderator clad in Hastelloy-N tubing, as illustrated in Figure 23.

Figure 22. S8DR Ground Test Assembly

The fuel-moderator material was composed of a hydrided alloy of zirconium and fully enriched ($93.15 \pm 0.15\%$ U^{235}) uranium. The uranium content constituted 10.5 ± 0.3 wt % of the unhydrided alloy. This material also contained 0.15 ± 0.05 wt % carbon, added in the form of zirconium carbide (ZrC) prior to the initial arc melting of the zirconium-uranium alloy. This carbon additive resulted in a refinement of the alloy grain structure, and a consequent improvement in the production yield during the hydriding process.

The reactor core vessel was designed to minimum weight for space application. The basic design was modified to allow thermocouple penetrations to the upper plenum. The allowable operating stresses for the core vessel design were based on the criteria of 0.2% creep during the 12,000-hr reactor lifetime. Pipe reactions at the inlet and outlet nozzles were not allowed to exceed 50 ft-lb.

Calculational methods from the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, and Section VIII, Unfired Pressure Vessels, were used in the design of the reactor vessel. The reactor core vessel was a right circular cylinder, consisting of a lower plenum head, an intermediate cylinder, and an upper plenum head, all fabricated from Type 316 stainless steel. The vessel (without the upper head) is shown in Figure 24.

The lower plenum head had a flat external bottom and an 18-in. radius convex internal bottom surface. This head had internal brackets for mounting the flow distributor, the baffle plate, and the lower grid plate. An external mounting ring was used to couple the core vessel to the support structure.

The intermediate cylinder, with a nominal wall thickness of 0.091 in. (0.085 in. minimum), was welded to the upper and lower plenum heads. It had internal lugs to support and position the upper grid plate, and external brackets on which to mount the external reflector and control system.

The upper plenum head contained a coolant outlet nozzle, 3 holddown lugs for the upper grid plate, 3 support lugs for the thermocouple bridge, and 27 thermocouples to measure coolant temperature at the reactor outlet. This head was composed of a reducer section which connected the intermediate cylinder to a 5-in. Schedule 10 pipe tee which contained the 2-in. outlet nozzle.

Figure 23. S8DR Fuel Element

Figure 24. S8DR Reactor Core Vessel

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The other end of the tee was connected to the double-bulkhead thermocouple penetration assembly. The thermocouples in the upper plenum were Chromel-Alumel, MgO-packed in Type 316 stainless steel sheaths. Each thermocouple sheath was brazed at both bulkheads where it penetrated the plenum head. Inside the plenum, the thermocouples were bundled in a 1-in. pipe, and then distributed on a bridge to locations at the outlet of the core coolant channels. The core vessel was designed for 35 psi operation with an inlet temperature of 1150°F and an outlet temperature of 1350°F.

Upper and lower grid plates were used within the vessel to position the fuel elements. A flow distributor and baffle plate were located beneath the lower grid plate to break up the flow coming into the lower plenum, prevent vortexing, and provide a constant pressure profile across the lower face of the flow baffle plate.

There were 42 internal reflector pieces in the core to fill the void space between the hexagonal core array and the cylindrical vessel. Thirty were of beryllium oxide, clad in Type 316 stainless steel, while the balance were solid Type 316 stainless steel.

c. Reactor Reflector and Control Drums

The reactor core vessel was surrounded by a beryllium reflector, which was divided into two stationary and six movable sections. Reactor control was provided through the movable sections. Figure 25 shows a drawing of the reactor with reflector and drive assembly.

The beryllium reflector assembly had a nominal thickness of 5-1/8 in. and a length of 18-1/2 in. The reflector assembly was divided axially into two halves. Each half consisted of a stationary reflector and three control drums, together with drives, bearings, and support brackets. The reflector assemblies were attached to and supported by brackets which were an integral part of the reactor core vessel. Latching mechanisms were provided at the upper end of the assembly to lock the reflector halves together.

The length of the reflector assembly for the S8DR had been increased over that used for the S8ER, to provide additional reactivity. This increase was provided by stationary sections above and below the control drums, which were essentially the same as the S8ER drums.

Figure 25. Schematic of S8DR Reflector and Drive Assembly

The drums were shaped as segments of right-circular cylinders with 4-1/2 in. radii. The basic drum had a length of 15.3 in. and a maximum thickness of 2-1/8 in. A 3-in. beryllium shim was attached to the back of each drum to make up the total 5-1/8 in. thickness. Mechanical stops for the drum were positioned at $1^\circ \pm 1^\circ$ from full in and at $135^\circ \pm 2^\circ$ for the out position. Each drum contacted a snubber at 109° from full-in.

All beryllium surfaces were anodized, to provide them with a high emissivity. The outside beryllium surfaces (those not facing the reactor vessel) were coated with an additional high-emissivity coating, since the emissivity of the anodized coating had been found to degrade slightly at the high temperature and vacuum conditions of the S8DR environment. The reflector assembly was designed to limit the beryllium temperatures to $<1350^\circ\text{F}$.

The anodized coating on the similar S8ER reflectors served also to inhibit corrosion. Since all high-temperature operation for S8DR performed with the reactor in a 10^{-5} torr vacuum, no beryllium corrosion was anticipated.

Each control drum rotated in a pair of upper and lower self-aligning bearings. These bearings were of a journal type, with the journal bushings held in a spherical cartridge which, in turn, could move in a socket to provide self-alignment. Contact surfaces between the ball and socket were coated with aluminum oxide, and then covered with a molybdenum disulfide base dry-film lubricant. The shaft which ran in the carbon-graphite ball was also coated with a dry-film lubricant. The bearings were designed for operation at 1150°F in an environmental pressure of 10^{-5} torr or less.

d. Control-Drum Actuator Assembly

The S8DR control-drum actuators used the same operational principal as did the SNAP 10 actuators. The S8DR actuators, however, were designed for the higher operating temperatures of the SNAP 8 environment.

The control-drum actuator was an 8-pole reluctance motor which produced torque by the interaction of magnetic forces between stationary teeth on the fixed poles and the notched teeth cut into the cylindrical rotor. The field coils for the poles were sequentially energized to produce a revolving magnetic field

which turned the rotor. The stator-pole tooth alignment was such that, as each pole was energized, the rotor teeth were in a position to provide maximum torque. While not energized, the rotor was held in position by a brake mechanism.

The control drum drives operated in conjunction with a scram mechanism that would rotate the control drums outward rapidly when an electromagnetic clutch was released. A mechanical snubber restrained the drum at the end of its scram motion.

e. Heat Transfer System

The heat transfer system consisted of primary and secondary NaK loops. Final heat rejection was by means of an air blast heat exchanger. EM pumps were used to circulate the coolant in each of the loops.

f. Neutron Shield

A prototype SNAP 8 shadow shield was mounted directly underneath the reactor. The purpose of the shield was to reduce the neutron and gamma flux at flight system instruments and sensitive components.

The shield used for the S8DR ground test was modified to accommodate test conditions. The lithium used was the Li^7 isotope, which had a slightly smaller neutron absorption cross section than Li^6 , and offset the added neutron flux reaching the shield by scattering. To further offset the added load due to ground-test conditions, the distance between reactor and shield was increased slightly.

g. Facility

The reactor test was conducted in Building 059 of the Atomics International Santa Susanna Field Laboratory. The system was installed in a large vault, located below grade level. The reactor was installed within a vacuum chamber in a shielded test pit below the test floor level. A vacuum of 10^{-5} torr was maintained in the chamber throughout the test.

3. Conceptual SNAP 8 Reactor Designs

Three conceptual SNAP 8 reactor designs were developed during the period of 1969 through 1971. These reactors were designated as the Zirconium Hydride Reference Reactor, the Reactor Core Test, and the Space Power Facility Reactor. Their design points may be summarized as follows:

	<u>Reference ZrH Reactor</u>	<u>Reactor Core Test</u>	<u>Space Power Facility Reactor</u>
Power (kwt)	600	600	300
Outlet Temperature (°F)	1250	1200	1200
Life	20,000 hr	20,000 hr	5 years

The Reference Zirconium Hydride Reactor was designed to meet system requirements for a 25-kwe thermoelectric power conversion system. The reflector control and shield systems were designed to provide radiation protection over a full 4π solid angle around the reactor, rather than only a shadow shield.

The Reactor Core Test was designed as a test reactor for the resolution of fuel element design problems still outstanding after the S8DR operation. It was the first SNAP reactor to utilize finned fuel elements to promote coolant cross flow.

The Space Power Facility was designed for operation in conjunction with a NASA program, for operation at the Plumbrook Space Power Facility. This reactor represented the last design effort in the SNAP 8 family of reactors.

D. SUMMARY AND CONCLUSIONS

The SNAP 8 system requirements led to development work in the limiting area of uranium-zirconium hydride fuel performance. Laboratory studies, individual fuel rod irradiations, and test reactor operations were used to investigate the problems associated with the fuel element design. These problems resulted from the combination of fuel material swelling at temperatures in excess of 1400°F and the loss of ductility of Hastelloy N cladding material under irradiation. Another problem area was the cyclic clustering and declustering of the fuel elements under the combined effects of thermal expansion and hydrogen redistribution. Fuel elements which, on the other hand, operated within their

TABLE 10
S8ER OPERATION SUMMARY

Operation	Date
Initiated modification of Bldg 010 for use with S8ER	August 16, 1961
S8ER dry critical experiments, Bldg 373	September-December 1962
S8ER water immersion experiments, Bldg 012	January-March 1963
Initiated dry critical testing, Bldg 010	May 19, 1963
Initiated wet critical testing	June 23, 1963
Initial significant power operation	November 22, 1963
Initial operation at 600 kwt/1300° F	December 11, 1963
Completed a total of 60 days of operation at 450 kwt and 1300° F, to meet chief program endurance objective	April 28, 1964
Completed power test operations, Bldg 010	April 15, 1965

S8ER OVERALL ACHIEVEMENTS

Total operating time	500 days (12,000 hr)
Total time above 400 kwt at 1300° F	365 days
Time at 600 kwt, 1300° F	100 days
Longest continuous run (September 18, 1964 to April 15, 1965)	>5000 hr
Energy release	>5.1 x 10 ⁶ kwh
Operating Efficiency (from first power operation, November 22, 1963, to completion of power operation, April 15, 1965)	91%

design environment, did not swell, did not cycle, and displayed a leak rate consistent with out-of-reactor tests. Hastelloy X had been selected as the preferred cladding material, and fins have been added to the outside of the element to promote cross-channel flow.

1. S8ER

The SNAP 8 reactor was designed to produce 600 kw of thermal power with a coolant outlet temperature of 1300° F. The SNAP 8 Experimental Reactor (S8ER), the first in a series of SNAP 8 nuclear tests, was operated to determine reactor performance characteristics and to demonstrate sustained reactor operation at power. Nuclear testing was performed from May 19, 1963 to April 15, 1965 at the S8ER Test Facility, located at the Atomics International Nuclear Field Laboratory in Ventura County, California.

Operation of the S8ER demonstrated:

- 1) Sustained power operation capability. The S8ER released $>5.1 \times 10^6$ kwh of energy during 500 days of nuclear operation, including 100 days at 600 kwt and 1300° F outlet temperature, and 365 days at more than 400 kwt and 1300° F outlet temperature.
- 2) Static and dynamic stability.
- 3) Acceptable integrity of the interim hydrogen barrier coating on the fuel cladding tubes.
- 4) Capability of tolerating rapid changes in power level. During the power coefficient measurements, the core was subjected to ~ 115 rapid changes in power (100 kwt in 1 min).
- 5) Verification of SNAP 8 reference design.

Summaries of the S8ER operations and experimental results are shown in Table 10.

2. S8DR

The S8DR test established the flight-readiness capability of the ex-core components of the reactor assembly. While the fuel elements exhibited failure modes similar to the S8ER elements, the failure rate was significantly reduced.

At the time of test termination, the fission gas levels had stabilized at an equilibrium value, and the excess reactivity loss rate could be linearly extrapolated to extend beyond design EOL values. There is every reason to expect that continued operation would have allowed the S8DR to fulfill all its design requirements, as did the S8ER tests.

A summary of the S8DR operating data is shown in Table 11.

TABLE 11
S8DR OPERATION SUMMARY

Critical (hr)	7533
Time at Power (300 kwt) (hr)	7023
Time at 1000 kwt (hr)	429
Time at Nominal 600 kwt (hr)	6688
At 1300° F	1670
At >1200° F	5575
At >1100° F	6688
Total Energy Release (kwht)	4.35×10^6

V. SNAP 4

A. INTRODUCTION

Analysis, development, and design activities accomplished on this program provided sufficient information to permit preliminary design selection of the SNAP 4 Prototype Reactor (S4PR) system and its components. The program was at a stage where final design, fabrication, assembly, and operation of a demonstration system was the next logical step. The system to be constructed would have provided a self-sufficient demonstration of the basic features and performance of the nuclear steam source, and would have allowed for ultimate demonstration of the complete power system capability.

This section is devoted to a summary physical and performance description of the system and components which would have been constructed, in order to provide the operational demonstration.

B. GENERAL

The complete SNAP 4 Power Generating System consisted of a reflector-controlled reactor, fueled with uranium-zirconium hydride, in a direct-cycle, saturated-steam, power-conversion system. A compact, high-speed, 12,000-rpm turbogenerator would provide 400-Hz electrical power, or the reactor-steam source could be connected to a 60-Hz turbogenerator unit or to a marine propulsion turbine. Process loops required to support operation of the system provided the following:

- 1) Separation of the water-steam mixture from the reactor, and recirculation of the high-pressure water to the reactor
- 2) Condensation of the steam after power extraction, and return of the condensate to the reactor
- 3) Filtration and deionization of the process fluid, to prevent buildup of corrosion products in the process system and subsequent fouling of the heat transfer surfaces, and gas handling, to recombine the radiolytically dissociated hydrogen and oxygen, and return the water and excess hydrogen to the system, to eliminate the need for water makeup and maintain the desired hydrogen overpressure to minimize the radiolytic decomposition rate.

The basic system which was to be constructed would have been a complete, self-sufficient nuclear steam demonstration of the components and subsystems which would be common to all ultimate applications of the power system. This would include the reactor, the basic activation and shutdown shield, and the recirculation, condensate, purification, and gas handling process systems. In order to operate the demonstration system, conventional reactor experiment controls and ultimate heat rejection equipment would have been provided. The basic system was designed for a thermal power of 12 Mw, and would contain compact lightweight valves and turbine-driven pumps, and would be hermetically sealed. Components and packaging of the prototype system would be as nearly identical to a deliverable prototype power system as possible; however, it would be necessary to provide certain special design features to gather experimental data and to make adjustments to developmental equipment. In addition, the design of the system would incorporate features to permit the adaptation of developmental power conversion, compact air-cooled heat rejection, and fully automatic control equipment concurrent with, or subsequent to, the installation of the basic reactor-steam source.

C. REACTOR

The reference SNAP 4 reactor was designed for a thermal power of 12 Mw, and a life of 12 Mwt-years. This power was to be generated in the form of 1200 psia, 10% quality steam. The basic fuel-moderator material was a homogeneous alloy of uranium and zirconium, hydrided to a nominal N_H of 5.35×10^{22} atoms H/cc fuel. This stable, hydrided alloy results in an extremely compact lightweight reactor with an unusually large fuel volume fraction. The compact configuration permits reflector control over a broad range of reactivity, and the solid hydrogenous moderator circumvents the reactivity fluctuation problem associated with coolant density variations experienced by other boiling water reactors.

Radial and axial power flattening were to be accomplished by the incorporation of a nuclear poison in the fuel alloy, prior to hydriding. The fuel rods were segmented within the cladding to permit axial flattening. By varying the composition of fuel and poison in two regions within the core, a peak-to-average power ratio of 1.3 could readily be obtained. Flattening the power enhanced the worth of

control elements, permitted a higher average core power density, and resulted in better overall fuel utilization. Burnable poisons were also to be used to reduce reactivity swing due to burnup to $<2\% \Delta k/k$ over core life.

Table 12 summarizes the physical and operating characteristics of the SNAP 4 Prototype Reactor (S4PR), and Figures 26 through 30 show the general arrangement of the reactor and associated process equipment for this basic system. Design details of the major components are described in the following sections.

1. Reactor Core

The reactor core occupied a volume of $\sim 1 \text{ ft}^3$, and had a length-to-diameter ratio of 1.5. The core contained 583 stainless steel clad, 0.440-in. OD, cylindrical fuel rods with 17 in. of active length. Each element contained four slugs of hydrided fuel, 0.408 in. in diameter; and the cladding was thin-wall (0.015 in.) Type 347 stainless steel tubing. Thermal bonding of the fuel to the cladding was obtained by maintaining an interference (shrink) fit throughout the operating cycle. Tubular end plugs were welded to the cladding at each end of the element, and portions of the tubular walls were machined away to allow coolant to flow through these openings and out the end of the tubular element. Minimum spacing of 0.022 in. between adjacent fuel elements was maintained under operating conditions by spirally wrapped wires on alternate fuel elements. The fuel element assembly is shown in Figure 31.

The top tubular end plugs were expanded into holes in the top grid plate, and locked the fuel elements in place. The bottom end plugs were free to expand axially into the lower grid plate. The core was subdivided into seven fuel element clusters — a central hexagonal cluster containing 127 fuel elements, and six outer, trapezoidal-shaped clusters, each containing 76 fuel elements. The central cluster contained 37 fuel elements which were replaceable and could be used for criticality shimming. The central cluster was positioned and supported structurally by the outer clusters, and the outer clusters were located and supported by the control drum bearing ring. The lower grid plate of each cluster was secured to the ends of widely spaced fuel elements. This construction permitted independent thermal expansion of each fuel element during reactor operation, and permitted removal and replacement of individual fuel element clusters.

TABLE 12
CHARACTERISTICS OF THE S4PR
(Sheet 2 of 3)

<u>Operating Conditions</u>	
Coolant flow rate	4.1×10^5 lb/hr
Coolant inlet pressure	1235 psia
Coolant outlet pressure	1212 psia
Coolant inlet temperature	540°F
Coolant outlet temperature	570°F
Recirculation ratio	10:1 (10% exit quality)
Average cladding temperature	667°F
Average fuel temperature	879°F
Maximum fuel temperature	1056°F
Heat transfer area	95 ft ²
Average heat flux	431,000 Btu/hr-ft ²
Maximum heat flux	540,000 Btu/hr-ft ²
Burnout safety factor (minimum)	3
Average power density in fuel	15.7 Mw/ft ³
Coolant mass velocity	2.85×10^6 lb/hr-ft ²
Coolant passage hydraulic diameter	0.0937 in.
Reactor pressure drop	22.8 psi
Reactor pumping power	21.7 water HP
<u>Nuclear Control Requirements</u>	
	<u>% $\Delta k/k$</u>
Minimum shutdown margin	2.0
Transition from cold to hot-and-boiling (core N_H change)	4.81
Transition from cold to hot-and-boiling (axial reflector)	0.50
Doppler effect (core resonance absorption)	0.57
Fuel element elongation	0.16
Grid plate expansion	0.12
Thermal flux spectrum temperature shift	0.04
Fuel element cell effect	negligible
Hydrogen loss from fuel	negligible
Equilibrium xenon poisoning (midlife)	2.0
Burnup control requirement	1.4
Total	11.60

TABLE 12
CHARACTERISTICS OF THE S4PR
(Sheet 3 of 3)

Nuclear control availability (12 reflector drums)		12.50 - cold 13.2 - hot
<u>Coefficients of Reactivity (Reactor Operating Conditions)</u>		
	<u>$\Delta k/k - ^\circ F$</u>	<u>$\epsilon / ^\circ F$</u>
Fuel elongation coefficient	-0.20×10^{-5}	-0.025
Core coolant expulsion coefficient	-0.57×10^{-5}	-0.071
Neutron spectral coefficient	-0.03×10^{-5}	-0.004
Doppler coefficient	-0.54×10^{-5}	<u>-0.069</u>
Net Prompt Coefficient	-1.3×10^{-5}	-0.17
Lower grid plate coefficient	-0.055×10^{-5}	-0.007
Upper grid plate coefficient	-0.133×10^{-5}	-0.017
Power coefficient	$-0.0044 \Delta k/k - \text{Mwt}$	$-55 \epsilon / \text{Mwt}$
Steam void coefficient	$-0.165 \Delta k/k$ per fractional void change	
Core hydrogen concentration coefficient (N_H in terms of $10^{24}H$ atoms/cc)	$+9.0 \Delta k/k - \Delta N_H$

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Figure 26. S4PR Reactor Vessel Cross Section

7564-0218

Figure 27. S4PR Nuclear Steam Generator and Container Assembly, Module No. 1

7564-0219

Figure 28. S4PR Water Purification Equipment, Module No. 2

7564-0220

Figure 29. S4PR Power Conversion Equipment, Module No. 3

Figure 30. S4PR Plant Arrangement

7564-0221

Figure 31. S4PR Fuel Element Assembly

7564-0241

The core structure unit included fuel elements and grid plates, control drums and their associated hardware, snubbing devices, and reflector cusps and filler pieces around the control drums. The entire unit was encased in a cylindrical plate of 5/8-in. thick, Type 304 stainless steel, and bearing rings were fastened to each end of this cylinder to provide the necessary support and rigidity required. The design permitted replacement of the entire core and reflector assembly, replacement of the core only, or replacement of a single fuel element cluster. The design features of the core structure unit are shown in Figures 32 through 35.

2. Reflector Assembly and Control Mechanism

Twelve control drums were arrayed around the reactor core in a circular pattern. Each drum was ~ 4 in. in diameter and $17\text{-}3/8$ in. long. Beryllium reflector material filled two-thirds of the drum cross section, and the remaining one-third was filled with hafnium hydride nuclear poison material. The drum was shrink-fit clad with a nickel sheath, and end caps were welded to the cladding and to the drive shafts which extended axially from the ends of the drums. The drum shafts were flanged, and were fastened to the beryllium sector by bolts and dowels.

To complete the reflector assembly, stationary cusp-shaped nickel reflector segments were located between and outboard of the drums, inside the core structure cylinder. The 12 cusps were secured between the upper and lower bearing rings by fasteners and dowel pins, thus assuring positive clearance between the cusps and the control drums.

A shaft which extended through the shield plug engaged the upper drum shaft through the sleeve-type coupling. This shaft connected the drive mechanism to each of the drums. One control drive mechanism was provided for each drum. The 12 control drum drive mechanisms were mounted on a single base plate in the space provided between the shield plug and the reactor vessel cover. Each control drive mechanism consisted of the following components: a scram spring, a scram clutch, a drive motor with gearing, a drum position sensing device, and their associated power and instrumentation wiring.

The scram spring provided rapid rotation of the control drum to the shutdown position. The spring was a flat coil spring, with the inner end attached to the

ATOMIC INTERNATIONAL <small>A DIVISION OF AMERICAN ARTIFICIAL MATERIALS CO.</small>	
SUPPORT - CORE STRUCTURE	
DRAWN: J.P. BARTON CHECKED: [blank] ENGINEER: [blank]	DATE: 4/8/51 SCALE: 1/2" = 1'-0" SHEET NO. 1 of 1

Figure 32. S4PR Support - Core Structure

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drive shaft and the outer end attached to the control drive mechanism housing. The spring was preloaded to assure sufficient positive torque to move the drum to the least reactive position. An electromagnetic clutch was provided to engage the drive mechanism to the drive shaft. When shutdown was required or loss of power occurred, the holding magnet was de-energized, and the clutch was released. In connection with the scram mechanism, a snubber was provided on the underside of the lower bearing support ring. This snubber, essentially a dashpot, consisted of a vane attached to the drum shaft and a cavity below the lower pivot bearing.

The entire control drive mechanism was located within the reactor pressure vessel, and was immersed in 200° F water at 1200 psia. The electrical power and instrumentation leads were brought out of the pressure vessel through a high-pressure sealed conductor disconnect plug. The design layout of the control mechanism is shown in Figure 36.

3. Reactor Vessel and Shield Plug

The reactor vessel was a vertically arranged, conventional pressure vessel of Type 347 stainless steel, and was of welded construction, consisting of a cylindrical shell, an elliptically dished bottom head, and a bolting flange at the upper end. The vessel was 6-1/2 ft long by 25 in. OD, and was supported at the bottom by a cylindrical skirt which was welded to the vessel body. The lower edge of the skirt was fitted with a flange plate, to provide bearing area and space for anchor bolts. Top closure of the vessel was provided by a bolted, spherically dished, top head. The vessel was designed for an internal pressure of 1400 psia and a temperature of 600° F.

Inside the vessel was fitted a liner which served as both the thermal (neutron absorbing) shield and the reactor core support member. This liner extended from a point near the bottom of the reactor vessel to an elevation just above the coolant nozzle, and was attached around the top by welding to the reactor vessel. Four bosses, welded to the outside of the liner near the bottom, provided for proper spacing and centering of the liner within the vessel, and permitted differential thermal expansion between the liner and the vessel. A ring extended inward near the bottom edge of the liner, to serve as the core structure support. The annular space between the liner and the reactor vessel provided a downcomer passage for reactor inlet coolant.

The vessel was fitted with a single nozzle, located on the side, about two-thirds of the vessel height up from the bottom of the vessel. This nozzle provided both the inlet coolant passage and the outlet steam-water mixture passage. The arrangement was that of two concentric passages, with the inner, circular passage carrying the steam-water mixture, and the outer, annular passage carrying the inlet coolant. The inner pipe penetrated the vessel liner, and was attached by welding. The two concentric pipes were divided into separate pipes just outside the vessel activation shield, for connection into appropriate process piping.

A shield plug was provided inside the vessel liner, above the core, of sufficient thickness to limit activation of structure and equipment to that required for maintenance access in the chamber provided to house the control drive mechanisms. A layer of thermal insulation was provided at the top surface of the shield plug, to thermally isolate the equipment chamber. Steel sleeves penetrated the shield plug, and served as conduits for the reactor control drum activating shafts. At the bottom of the shield plug, a streamlined reentrant steam-water flow passage provided for relatively unimpeded flow of the steam-water mixture from the upper plenum to an eccentric collecting passage.

The top cover for the reactor vessel consisted of a bolting flange, a ring section from a spherically dished plate, and a center conduit port. The port provided for the passage of a conduit for electrical and instrumentation conductors to the control drive mechanisms. The conductors terminated at the upper end of the conduit, in high-pressure, sealed, multipin disconnect plugs. The port cover was underneath the reactor vessel cover, and was bolted to the cover to permit removal of the reactor cover without disturbing the electrical conductors.

The reactor vessel was thermally insulated at the bottom and sides by a layer of fibrous insulating material. The cover plate was not insulated, since its temperature was essentially that of the surrounding shield water.

The reactor vessel layout is shown in Figure 26.

4. Reactor Shield

The design of the reactor shield for operational systems would be highly dependent upon the maintenance and access requirements of the specific application. Any application would have as a minimum requirement that the area

Figure 36. S4PR Installation Assembly Control Mechanism

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surrounding the reactor vessel be accessible with the core removed, and that the surrounding structure and components not become activated to the point that contact maintenance access would be denied. These two criteria were selected as the shielding criteria for the SNAP 4 Prototype System.

The first criterion was met by a layer of lead immediately outside the reactor vessel thermal insulation. The thickness of the lead was proportioned to match the amount of gamma activity in the vessel wall and in other vessel and reactor components. This layer of lead was placed under the support skirt and bearing plate, and extended upward around the reactor vessel to the bolting flange.

The second criterion was met by the water contained within the aluminum shield water and containment vessel. The shield water vessel was an 8-ft diameter cylinder with two flanged and dished pressure vessel heads. The shield water provided a means of quenching any steam released as a result of reactor vessel or piping system failure, and the thickness of the shield water vessel was designed to contain the resultant pressure increase from such a failure.

The combined lead and water shield permitted access by maintenance and operating personnel to the area outside the shield water vessel, with the reactor shut down. This area was inaccessible when the reactor was operating, and concrete biological shielding was provided in the facility to bring dose rates in operating areas to permissible levels.

The reactor vessel and lead shield were located and secured within the shield water vessel by an arrangement of two concentric cylinders which extended upward from the bottom of the shield water vessel. The inner cylinder mated with the reactor vessel support skirt bearing plate, and the outer cylinder mated at the mean diameter of the lead shielding.

The arrangement of shielding is shown in Figures 26 and 27. This arrangement provided the simplest possible shielding fabrication, yet allowed the conducting of shielding experiments with the operating SNAP 4 reactor, to obtain design information for subsequent operational shields.

D. PROCESS SYSTEM

The process system for the basic SNAP 4 Prototype System was composed of all the fluid (liquid and gas) subsystems which were required to support the safe operation of the reactor heat source. The primary functions of the process system were to: (1) dissipate the power generated by the reactor, and (2) control the process fluid chemistry. Figure 37 is a preliminary piping diagram and heat balance for the system. The entire process system was hermetically sealed to prevent contamination of the environment and eliminate the need for external gas and liquid waste handling and water makeup equipment. The subsystems which made up the process system are as follows:

- 1) Recirculation Subsystem: Steam separator and recirculation pump
- 2) Steam Subsystem: Condenser, auxiliary (pump) turbines, steam flow control station, steam surge volumes, and steam dumping line
- 3) Feedwater and Condensate Subsystem: Feedwater and condensate pump, first and second regenerative heat exchangers, nonregenerative heat exchangers, filter, and deionizer
- 4) Recombiner Subsystem: Recombiner, eductor, and mixing chamber
- 5) Valve Operating Subsystem
- 6) Pressure Relief Subsystem
- 7) Poison Injection Subsystem
- 8) Reactor Decay Heat Removal Subsystem.

1. Steam Separator

At design thermal power of 12 Mw, the reactor coolant would leave the reactor at a flow rate of 410,000 lb/hr, in the form of a steam-water mixture consisting of 10 wt % steam and 90 wt % water. A steam separator was provided to separate this mixture and provide dry saturated steam to the power and auxiliary turbines, and to return the high-pressure water to the reactor. The requirement to remove at least 99% of the vapor from the liquid was as necessary as was the requirement to remove at least 99% of the liquid from the vapor, to prevent cavitation-erosion of the recirculation pump impeller. Two stages of

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Figure 38. Steam Separator (Conventional), Plan View

separation were required to meet these requirements, and could be a combination of gravity-mechanical or mechanical-mechanical. A conventionally designed, commercially available unit of the type, shown in Figures 38 and 39, would meet the performance requirements for this unit.

2. Recirculation Pump

The recirculation pump provided the means by which the high-pressure water was removed from the steam separator and returned to the reactor. The condensate being returned to the reactor from the condensate and feedwater pump joined the recirculation stream ahead of the recirculation pump to provide subcooling and prevent cavitation; thus, the recirculation pump would handle the entire reactor flow rate of 410,000 lb/hr. The head developed by the recirculation pump would equal the pressure drop in the recirculation loop, or 45 psi at full flow.

The pump would be a single-stage, axial-flow type impeller operating at 9800 rpm. The impeller diameter would be $\sim 4\text{-}1/2$ in. The driving unit would be a partial admission, single-stage, steam turbine of the Terry type or conventional axial flow velocity compounded. The turbine wheel diameter would be ~ 6 in. in diameter. Steam flow rate to the turbine would be ~ 2600 lb/hr.

The combined turbine and pump would be mounted on water-lubricated bearings, and the complete unit would be hermetically sealed. The overall envelope for this turbine-driven recirculation pump would be 8 by 8 by 14 in., and the unit would weigh ~ 200 lb.

3. Condenser

The condenser would receive 41,000 lb/hr of steam from the reactor at full power. The actual flow rate to the condenser would be controlled by a steam flow control valve, located between the steam separator and the condenser. The function of the condenser was to condense the steam to a saturated liquid, thereby dissipating the reactor heat. The condensed steam would be slightly subcooled, 7 to 10° F, and collected in the condenser hotwell. The hotwell served as a storage and surge volume for the condensate and feedwater system.

The condenser would be designed to condense the specified amount of steam, at condensing pressures varying between 1 and 25 psia. The envelope for the

Figure 39. Steam Separator (Conventional)

condenser would be ~4 ft in diameter by 12 ft overall length, and would be designed and fabricated in accordance with conventional nuclear practices. The tube bank would consist of ~1650 5/8-in. OD stainless steel tubes, arranged in a U-tube configuration. The unit would be of all-welded construction. Steam would be condensed on the outside of the tubes, and an inlet transition section and baffles would be provided to reduce steam entrance velocity, to prevent excessive tube erosion. Provision would be made for removal of noncondensable gases during operation, to prevent gas blanketing of the heat-transfer surfaces.

In addition to the normal steam inlet to the condenser from the steam flow control station, all high-pressure relief systems would discharge to the condenser, and the condenser was designed to condense the resulting vapor and contain the associated minor pressure surge.

4. Feedwater and Condensate Pump

The condensate and feedwater pumps were mounted on a common shaft, and driven by a single turbine driver. The condensate stage was a 2-in. diameter centrifugal flow impeller which would handle 46,000 lb/hr of condensate, and increase the pressure from that of the condenser to 125 psia. After leaving the condensate pump stage, the condensate passed through the water purification equipment, and returned to the inlet of the feedwater pump. A small quantity, (~2500 lb/hr), of the condensate pump discharge was recirculated to an eductor which removed noncondensibles from the main condenser, and the fluid then returned to the condensate pump section. Additional condensate recirculation might have to be provided to operate a jet-boost pump to suppress cavitation, or a mechanical inducer could be provided.

The feedwater pump consisted of four equal stages of 3-in. diameter centrifugal flow impellers, and operated at 18,000 rpm. The feedwater pump received the condensate stream after it had passed through the water purification equipment, and raised the pressure to the 1220 psia required to inject the fluid into the recirculation stream.

The turbine driver was mounted between the two pumps, and was the same turbine wheel as used on the recirculation pump. Although still operating as a partial admission turbine, the feedwater and condensate pump turbine had four inlet nozzles, as contrasted with a single nozzle for the recirculation pump turbine.

The turbine and pumps were mounted on water-lubricated bearings, and the complete unit was hermetically sealed. The overall envelope for this turbine driven feedwater and condensate pump was 8 by 8 by 24 in. , and the unit weighed ~250 lb.

5. Regenerative and Nonregenerative Heat Exchangers

The water purification subsystem required one nonregenerative and two regenerative heat exchangers to reduce the condensate temperature to one that would not induce resin damage in the demineralizer, and at the same time to minimize the amount of waste heat which had to be rejected from the cycle.

The arrangement of these heat exchangers in the flow cycle was such that condensate from the condenser hotwell passed through the tube side of the first regenerative exchanger, where its temperature was dropped to 140° F, and then entered the suction of the condensate pump. A small side stream (~2000 lb/hr) was drawn from the recirculation loop and passed through the tube side of the second regenerative exchanger, where its temperature was reduced from 568 to 240° F. After leaving the second regenerative exchanger, the pressure of the fluid was reduced, and the stream mixed with the condensate pump discharge before the combined stream passed through the tube side of the non-regenerative heat exchanger. The temperature of the fluid was reduced to 130° F in the nonregenerative exchanger, and the fluid then passed through the demineralizer and guard filter, the shell side of the first and second regenerative exchangers, and entered the feedwater pump suction at 240° F.

The three heat exchangers had many common design features, and are therefore discussed together. These units were of relatively conventional nuclear design and construction. They were fabricated of welded, stainless steel, and utilized 3/8-in. or 1/2-in. tubing in a baffled, counterflow, U-tube configuration. The design pressure for the shell and tube sides of the first regenerative exchanger, the shell and tube sides at the nonregenerative exchanger, and the shell side of the second regenerative exchanger was 150 psia. The design pressure for the tube side of the second regenerative exchanger was 1500 psia. The overall envelopes and weights of these units were as follows:

- 1) First regenerative: 3 ft OD by 10 ft (1850 lb)
- 2) Second regenerative: 6 in. OD by 6 ft (300 lb)
- 3) Nonregenerative: 12 in. OD by 10 ft (500 lb).

In addition to these heat exchangers, which were associated with the water purification subsystem, there was an additional regenerative-type heat exchanger which condensed the pump turbine exhaust steam, and returned the heat energy and fluid to the process stream. This exchanger was of the "Helicoil" type, which is a compact shell and tube exchanger in which the tubing is arranged in a coil between header pipes, rather than the conventional straight or U-tube configuration. The unit was fabricated of stainless steel, and was of an all-welded construction.

6. Deionizer

To prevent fouling of heat-transfer surfaces, particularly in the reactor, was established that the process water had to be maintained within the following limits:

Oxygen	0.4 ppm (max)
pH	7.0 ± 2
Corrosion Product Concentration	10 ppb (max)
Conductivity	10 M Ω -cm
Additives	None

To maintain the required pH, corrosion product level, and conductivity, a mixed-bed deionizing unit of nuclear grade, strongly acid cation resin in the hydrogen form, and a nuclear grade, strongly basic anion resin in the hydroxyl was used. Approximately 7 ft³ of resin were required to provide the exchange capacity to remove the corrosion products generated in the system in 1 year of operation.

7. Recombiner

An estimated 4.5 lb per day of hydrogen and oxygen would be produced in the system, due to radiolytic decomposition of the reactor coolant. To eliminate the need for off-gas handling and a makeup feed system, a catalytic

recombiner was provided, which would continuously recombine the hydrogen and oxygen, and reinject the resultant water and excess hydrogen back into the system.

The catalytic recombiner consisted of a cylindrical chamber, ~3 in. OD by 12 in. overall length, filled with platinum-coated alumina pellets. The pellets were held in place by screens at each end of the chamber. Since the catalyst will not function when wet, a small cartridge-type electrical heater was provided in the catalyst bed near the inlet.

8. Valve Operating Subsystem

The valve operating subsystem provided the means by which process system valves were positioned, in response to control and system flow signals. A typical process valve control and operating loop consisted of a controller which electrically compared the desired flow to the measured flow, and generated an appropriate error or null signal which controlled the position of a piston in a hydraulic servo valve. A supply of 800 psia water, taken from the high-pressure feedwater pump, served as the actuating fluid. The position of the servo valve determined to which side of the process valve actuating piston the actuating fluid would flow. The servo valve also allowed water from the opposite side of the actuating piston to be bled to the condenser hotwell, as it was displaced by actuating piston motion. The flow capacity of each servo valve and valve actuator was determined by the size and response characteristics of the valve to be controlled.

An emergency and/or startup mode of operation of the valve actuating system was provided, by the inclusion of a water reservoir which could be pressurized by means of a high-pressure gas accumulator (bottle). This system had sufficient capacity to allow initial valve operation for startup to put the recirculation loop in a condition for self-pressurization.

9. Pressure Relief Subsystem

A pressure relief system was provided, to insure that the system pressure could not exceed the design pressure, and thereby create a potentially hazardous situation. The pressure relief system consisted of two, main, self-actuating, bellows-sealed, adjustable setpoint and blowdown control valves.

One of these valves was installed in parallel with the main steam flow control station, and would be actuated if steam pressure rose to 1300 psia. The valve capacity would be equal to 120% of steam system design flow, and would discharge directly to the main condenser. The second valve was connected to the steam separator, and would actuate if separator pressure rose to the design pressure of 1400 psia. This valve would also be designed to handle a flow rate which corresponded to the maximum possible system transient flow rate, and would also discharge to the main condenser. Actuation of the steam separator pressure relief valve would also provide a power level scram input signal to the reactor control system.

The pressure relief valves would be constructed of stainless steel, and the valve seats and discs would be hard-surfaced to prevent wiredrawing or damage due to "valve chatter."

10. Poison Injection Subsystem

Due to the experimental-developmental nature of the S4PR prototype system, a third safety was provided, in the form of a nuclear poison injection system. This system injected a concentrated borate solution into the reactor inlet, if other means of reactor shutdown failed or malfunctioned.

Borated water was contained in a pressure vessel, and was maintained at a pressure of 1300 psia by means of a hydrogen gas overpressure. The hydrogen gas pressure was supplied from a bottled supply of gas, and pressure was controlled in the borated water tank by a pressure-regulating valve. Two shutoff-isolation valves were installed in the piping between the water tank and the reactor inlet. A drain line was provided between these two valves to prevent leakage of the poison solution into the reactor inlet. When the system was actuated, the drain line isolation valves were closed, and the shutoff-isolation valves opened, allowing the gas overpressure to inject the solution into the reactor. A check valve was provided in the water tank outlet to prevent reverse flow as the tank was emptied.

The poison injection system was recharged by venting the pressure, filling with additional borate solution, and then re-pressurizing with the hydrogen cover gas. The poison injection system was designed for a pressure of 1750

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Figure 40. Schematic Diagram of S4PR Control System

psia, and a pressure relief valve was provided to prevent the pressure from exceeding 1500 psia.

11. Decay Heat Removal Subsystem

A simple decay heat removal system was located within the water shield, and utilized the shield water as a heat sink. The heat exchanger consisted of a length of finned tubing which was connected between the inlet and outlet reactor piping. A check valve on the reactor inlet piping side of the exchanger prevented flow through the decay heat exchanger during normal system operation, but opened and permitted flow through the decay heat exchanger whenever reactor inlet pressure was equal to the outlet pressure. The arrangement of the decay heat exchanger system is also shown in Figure 27.

E. CONTROL AND INSTRUMENTATION SYSTEM

The complexity of the control and instrumentation components and subsystem was held to a level which was dictated only by the safety and operational needs of the reactor-steam system. This simplicity lent itself advantageously to eventual programmed plant startup and shutdown, and to adaptation of a fully automatic, self-diagnostic control system.

1. Control System

The S4PR control system was a conventional experimental type of system and required attendance during system operation. Manual operations were required for startup and shutdown, setting out power level, diagnosing system behavior, and correcting operational irregularities. Automatic control was provided for normal operation about the desired operational power level. Automatic scram and setback were provided as well as operator manual override of these automatic functions.

The plant control system consisted of a number of subsystem or single control loops, some of which were mutually coupled. Each loop operated by comparing a reference signal to a signal derived from the controlled variable. The differences between the measured and reference signals generated an error signal which actuated a controller to reduce the error to zero. In some loops, rate of change of the error signal was detected, and provided as an additional control signal to enhance stability. The control system, shown schematically in Figure 40, consisted of the following subsystems:

Figure 41. Block Diagram of S4PR Nuclear Instrumentation System

- a) Neutron Flux Control
- b) Pressure Control
- c) Recirculation Flow Control
- d) Feedwater Flow Control
- e) Steam Separator Liquid Level Control
- f) Condenser Heat Removal and Pressure Control.

2. Nuclear Instrumentation System

The nuclear instrumentation system (Figure 41) provided reactor neutron flux level and period information to the plant protective system, the plant control system, and the control room. This system monitored reactor flux through seven detector channels which covered at least 10 decades of power in three overlapping ranges. The startup (source) range detection system consisted of two channels employing fission chambers; the intermediate range detection system consisted of two channels employing compensated ion chambers; and the power range detection system consisted of three channels employing uncompensated ion chambers.

The nuclear signal conditioning and power supply equipment for these detectors were standard, commercially available equipment, using solid-state components in a "pull-out" modular type of construction. The startup range signal conditioning equipment contained a linear pulse amplifier with two outputs, one for readout instrumentation and the other for the plant control and plant protective systems. The intermediate and power range signal conditioning amplifier outputs connected to the plant control and plant protective systems.

Recording instrumentation consisted of the following:

- 1) Log countrate recorder, to record either startup channel
- 2) Log N recorder, to record either intermediate range channel
- 3) Power level recorder, to record any one of the three power range channels.

Figure 42. Block Diagram of S4PR Plant Protective System

3. Plant Protective System

The plant protective system protected the reactor and system from damage, or functioned to minimize the extent of damage, during all phases of plant operation. The plant protective system, illustrated in Figure 42, consisted of startup interlocks, alarms, automatic reactor setbacks or scrams, activation of the decay heat removal loop, and poison injection into the core. The plant protective system was completely separate from the normal plant control and instrumentation systems, except for their common use of nuclear detectors, amplifiers, and control drum actuators.

Reactor scram was activated by any of the following:

- 1) Excessive period (reactor)
- 2) Excessive power level (reactor)
- 3) High core temperature
- 4) High reactor outlet steam pressure
- 5) Low reactor inlet pressure
- 6) Low separator water level
- 7) Low flow to condenser
- 8) Low reactor shield water level
- 9) Low reactor inlet flow
- 10) Manual.

4. Radiation Monitoring System

The radiation monitoring system, shown in Figure 43, was provided to protect plant personnel and equipment from excessive radiation, and to provide information required for the health physics services.

Standard dosimeters, film badges, sealers, hand and foot counters, and portable survey meters were utilized. In addition, the following installed systems were provided.

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Figure 43. Block Diagram of S4PR Radiation Monitoring System

- 1) Area Radiation Monitors at ten locations throughout the test facility, with a common power supply and recorder located in the control room.
- 2) Stack Monitor – Gaseous effluent from the system operating area was drawn off the main exhaust, and monitored in a sampling system.
- 3) Process Fluid Monitoring – Samples of reactor coolant, and water entering and leaving the purification equipment, were monitored for fission product activity level.

5. Process Control Monitoring and Diagnostic Instrumentation Components

One standard type of variable-reluctance type of pressure transducer was selected for all pressure transmission, including differential pressure head, level, and flow measurements. Flow-tube type elements were used for all flow measurements, except for extremely low flows. Extremely low flows were measured by means of precision small-bore orifice elements. The differential pressure produced by the flow tubes or orifices were sensed and transmitted by the differential, variable-reluctance pressure transducers.

All temperature sensing elements were stainless steel sheathed, MgO insulated, Chromel-constantan thermocouples. Thermocouples for the measurement of fluid temperatures were protected by immersion wells, constructed of the same material as the piping or vessel into which it was installed.

Figure 44. 2-kwe System Technology Comparison

● MAXIMUM COOLANT TEMPERATURE (°F)	1050	1200
● HEIGHT (ft)	21.5	14.5
● BASE DIAMETER (ft)	8	4.5
● RADIATOR AREA (ft)	250	100
● WEIGHT (lb)	1950	1100

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Figure 45. 5-kwe System Program Schedule

VI. 5-kwe REACTOR THERMOELECTRIC SYSTEM PROGRAM

A. INTRODUCTION

Since the start of the U. S. space program, power and lifetime requirements have modestly, but steadily, grown. A decade ago, power levels of several watts, for durations of 1 or 2 years, readily met most mission requirements. However, some DOD and NASA missions in the late 1970's and early 1980's show projected power requirements to 10 kwe, and durations to 5 years. In addition, significant potential mission advantages were identified with the use of reactor power systems for some of these missions.

Although the SNAP 10A System, flown in 1965, successfully demonstrated all aspects of reactor system operation in space, substantial technology improvements in components and systems have occurred since then. Figure 44 illustrates the system gains projected through the use of existing component technology. A 2-kwe system, designed using SNAP 10A System (1965) technology, would weigh nearly a ton, and require 250 ft² of radiator. Using the component technology of the 1970's, the same system would weight 1100 lb, and would require only 100 ft² of radiator area. The gains primarily result from the use of higher-efficiency PbTe thermoelectric material, rather than SiGe, and through an increased reactor temperature capability.

The purpose of the 5-kwe Reactor Thermoelectric System Program was to provide potential users with a reactor power system option for future missions, and to demonstrate, on a system level, the gains realized in component performance since SNAP 10A. The system was designed to produce 5 kwe for 5 years. The system configuration was consistent with unmanned spacecraft operations and configuration. Shadow shielding was used to reduce radiation levels to 10¹² nvt and 10⁶ rads at the power system - spacecraft mating plane. Although the demonstration was to be a ground test, the system was designed to meet all requirements for a space system. The system was designed to be consistent with all phases of prelaunch operations, to withstand the launch environment of the Titan launch vehicle family, to start up automatically in space, and to operate at full power for 5 years in the vacuum, thermal, and meteoroid environment of space.

The program was funded by the joint AEC-NASA Division of Space Nuclear Systems. Figure 45, shows the original program schedule. The conceptual design was started in February 1972, and completed in October 1973, at which time the Preliminary Design Phase was initiated. On January 8, 1973, direction was received to close out all activities related to the 5-kwe System Program.

The purpose of this report is to summarize the design and potential performance of the 5-kwe System, as it existed at the time of program close-out.

B. DESIGN AND PERFORMANCE SUMMARY

This section presents system design requirements and reference system characteristics.

1. System Requirements and Constraints

In addition to the overall system requirements outlined in the preceding section, several other requirements and constraints were placed on the system.

Because a specific application was not defined, specific mission-related system-weight and size limitations were not established. However, general dimensional limits were established, considering existing Titan launch shroud sizes and typical spacecraft configurations. In addition, the system was configured to be consistent with ground test facility limitations. The results of past application studies were used to establish approximate weight limits with the Titan III-B - Burner II launch vehicle.

A reactor outlet temperature limit of 1200°F was selected, not to identify the upper limit of zirconium hydride (ZrH) reactor temperature potential, but to remain well within current state-of-the-art technology.

Component designs were to be tailored specifically to the 5-kwe System; however, the same technology was to be applicable to power systems over a 1- to 10-kwe range.

2. Reference System Characteristics

This section describes system configuration, performance, design margins, and weights.

a. System Configuration and Performance

Figure 46 shows component arrangement schematically, and Figure 47 is a layout drawing of the 5-kwe system. The system had a primary and a heat-rejection NaK loop. The primary-loop NaK flowed into the top of the core vessel through two inlet lines, was heated in the core, and flowed out through two outlet lines. The NaK then flowed through the PbTe thermoelectric modules, where it transferred its heat to the hot cladding of the modules. The fluid was then pumped, by the electromagnetic pump, back to the reactor core for reheating. The heat-rejection loop NaK picked up the waste heat at the cold cladding of the tubular modules, and was then cooled as it flowed through the tubes of the conical - cylindrical radiator. Both loops were pumped by a single dc conduction pump assembly powered by three pump tubular modules. The primary loop required two volume accumulators, and the heat rejection loop required one. All three were identical design.

Expansion joints were used in the system to permit thermal expansion and contraction of subassemblies. The 16 power tubular modules were arranged in a basket within the radiator. The radiator cooled the heat-rejection loop NaK, and also acted as a primary structural support.

The overall height of the system was 23 ft, and it had a maximum diameter of 6 ft. The radiator area was 267 ft².

Table 13 is a summary of beginning-of-life (BOL) and end-of-life (EOL) performance. The system was designed to produce 5-kwe for 5 years, at a maximum reactor outlet temperature of 1200°F. Because the tubular modules degraded with time at temperature, the maximum reactor outlet temperature occurred at EOL. Consequently, prior to EOL, the temperature was 1200°F. At BOL, the temperature was 1148°F, and gradually increased as degradation occurred.

Figure 48 shows the 5-year variation of reactor outlet temperature, reactor thermal power, and average radiator temperature for a constant electrical power of 5 kwe. Figure 49 illustrates power module thermal input, degradation rate (in power), and total degradation (in power) as a function of time. The relationship between power module degradation and system performance can be seen by comparing these figures.

6532-40103

Figure 46. 5-kwe System Schematic

Figure 47. 5-kwe System Design Layout (Dwg R-N652630001) AI-AEC-13067

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TABLE 13
5-kwe SYSTEM BOL-EOL PERFORMANCE SUMMARY
(Sheet 1 of 2)

Parameters	BOL	EOL
Reactor		
Power (kwt)	95.6	101.3
Outlet Temperature (° F)	1148	1200
Core temperature rise (° F)	88.7	94.8
Power Modules		
Average hot clad temperature (° F)	1091	1140
Average cold clad temperature (° F)	527	543
Thermal input (kwt)	80.36	84.79
Overall efficiency (%)	6.38	6.05
Power degradation (%)	-	6.67
Efficiency degradation (%)	-	4.70
Output power (kwe)	5.13	5.13
Power bus bar losses (I^2R) (kwe)	0.13	0.13
Power at mating plane (kwe)	5.0	5.0
Voltage at mating plane (vdc)	30.0	30.0
Radiator		
Thermal input (kwt)	89.3	94.9
Inlet temperature (° F)	589	611
Temperature drop (° F)	132	144
Average temperature (° F)	522	538
AI-93 emissivity	0.90	0.90
AI-93 absorptivity	0.50	0.50
Fraction bonded fin/tube area (%)	80	72
Direct solar flux (Btu/ft ² -hr)	141.1	141.1
Earth reflected flux (Btu/ft ² -hr)	23.5	23.5
Direct earth flux (Btu/ft ² -hr)	20.17	20.17
NaK Pump		
Primary loop flowrate (lb/sec)	4.87	4.83
Primary loop developed head (psi)	1.11	1.09
Secondary loop flowrate (lb/sec)	3.01	2.94

TABLE 13
5-kwe SYSTEM BOL-EOL PERFORMANCE SUMMARY
(Sheet 2 of 2)

Parameters	BOL	EOL
Secondary loop developed head (psi)	1.57	1.50
Current input (amp)	1728	1691
Voltage drop (vdc)	0.210	0.209
Primary throat magnet effective field (gauss)	2505	2380
Secondary throat magnet effective field (gauss)	2000	1900
Pump Modules		
Average hot cladding temperature (°)	1079	1128
Average cold cladding temperature (°F)	531	547
Thermal input (kwt)	9.77	10.39
Overall efficiency (%)	3.71	3.41
Power degradation (%)	-	8.13
Efficiency degradation (%)	-	6.75
Output power (kwe)	0.362	0.354
Piping System		
Primary loop flowrate (lb/sec)	4.87	4.83
Primary loop pressure drop (psi)	0.97	0.95
Secondary loop flowrate (lb/sec)	3.01	2.94
Secondary loop pressure drop (psi)	1.37	1.30
Thermal Losses		
Primary loop thermal losses to space (kwt)	1.27	1.43
Primary loop radiative shunt losses to secondary loop (kwt)	3.24	3.71
Primary loop conductive losses to secondary loop through pump bus bar (kwt)	0.97	1.02
System		
Carnot efficiency (%)	36.2	37.2
Overall efficiency (%)	5.23	4.93

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Figure 48. 5-kwe System Performance
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6532-40105

Figure 49. 5-kwe System Power Module Performance

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The 5-kwe Reactor Thermoelectric System was to be tested in the Building 059 facility at the Santa Susana Field Laboratory of Atomics International. The system was to be operated in an inverted position (reactor down), with a cold wall provided for thermal rejection from the radiator. Detailed analyses were performed to determine the capability of the radiator heat sink, utilizing various temperatures and emissive coatings. It was found that an average cold-wall temperature of 100°F, and an average emissivity of 0.85, provided an environment very similar to space. Although the final design of the facility was incomplete, preliminary trade studies indicated that utilization of iron titanate coating (emissivity of ~ 0.90) on most internal sections of the cold wall could easily provide the required average emissivity. The available cooling towers are capable of maintaining the necessary 100°F cold wall temperature.

Since the ground and space environments are essentially equivalent, the detailed performance of the 5-kwe System remains identical in both situations (although minor second-order variations in the thermal rejection temperature dependence may occur).

b. Design Margin Performance

This section describes the margins applied to each component, and their effect on system performance.

Trade studies were performed to analyze those components which should exhibit performance design margins. Four components were selected: (1) the reactor, (2) the EM NaK pump, (3) the radiator, and (4) the volume accumulator units. The power and pump modules were not considered in this analysis, because they were developmental items whose performance was specified by system requirements.

The ZrH reactor incorporated margins in three areas, since it was designed to operate continuously for 5 years at 110 kwt, 1200°F outlet temperature, and 100°F core temperature rise. Comparison of these design requirements with the operating requirements of the system indicated the following design margins: (1) 12% on reactor power (average value, 98.5 kwt), (2) 26°F on outlet temperature (average value, 1174°F), and (3) 8°F on core temperature rise (average value, 92°F). These design margins were noninteracting with the system

performance statepoints, because the reactor was simply the heat source for the power conversion system, and all thermal requirements were within the design values for the reactor. The system envelope was affected by this margin, however, in that the reactor envelope dimensions were increased to allow for the overpower design requirements.

The electromagnetic NaK pump was designed with pressure-head capability greater than that required by the piping systems (at a given flowrate). All piping system pressure-drop calculations were done on a worst-case component performance basis. In this way, the pressure-head design margin was not utilized to neutralize unrealistic pressure-drop predictions. Each pump throat exhibited 15% greater head than was necessary to drive its respective piping loop. These design margins interacted with the system performance statepoints, because utilization of the additional available pressure head resulted in: (1) increased flow rates in both loops, (2) increased pressure drop in both loops, and (3) lower temperature differentials in both loops. Thus, analysis of a system which included the pump performance capabilities indicated a more benign operating statepoint.

The radiator design margin was incorporated in three areas: (1) utilization of a lower-than-measured emissivity for the AI-93 coating, (2) utilization of a higher-than-measured average solar absorptivity for the AI-93 coating, and (3) utilization of environmental thermal fluxes associated with a worst-case orbit (maximum sun). The AI-93 coating exhibited an emissivity of 0.92, while the radiator was designed for 0.90. Thus, the design margin was $\sim 2\%$ (a value equivalent to ~ 2 kw rejection capability, because emissivity appeared directly in the radiation calculation). The solar absorptivity of AI-93 increased during radiator life, being ~ 0.30 at BOL and 0.50 at 5 years (average value, 0.40). The radiator was designed for an absorptivity of 0.50 (continuous, 5-year), and thus exhibited a 67% margin at BOL and no margin at EOL. The expected environmental thermal flux associated with a "nominal" orbit was $64.4 \text{ Btu/ft}^2\text{-hr}$ (direct solar + direct earth + earth reflected). The radiator was designed to accept a flux of $\sim 101 \text{ Btu/ft}^2\text{-hr}$. This was equivalent to a 57% design margin on environmental input. The combined effects of these margins could be converted to a 6% margin in thermal rejection capability. This margin interacted with the system performance statepoints, because utilization of the additional

available performance capability would result in a reduced average radiator temperature. All system temperatures were subsequently reduced, thus reducing thermoelectric module degradation rates and increasing reactor life-time capabilities.

The volume accumulator units (VAU) were designed to compensate for one-half of the expansion volume of the primary loop at the following temperature condition: EOL average loop temperature plus 50°F, to accommodate scram setpoint plus 50°F design margin. Thus, each VAU was designed for 191 in.³ of NaK displacement. The expansion volumes associated with the average EOL operating temperatures of the system were 341 in.³ (primary) and 111 in.³ (secondary). Thus, the two units in the primary loop exhibited an EOL design margin of 41 in.³ (12%), while the single secondary unit exhibited a design margin of 80 in.³ (72%). This component design margin did not interact with the system performance statepoints, because it provided the sink for coolant expansion only. However, the VAU size increase associated with the margin assessment was incorporated in the internal power conversion system equipment placement.

The additional pumping and heat rejection capabilities of the system could be incorporated into the performance analyses, with the generation of specific statepoints. Table 14 details the BOL and EOL (5-year) performance parameters for the 5-kwe System, operated with the design margin capabilities included. Comparison with Table 13 indicates the substantial reductions in component operating requirements.

Note that the reactor outlet temperature at 5 years is well below the 1200°F operating limit. This system was capable of delivering 5.0 kwe over a period significantly longer than the 5-year design life-time. This period was limited by: (1) the reactor outlet temperature (1200°F), (2) the available control reactivity. Detailed analyses have shown that the system was reactivity limited, rather than temperature limited, as illustrated in Figure 48. No additional control steps were available to the reactor after 7.2 years, while the outlet temperature always remained below 1200°F. It should be noted, however, that the system was capable of operating at continuously decreasing power levels (<5 kwe) while the reactor power and temperature drifted down (based on the temperature coefficients). This is indicated schematically by the dotted line in Figure 50.

TABLE 14
UTILIZATION OF COMPONENT DESIGN MARGIN
CAPABILITIES FOR 5-kwe SYSTEM
(Sheet 1 of 2)

Parameters	BOL	EOL (5 years)
Reactor		
Power (kwt)	93.8	97.7
Outlet temperature (° F)	1118	1159
Core temperature rise (° F)	84.4	88.3
Power Modules		
Average hot clad temperature (° F)	1063	1102
Average cold clad temperature (° F)	506	523
Thermal input (kwt)	79.00	81.99
Overall efficiency (%)	6.49	6.26
Power degradation (%)	-	4.64
Efficiency degradation (%)	-	3.25
Output power (kwe)	5.13	5.13
Power bus bar losses (I^2R) (kwe)	0.13	0.13
Power at mating plane (kwe)	5.00	5.00
Voltage at mating plane (vdc)	30.0	30.0
Radiator		
Thermal input (kwt)	87.6	91.4
Inlet temperature (° F)	561	584
Temperature drop (° F)	123	131
Average temperature (° F)	501	518
Al-93 emissivity	0.92	0.92
Al-93 absorptivity	0.30	0.50
Fraction bonded fin/tube area (%)	80	72
Direct solar flux (Btu/ft ² -hr)	105.4	105.4
Earth reflected flux (Btu/ft ² -hr)	16.5	16.5
Direct earth flux (Btu/ft ² -hr)	17.02	17.02
NaK Pump		
Primary loop flowrate (lb/sec)	5.02	5.00
Primary loop developed head (psi)	1.03	1.02

TABLE 14
UTILIZATION OF COMPONENT DESIGN MARGIN
CAPABILITIES FOR 5-kwe SYSTEM
(Sheet 2 of 2)

Parameters	BOL	EOL (5 years)
Secondary loop flowrate (lb/sec)	3.16	3.11
Secondary loop developed head (psi)	1.51	1.45
Current input (amp)	1738	1708
Voltage drop (vdc)	0.209	0.209
Primary throat magnet effective field (gauss)	2505	2380
Secondary throat magnet effective field (gauss)	2000	1900
Pump Modules		
Average hot cladding temperature (°F)	1051	1090
Average cold cladding temperature (°F)	510	527
Thermal input (kwt)	9.60	10.01
Overall efficiency (%)	3.79	3.57
Power degradation (%)	-	5.63
Efficiency degradation (%)	-	4.67
Output power (kwe)	0.364	0.357
Piping System		
Primary loop flowrate (lb/sec)	5.02	5.00
Primary loop pressure drop (psi)	1.03	1.02
Secondary loop flowrate (lb/sec)	3.16	3.11
Secondary loop pressure drop (psi)	1.51	1.45
Thermal Losses		
Primary loop thermal losses to space (kwt)	1.19	1.31
Primary loop radiative shunt losses to secondary loop (kwt)	3.04	3.37
Primary loop conductive losses to secondary loop through pump bus bar (kwt)	0.97	1.00
System		
Carnot efficiency (%)	36.4	36.9
Overall efficiency (%)	5.33	5.12

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Figure 50. 5-kwe System Performance, Utilizing Component Design Margins

c. System Weight Summary

Table 15 summarizes component, subassembly and system weights. The total system weight is 1822 pounds.

d. System Performance During the 5-Year Mission Lifetime

This section presents a brief discussion of the 5-kwe System performance over the 5-year operating lifetime. This discussion is included to clarify the difference between beginning-of-life (BOL) and end-of-life (EOL) design requirements. The degradation phenomena associated with the system are tabulated, to illustrate the conditions that exist which cause a change in the design statepoint over life. An attempt is made to explain the intrinsic coupling between component design margins and system performance.

As was previously discussed, the system was designed to provide a constant 5-kwe output throughout life. However, degradation occurred in the operating characteristics of several components utilized in the 5-kwe System. Because these components exhibited "better" performance at beginning of life, the BOL reactor outlet temperature was well below the operating limit of 1200°F. As degradation occurred during the system life, the outlet temperature was increased to maintain electrical power output. Certain degradation rates were strongly temperature dependent (exponential behavior). This fact complicates prediction of BOL to EOL performance. Table 16 summarizes those component parameters which exhibit degradation.

It was thus evident that the EOL operating statepoint was a function of the operating history of the system. In that the system was constrained to operate at 1200°F reactor outlet temperature at EOL (providing 5 kwe at 30 v dc), undegraded system designs (i. e., BOL) had to be traced over an assumed operating history. System designs had to be rigorously analyzed to couple the component degradation to the system performance, thus arriving at the proper EOL operating statepoint.

TABLE 15
5-kwe SYSTEM WEIGHT SUMMARY

Component	Weight (lb)
Reactor	
Core	290
Reactor vessel	38
Piping (to shield)	9
Reflector-control assembly	160
Total	497
Shielding	
Borated stainless steel gamma-ray shield	98
LiH neutron shield and honeycomb	195
Neutron shield casing	59
Total	352
Power Modules (16)	
Total	192
EM NaK Pump	
Pump throats, magnets, and structure	55
Pump modules (3)	24
Total	79
Piping loops	
Primary piping (wet)	62
Primary loop expansion joints (4)	12
Primary loop foil insulation	5
Secondary piping (wet)	70
Total	149
Volume Accumulators	
Primary units (2)	28
Secondary unit	14
Total	42
Radiator	
Fins and armor	179
Tubes and NaK	124
Rings and stringers	57
Pump support flange	9
Access panels	15
Rivets	37
Al-93 surface coating	16
Support skirt	6
Thermal baseplane baffle	7
Miscellaneous bracketry	10
Total	460
Electrical Wiring	
Power line	11
Auxiliary pump startup bus bar	10
Actuator wiring	9
Instrumentation and wiring	21
Total	51
System Total	1822

TABLE 16
5-kwe SYSTEM COMPONENT PARAMETERS WHICH
EXHIBITED DEGRADATION

Parameters	Strongly Temperature Dependent	Temperature Insensitive
Power module efficiency	XX	
Power module power output	XX	
Pump module efficiency	XX	
Pump module power output	XX	
NaK pump magnetic field strengths		XX
NaK pump bus bar braze joints		XX
Radiator coating solar absorptivity		XX
Radiator fin-tube bond thermal resistance		XX

C. REACTOR

1. Reactor Description

a. Reactor Configuration

The reactor designed for integration into the 5-kwe Reactor Thermoelectric Power System is illustrated by Figure 51. It consisted of: (1) a core, 10.5 in. in diameter by 16 in. long (fuel length), made up of 85 1-in. diameter fuel elements packed in a triangular array, plus 12 internal reflector segments (stainless steel clad BeO) to fill the void space between the periphery of the fuel element array and the circular reactor vessel, (2) inlet and outlet grid plates plus a core liner, to support the fuel elements within the vessel and properly control coolant flow through the fuel element array, (3) the reactor vessel, which was the containment for the NaK coolant and served as the basic structure of the reactor assembly, and (4) the reflector-control assembly, a tapered beryllium reflector, assembled in two identical halves, each having an axially sliding

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Figure 51. 5-kwe Reactor

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control segment, and associated drive and actuator. The reflector mounting hardware allowed the reflector halves to rotate 15° outward for end-of-life shutdown.

The NaK coolant was fed to the reactor through the two pipes which traversed the reflector through grooves in the outer face at the split line between the two halves. Flow was directed within the inlet plenum by vanes and baffling to produce a flat pressure profile over the inlet grid plate. After passing through the core, the flow exited from the outlet plenum through two nozzles, oriented 15° off the reflector split-line axis.

A unique aspect of this reactor, relative to all previous SNAP reactors, was the tapered reflector with sliding control segments. All previous SNAP reactors used rotating drums for control. The primary advantage of this new design was the minimization of the diameter of the neutron shield required to shadow the payload.

b. Core Assembly

The core was designed as a unitized assembly of the 85 fuel elements, inlet and outlet grid plates, fuel element locking rods, internal reflectors, and a core liner. The unit was bench assembled, with poison splines in the coolant channels to insure subcriticality, and then inserted into the reactor vessel. Figure 52 illustrates the component arrangement.

Of the 85 fuel elements in a core assembly; 27 were finned in a left-hand direction, 27 in a right-hand direction, and 31 were unfinned (neutral). The elements were arrayed in a triangular lattice, as shown on Figure 53. Figure 54 shows the details of the array, and the basic dimensional data for the core.

c. Fuel Element

Each fuel element consisted of a segmented (5-piece) fuel rod of hydrided 10% uranium-zirconium alloy, sealed within a Hastelloy X cladding tube. A burnable poison was included as a vapor-deposited coating of gadolinium oxide, applied to the radial surfaces of the fuel slugs. A ceramic coating on the inside surfaces of the cladding was the primary hydrogen-retention barrier. Basic dimensional data for the fuel element are tabulated in Table 17.

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Figure 52. 5-kwe Core Section Details

6531 40134

Figure 53. 5-kwe Core Plan

BASIC DATA	
85 ELEMENTS:	1.025 in. PITCH
LINER ID:	10.450 in.
LINER THICKNESS:	0.015 in. (x2)
CLEARANCE:	0.010 in. (x2)
VESSEL ID:	10.500 in.
VESSEL WALL:	0.050 in. (x2)
VESSEL OD:	10.600 in.
FUEL LENGTH:	16.00 in.
FUEL ELEMENT SPACING	
CLAD-CLAD:	0.020 in.
FIN-CLAD	0.010 in.
CLADDING TO INTERNAL	
REFLECTOR	AS SHOWN

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Figure 54. 5-kwe Core Plan Details

TABLE 17
5-kwe SYSTEM FUEL ELEMENT DIMENSIONS

	Lower Limit	Mean	Upper Limit
Pitch, 1.025		1.025	
Spacing		0.020	
Cladding OD	1.002	1.005	1.008
Cladding Wall, 0.025 ± 0.001 (x 2)	0.024 0.048	0.025 0.050	0.026 0.052
Cladding ID, 0.955 ± 0.001	0.954	0.955	0.956
Glass Thickness, 0.0025 ± 0.0005 (x 2)	0.002 0.004	0.0025 0.005	0.003 0.006
Glass ID	0.948	0.950	0.952
Diametral Gap, 0.011 to 0.013*	0.011	0.012	0.013
Fuel Diameter, $0.9xx \pm 0.0005\uparrow$	0.935	0.938	0.941
Fuel Length (nominal)		16.0	
Straightness (TIR)		0.012	

*Specified Dimensions — all others are reference

†Fuel diameter ground and/or selected to give 0.011 to 0.013 in. diameter gap

The finned elements had three fins, spaced 120° apart which helixed through three turns on a 5-in. pitch. The fins, 0.12 in. wide, were formed by hobbing a heavy-wall tube. The hobbing left a fin height of 0.050 in.; 6 flats were then milled to a fin height of 0.010 in.

A vibration suppressor was installed between the fuel and the cold end cap, as illustrated on Figure 55. This device, which resembled a collar button, prevented axial movement of the fuel rod under the dynamic loadings incurred during launch. It was designed to yield, by collapse of the prebent column, to accommodate axial fuel growth during reactor operation, without overstressing the cladding tube.

The hot end cap was basically a spherical-head closure, butt welded to the cladding tube. The end pins had a 15° chamfer to provide a lead-in for the grid

72-M6-12-11C

Figure 55. 5-kwe Fuel Element

plate during core assembly. The pin diameter on all finned elements, and on the twelve outer neutral elements, was sized for a 0.020 in. (nominal) diametral clearance in the grid plate hole; and, on the inner neutral elements, the pin was sized for a close fit (0.002 to 0.005 in. diametral clearance).

The cold end cap was a flat-head closure, with an interference fit between the ceramic-coated tube ID and the coated OD of the engaging socket portion of the cap. After the cap was inserted, the engaged section of the cap and tube was heated to the softening temperature of the ceramic to fuse the coating into a continuous hydrogen barrier. The cold end pins were all sized for a close fit (0.002 to 0.005 in. diametral clearance) engagement in the grid plate holes. This close-fit engagement was limited to a 0.125-in. length engagement, adjacent to the end surface of the cap. The remainder was reduced in diameter, to facilitate assembly of finned elements adjacent to finned elements.

2. Nuclear Performance

From the results of the reactor parametric studies and the Reactor Thermoelectric System optimization studies, the reactor design, summarized in Table 18, was selected as the conceptual design. The reactor was designed for continuous operation for 5 years at 110 kwt. The thermoelectric system power requirements to produce 5 kwe did not exceed 102 kwt. This had the effect of providing additional end-of-life reactivity margin.

a. Reactivity and Reactivity Lifetime

The reactivity requirements for the design power level and the operating power level are shown in Table 19. The reactor nuclear characteristics are shown in Table 20. As may be seen, a cold, unpoisoned reactivity of \$7.74 was needed to provide the reactivity requirements for 5 years' operation at 110 kwt with a \$0.50 end-of-life reactivity margin. At the operating power level, the EOL reactivity margin increased to \$1.26.

b. Prepoison Selection and Loading

The nuclear parameters which influence the selection of a prepoison material are the burnout rate, the effect on the prompt temperature coefficient, and the specific worth (\$/gm). The non-nuclear parameters are the compatibility of the poison with other core materials, the availability and cost of the poison, and the method of depositing the poison in the core.

TABLE 18
5-kwe SYSTEM REACTOR DESIGN SUMMARY
(85-Element, 1.025-in. Pitch Reactor)

Thermal power (kwt)	110
Number of fuel elements	85
Active fuel length (in.)	16
Lattice pitch (in.)	1.025
Inter-element spacing (mils)	20
Cladding thickness (mils)	25
Glass thickness (mils)	2.5
Gap thickness (mils)	6.0
Fuel diameter (in.)	0.938
Active core diameter (in.)	9.920
Active core volume (liters)	20.3
Core vessel OD (in.)	10.600
Vessel-reflector clearance (in.)	0.060
Reflector midplane thickness (in.)	2.386
Reactor midplane envelope (in.)	15.492
Taper angle (degree)	8.5
Segment stroke (in.)	4.0
Movable angle (degree)	235
Fissile mass (kg U ²³⁵)	9.0
N _H in Fuel (10 ²² at/cc)	5.72

TABLE 19
5-kwe SYSTEM REACTIVITY REQUIREMENTS
(\$)

	Design	Operating
Temperature defect	2.18	2.09
Power defect	0.04	0.03
Equilibrium Xe ¹³⁵	0.18	0.15
Hydrogen redistribution	0.10	0.09
Total recoverable losses	2.50	2.36
U ²³⁵ burnout	0.82	0.69
U ²³⁶ buildup	0.31	0.26
U ²³⁸ burnout	-0.02	-0.02
Gross fission products	0.92	0.78
Equilibrium Sm ¹⁴⁹	0.94	0.90
Hydrogen loss	1.16	0.75
Residual prepoison	0.61	0.76
Total nonrecoverable losses	4.74	4.12
Total requirements	7.24	6.48

TABLE 20
5-kwe SYSTEM NUCLEAR REQUIREMENTS
(\$)

	Design	Operating
Cold, unpoisoned reactivity	7.74	7.74
Hot prepoison loading	3.00	3.00
Recoverable losses	2.50	2.36
Hot, poisoned reactivity	2.24	2.38
Nonrecoverable losses	4.74	4.12
Prepoison burnout	2.39	2.24
EOL reactivity margin	0.50	1.26
Control worth	7.74	7.74
BOL shutdown margin	3.00	3.00

The reason for including a prepoison in the reactor is to reduce the control requirements by reducing the reactivity loss over life. The burnout of the prepoison helps to offset the losses from fuel depletion, fission product buildup, and hydrogen loss.

The poisons which have both nuclear and physical properties which make them suitable for use in SNAP reactors are samarium, gadolinium, gadolinium enriched in Gd^{155} , europium enriched in Eu^{151} , and erbium.

c. Control Worth

Reactivity control was provided by two sliding reflector segments. The movable segments subtended 235° of the reactor circumference, and opened 4-in. windows about the vertical center of the core. The sliding segments moved along the 8.5° taper angle, rather than parallel to the core vessel surface. This also opened gaps between the reflector and core vessel, and between the sliding segment and fixed reflector, which enhanced the control worth.

A reliability analysis indicated that more than two segments did not yield a significant increase in reliability. Two segments were selected, from the standpoint of simplicity and cost.

A total control worth of \$7.74 was determined from two-dimensional DOT calculations in R, Z geometry. In the DOT calculations, it was necessary to assume that the segments subtended the full 360° . The worth obtained from the DOT calculations was multiplied by $235/360$ to obtain the worth of the 235° segments. Calculations were made with the segments closed, 2-in. open, and 4-in. open. The worth of the first 2-in. opening was \$3.95, and the worth from 2 - to 4-in. open was \$3.79. This indicated that the control worth was very nearly linear over the 4-in. stroke.

d. Temperature Coefficients

Partial temperature coefficients of reactivity are shown in Table 21. The fuel coefficient is made up of the spectral, Doppler, and axial fuel expansion coefficients. The isothermal coefficient is the sum of the fuel, grid, and reflector coefficients. The isothermal coefficient was used to determine the reactivity loss in going from ambient to operating temperatures. The partial temperature coefficients were used to predict the kinetic behavior of the reactor.

TABLE 21
5-kwe SYSTEM PARTIAL TEMPERATURE COEFFICIENTS
($\epsilon/^\circ\text{F}$)

Spectral	+0.007
Doppler	-0.022
Fuel axial expansion	<u>-0.051</u>
Fuel coefficient	-0.066
Inlet grid	-0.060
Outlet grid	-0.020
Reflector heating	<u>-0.034</u>
Isothermal	-0.18

The coefficients were determined, using the k-difference method. The various effects which cause the reactivity to change with temperature, such as Doppler broadening of cross sections and thermal expansion of the different reactor components, were evaluated individually, using the APC code. The calculations were performed at BOL conditions. The gadolinium prepoison loading had a positive effect on the spectral component of the fuel coefficient. As the gadolinium burned out over life, the fuel coefficient became more negative. The effect of the burnout of gadolinium was partially offset by the buildup of Sm^{149} fission product. The fuel coefficient at EOL was evaluated to be $-0.108\epsilon/^\circ\text{F}$.

3. Thermal-Hydraulic Performance

The core power requirements vary over the lifetime of the reactor, because of the degradation of the thermoelectric components of the total system. In order to provide margins for uncertainties, the reactor was designed to operate at a level somewhat more severe than system requirements. The "design to" conditions were applied with the beginning-of-life (BOL) configuration when the fuel-cladding gas gap was largest; thus, the thermal gradients were somewhat higher than in the end-of-life (EOL) configuration. Because the exact manufactured configuration of each individual element was not known, other than that its dimensions were within a specified tolerance range, nominal operational conditions had to be calculated for an "ideal" average element. The "ideal"

was primarily the assumption that the element was perfectly straight, and that the fuel segments were all centered in the cladding. The calculational methods used are described in Reference 40. Table 22 shows nominal operating thermal and hydraulic conditions for "design to," BOL, and EOL.

TABLE 22
5-kwe SYSTEM NOMINAL OPERATING CONDITIONS

Parameter	Design*	BOL	EOL
Thermal power (kwt)	110	95	101
Coolant outlet temperature (°F)	1200	1148	1200
Coolant temperature rise, (°F)	100	88	95
Peak fuel centerline temperature (°F)	1310	1247	1286
Peak fuel average temperature (°F)	1292	1232	1270
Maximum cross element ΔT (°F)	21.4	18.8	17.2
Core flowrate (lb/hr)	17900	17500	17400
Average channel velocity (fps)	1.71	1.67	1.66
Average Reynolds number	6940	6760	6740
Minimum Reynolds number	2990	2910	2900
Core pressure drop (psi)	0.19	0.18	0.18

*BOL power distribution

D. THERMOELECTRIC TUBULAR MODULE

The reference design of the thermoelectric power conversion module was selected through a series of trade studies and system performance evaluations. The initial design of the power conversion module was specified as a fractional performance improvement above that measured for the then-current Westinghouse TEM-X modules. The final design of the module was based on a detailed evaluation by Westinghouse Astronuclear Laboratory, utilizing specific performance improvements from achievable TEM-X design modifications.

This section discusses the detailed performance requirements and physical design of the power module. Performance expectations of the module are included, and integration into the 5-kwe Reactor Thermoelectric System is discussed.

1. Design Goals

The following goals were established for the Compact Thermoelectric Developmental Program at Westinghouse Astronuclear Laboratory. These goals were consistent with achievable improvements in module performance which should result from modifications to then-currently tested designs.

The comparative base for performance of the power conversion was chosen as the TEM-X S/N-3 series of power conversion modules. These modules were tested at WANL and at the Santa Susana Field Laboratory of Atomic International. Their performance was considered the best current example of state-of-the-art thermoelectric power module technology. The following performance goals were specified for the reference power conversion modules:

- 1) The reference power module was to have an undegraded, matched-load efficiency of 6.50% at the following average cladding temperatures: $\bar{T}_{\text{hot}} = 1085^{\circ}\text{F}$, and $\bar{T}_{\text{cold}} = 507^{\circ}\text{F}$. These temperatures corresponded to those utilized during the majority of TEM-X module testing. The 6.50% efficiency represented a 30% improvement over the TEM-X S/N-3 series of modules, resulting from utilization of ternary thermoelectric materials and design modifications.
- 2) The reference power conversion module was to have a matched load power degradation of 1.00%/year at 1085°F . Degradation rate was defined as the percentile decrease in module matched-load power (per year) at an average hot-cladding temperature of 1085°F . Additionally, this degradation was to have the following temperature dependence:

$$\text{Rate} = A \exp(-Q/RT)$$

where:

- Rate = matched load power degradation rate (%/yr)
- A = frequency factor (%/yr) ($\sim 7.84 \times 10^7$)
- Q = activation energy of degradation phenomenon
(cal/gm-mole) ($\sim 31,000$ based on TEM-X data)
- R = Boltzman gas constant (1.1038 cal/gm-mole- $^{\circ}\text{R}$)
- T = absolute average temperature of module hot cladding ($^{\circ}\text{R}$)

The effects of the axial temperature gradient variation (on the primary, hot side) were considered in the performance analysis. The degradation rate, as specified, was that for a 100° F average axial gradient throughout the 5-year module operating period.

2. Power Module Performance Requirements

The performance requirements for the power conversion modules were established through system optimization analysis of the 5-kwe Reactor Thermoelectric System. The power output and voltage output requirements for the power conversion modules were specified for the system end-of-life conditions. Analyses had demonstrated that module efficiency varied with the module-to-load power transfer conditions. The overall efficiency was maximized for load voltages slightly greater than one-half of the module open-circuit voltage (defined as the "optimized-load" condition). The electrical power transfer was maximized for load voltages exactly one-half of the open-circuit module voltage (defined as the "matched-load" condition).

The module was designed to operate very near the matched-load condition at the end-of-life (degraded) performance statepoint. Systems analyses subsequently showed that the beginning-of-life module performance statepoint was very near the "optimized-load" condition. Thus, the module nominally operated in the maximum-efficiency to maximum-power-transfer regime during its 5-year lifetime.

The 5-kwe Reactor Thermoelectric System utilized 16 power conversion modules. The "load" for these modules was defined as the system load (5 kwe at 30 v dc) plus the load imposed by the electrical power transmission line. This total module load was 5.13 kwe at 30.8 v dc. Thus, each module had to supply 320.6 w to the total load at a voltage of 3.85 v dc (two parallel strings of eight modules in series were used to satisfy the voltage requirement).

The degradation phenomenon was understood to be the result of a diffusion of tellurium from the p-type washer (thermoelectric element) to the n-type washer, increasing the material resistivity of the latter. Thus, analytic predictions of end-of-life (degraded) performance were made by artificially increasing the n-leg washer resistivity (until the expected decrease in matched-load output power was achieved). The module thermoelectric-couple washer

thicknesses were concurrently optimized for maximum efficiency (with the n-leg washer in the degraded state). This optimization was performed by varying the relative thicknesses of the two washers.

The developmental program goals specified undegraded-performance requirements, while the system integration analyses specified degraded (EOL) performance requirements. Evaluations of the module operating statepoints throughout the 5-year system lifetime had indicated that the total degradation in matched load power should be ~ 6.5 to 7.5% (depending on the specific system design). To facilitate selection of the final reference power module design, and to minimize the effects of system design perturbations on this module, the degraded (EOL) design point was defined to occur with a 7.0% decrease in matched-load module power. The system would therefore operate near the matched-load condition at end of life.

The end-of-life module temperatures (average hot cladding, $\bar{T}_{\text{hot}}$, and average cold cladding, $\bar{T}_{\text{cold}}$) were determined by iterative systems analyses. The beginning-of-life temperatures were calculated concurrently with the EOL performance statepoint. Table 23 summarizes the degraded and undegraded performance requirements for the reference power conversion module.

TABLE 23
5-kwe SYSTEM POWER CONVERSION MODULE
PERFORMANCE REQUIREMENTS

Degraded (EOL) requirements at specific $\bar{T}_{\text{hot}}$ and $\bar{T}_{\text{cold}}$ with 7.0% decrease in matched load power output:

320.6 w/module (power output)

3.85 vdc volts per module (at total load)

P-leg and N-leg thermoelectric couple washer thickness ratio
optimized for maximum module efficiency with increased
N-leg washer resistivity (artificially degraded module)

Undegraded requirements:

6.50% overall efficiency at $\bar{T}_{\text{hot}} = 1085^\circ\text{F}$ and $\bar{T}_{\text{cold}} = 507^\circ\text{F}$
(at matched load, utilizing washer thickness ratio established
for degraded operation).

1.00% /year degradation rate (in matched load power output)
at $\bar{T}_{\text{hot}} = 1085^\circ\text{F}$.

It should also be noted that the degradation in matched-load power output was accompanied by a degradation in the module thermal-to-electrical efficiency. Efficiency degraded at a rate $\sim 70\%$ of that for matched-load power. The rates were therefore linearly proportional in the first approximation. Negligible degradation occurred in the module open-circuit voltage, as indicated by analysis and by TEM-X S/N-3 test data.

3. Module Design and PCS Performance Characteristics

The design of the power conversion modules has been detailed by Westinghouse Astronuclear Laboratory. The module performance associated with this design was calculated at distinct operating conditions and temperatures. Existing performance predictions were fitted accurately to these discrete operating points, and the resulting module characteristics are included in the following sections.

The reference power conversion module design (Figure 56) was based on the TEM-X S/N-3 series of modules, with the following improvements:

- 1) Utilization of ternary thermoelectric materials for the n and p washers (97% achievement of predicted available performance)
- 2) Utilization of electron beam vapor deposited (EBVD) tungsten on the thermoelectric couple faces and on the mica intercouple washers, as a diffusion barrier against tellurium migration
- 3) Incorporation of 0.0015-in. thick mica insulating washers, to minimize the thermal shunting path through these washers.
- 4) Utilization of tungsten inner-conduction-ring interface shoes on both the p and n washers, to minimize tellurium diffusion.

The pertinent design information for the reference module is presented in Table 24.

The design performance of the reference power conversion module is presented in Table 25, as reported by WANL. The temperatures reported are those existing in the 5-kwe Reactor Thermoelectric System at end-of-life conditions. The matched load performance characteristics for this reference module are illustrated in Figure 57.

NOTES: UNLESS OTHERWISE SPECIFIED ON THIS DRAWING

1. APPLICABLE DOCUMENTS

A1 SPEC. EC652N420001 REV1A

POWER MODULE 5KWE

Z+H REACTOR THERMO-ELECTRIC SPACE POWER SYSTEM

NOZZLE	ID +.002 -.000	OD +.000 -.000
A	.751	.791
B	.751	.791
C	.460	.500
D	.460	.500

Figure 56. Thermoelectric Con-
verter Module
(Dwg N652420004)

AI-AEC-13067

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TABLE 24
5-kwe SYSTEM REFERENCE POWER CONVERSION
MODULE DESIGN INFORMATION

Inner cladding diameter (ID) (in.)	0.750
Outer cladding diameter (OD) (in.)	1.644
Number of thermoelectric couples	48
Active circuit length (in.)	15.024
n-type ternary thermoelectric washer thickness (in.)	0.161
p-type ternary thermoelectric washer thickness (in.)	0.149
Mica intercouple insulator thickness (in.)	0.0015

TABLE 25
5-kwe SYSTEM REFERENCE POWER CONVERSION MODULE PERFORMANCE
($\bar{T}_{\text{hot}} = 1140^{\circ}\text{F}$, $\bar{T}_{\text{cold}} = 542^{\circ}\text{F}$ at matched load conditions)

	Undegraded	Degraded
Load Voltage (v dc)	3.86	3.87
Power Output (w)	345	321
Power Input (kwt)	5.420	5.300
Overall Efficiency (%)	6.37	6.06
Matched Load Power Degradation (%)	0	6.96
Efficiency Degradation (%)	0	4.87

6532-40128

Figure 57. 5-kwe System Power Conversion Module Power Output and Efficiency Characteristics

It should be noted that the efficiency degraded $\sim 70\%$ of the matched-load power degradation. The small increase in load voltage (a result of decreased current flow in the module) was considered negligible.

VII. SNAP APPLICATIONS

A. INTRODUCTION

The state-of-the-art reactor-powered electrical systems could provide from 3 months to 10 years of life with reasonable weight and high reliability. Three well-developed electrical power conversion systems could be readily coupled to these reactors: thermoelectric, Rankine, and Brayton systems. A substantial technology base existed in each method of power conversion. Efficiencies ranged from 5% for the thermoelectric system to >28% for the Brayton system. Reliability of operation, long-life characteristics, and cost were different for each conversion system, and the selection of one system over another was generally governed by the mission requirements. The auxiliary devices, such as NaK pumps, NaK accumulators, heat exchangers, instrumentation, radiators, and startup and control systems, had been developed in parallel with the reactor and power conversion systems. The status for these devices was therefore one of ability to support reactor applications for any mission and at any time. Section VII-B discusses key design parametric data for component selection, power levels available, integration requirements, and design data for a significant number of applications. Extrapolation to different sizes, powers, weights, shielding, and auxiliary systems was also possible, and accurate enough for initial planning efforts. Figure 58 presents the applications that are described.

B. DESCRIPTION OF APPLICATIONS

A number of applications had been studied during the life of the SNAP Program, which utilized various power conversion systems. The intent of these applications studies was to determine the design characteristics of a system or group of systems which would satisfy the requirements of the individual applications. These applications varied from deep-ocean implantation to geosynchronous orbiting satellites. The conditions of operation for these varied applications were determined, and evaluated as to their impact on the satellite, vehicle, and other interfacing systems. In each design, consideration was given to the operational safety and reliability requirements of the application, matching these to the design performance values. Eight applications were selected as being most

a. NASA Lunar Base 7-00-174-6

b. NASA Space Base 8-J3-099-35

c. Hardened Missile Base 8-AU26-133-2B

d. Terrestrial Systems

e. NASA Space Station 9-A2-006-72

f. Tactical Communication Satellite 7-O11-249-2A

g. NASA Communication Satellite

h. Marine Systems

Figure 58. SNAP Applications

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typical, and which illustrated the broad capability of the zirconium hydride reactor technology. These applications are described in the following sections.

1. NASA Lunar Base

This base was envisioned as a normal extension of the Apollo and Advanced Apollo Programs, for the purpose of extended manned exploration of the lunar body. The base complement consisted of 12 men for a period of 6 months, with no intermediate resupply. The power system to support the base supplied 20 to 35 kwe for at least 1 year, with prestorage of 6 months prior to operation. Weight of the system was not to exceed 29,475 lb. Three reactor power levels (400, 600, and 1000 kw) were investigated, with two types of power conversion systems: (1) SiGe direct radiating, and (2) PbTe compact intermediate loop thermoelectric.

The power plant would be landed as a separate, unmanned payload, in the vicinity of the base, mounted on a lunar landing vehicle (LLV), and transported to the deployment site. Three deployment modes were investigated: (1) an off-loaded arrangement, where the reactor was buried in a hole dug in the lunar soil, with assembly, checkout, and startup taking place there, (2) an integral assembly with a 4π shield and a fixed radiator, assembled and checked out on earth before launch and (3) an integral 4π shielded system with a fixed radiator, and with a deployable radiator which would be assembled at the site.

2. Zirconium Hydride Organic Rankine Power System for Terrestrial and Marine Applications

A number of important applications for underwater and earth power systems were identified by the National Ocean and Atmospheric Agency (NOAA):

- 1) Remote Land Base – for communications, mining and oil drilling, and base support
- 2) Marine – for vehicle propulsion, communications, mining and oil drilling, anti-submarine warfare, and base support.

The reactor organic Rankine system was selected, because it possessed several attributes important to these applications: high efficiency, compactness, relatively low weight, long life, reliability for unattended operation, no requirement for air environment for operation, and a favorable temperature match between the reactor and the power conversion system fluid.

The study of power systems that would satisfy the applications was based upon the use of the ZrH reactor and the organic Rankine power conversion system. The use of an organic working fluid, rather than water, was based upon certain unique characteristics of the organic fluid cycle:

- 1) The expansion of the saturated organic vapor through the turbine superheats the vapor. This prevents the erosion of the turbine blading, due to liquid droplets in the vapor, and further negates the need to superheat the vapor, as is required when using water as the working fluid.
- 2) There is a relatively low energy release during expansion of the fluid through the turbine. This allows the use of a single turbine stage, large buckets, and maximum energy capability for the single stage. Another characteristic is that the low-speed exit conditions allow the turbine to be designed for a much lower peripheral speed than when water is used.
- 3) The ideal organic fluid cycle achieves a higher Carnot efficiency than the water Rankine cycle at the same operating temperatures.

Four power levels were selected for consideration in the applications (50, 150, 300, and 500 kwe), with nominal lifetimes of 3 years. Three fluids were investigated: (1) pyridine (C. P. -32), (2) monoisopropyl biphenyl (MIPB), and (3) a eutectic mixture of biphenyl and biphenylether (Dowtherm A). The maximum reactor outlet temperature was 800°F, and reactor thermal powers varied between 250 and 2500 kwt.

Reference system designs were selected for the four power levels, using pyridine as the working fluid. These system efficiencies varied between 23.3 and 26.3%. Reactor temperatures were 800°F outlet, and 200°F inlet.

3. Hardened Missile Base

Most of the requirements for the hardened missile bases are classified; but the design requirements for the power system are, for the most part, not classified. These requirements were: (1) 10-year continuous operational life, (2) immunity to outside environment, meteorological conditions, and political disorders, (3) 12-kwe conditioned power, and (4) easily maintained and transported.

The ZrH reactor, coupled with two power conversion systems, was studied to determine the most cost effective system. The two conversion systems studied were PbTe thermoelectric devices, tubular, and cylindrical. Both types of converters were selected, because of their long-lived operational characteristics. The difference between the converter shapes was slight — the reactor outlet temperature for the cylindrical converter was 104°F higher than for the tubular converter, for the same converter efficiency. Since reactor life is a function of core temperature (i.e., life is degraded with higher temperatures), greater lifetimes were theoretically possible with the tubular converter. However, both systems were satisfactory for the requirements.

4. NASA Space Station

Post-Apollo space missions were projected to include the deployment of large, semi-permanent, manned space stations, operating in low earth orbit. The trends in station size were toward substantially increasing its size, crew complement, lifetime, function, and general utility, which caused the supporting electrical requirements also to increase. The power systems studied for this application included the reactor Brayton and reactor thermoelectric power conversion methods. The power requirements were 25 kwe (unconditioned) for a 2-year minimum, with a 5-year goal. The crew size was 6, with an increase to 9 possible for short intervals. The launch mode was integral, fully assembled, checked out, and ready for startup in orbit.

The thermoelectric power system consisted of four separate power generating quadrants which were totally independent, to increase the partial power reliability in the event of quadrant failure. The system delivered 38.5 kwe (29 kwe conditioned) at a thermal power of 680 kwt and a reactor outlet temperature of 1150°F. The weight of the system was 23,800 lb.

The Brayton power conversion system consisted of three independent Brayton power units. Each unit was fully redundant in loops, pumps, heat exchangers, and controls. The system delivered 36.4 kwe (29 kwe conditioned) at a reactor outlet temperature of 1200°F and 127 kwt thermal power. The weight of this system was 24,200 lb.

5. Applications Technology Satellite

Advanced mission studies pointed up the need for a geosynchronous orbiting satellite for technology payloads. The payloads would require long-lived power sources at moderate power levels. The power systems studied were based upon the ZrH reactor, using two fuels (U^{235} and U^{233}) with PbTe converters. The requirements for the power systems were: (1) 5.5 kwe at 30.5 v dc, (2) 3-year life required, with 5-year design goal, and (3) minimum weight. The reference system selected was the U^{233} reactor system, weighing 1575 lb. The reactor outlet temperature was 1200° F, and it produced 85 kw.

6. Zirconium Reactor Brayton System for Marine Application

The analysis of the requirements for this application was directed toward providing the maximum power possible in the smallest possible package, while maintaining neutral buoyancy in a short-term mission. The study focused mainly on taking advantage of reactor Brayton characteristics to obtain an efficient and small envelope power system. The referenced system selected produced 20 kwe net conditioned output, occupied 11 ft³ of volume, and weighed 700 lb. The reactor produced 77 kw, and the overall system efficiency was 26.0%. Reactor temperatures were 1200° F outlet, and 1100° F inlet.

7. NASA Space Base

The space base, as envisioned, evolved from the space station, using common modules, designed and launched for the space station. The space base was substantially larger, and encompassed more experiments. It also provided for a multi-disciplinary research and development task capability. The base also served as an operational center for a wide variety of space missions. The space base was manned by a crew of 48, with power requirements of 100 kwe and growth potential beyond 100 kwe. There was complete redundancy of power systems, including the reactor. The power system study included mercury Rankine, organic Rankine, Brayton, and thermoelectrics. The power system was designed for a minimum maintained lifetime of 10 years. Each reactor power conversion system could supply the complete power requirements for an extended period of time. All of the power conversion systems studied were based upon the ZrH reactor technology. Two scoping studies included the fast reactor Brayton cycle, and the fast reactor thermionic power system.

Power levels varied between 1200 kwt for the Rankine and Brayton systems, and 1800 kwt for the thermoelectric system. The latter reactor was identified as the advanced reactor system. The selected system was the reactor Brayton system, followed by the mercury Rankine and the organic Rankine systems.

8. Space Station - Reactor Brayton System

The space station was a large, semi-permanent, manned habitat in low earth orbit, to be used for scientific experimentation, communications, surveillance, and as a control center for other missions. There was a substantial demand for electrical power that was highly reliable, over extended operational periods. The power system studied for this application was limited to the reactor Brayton system. The power requirements were 25 kwe for a period of 10 years. The launch mode was fully integral, in a checked-out and ready-to-operate condition. There was complete redundancy in the power system.

C. APPLICATIONS - SYSTEM SCHEMATICS

In order to provide additional understanding of the system summary information, the respective schematics have been included. These schematics typically identify the method of heat transfer, number and kinds of loops, temperatures, and heat balances. In addition, the major components of each system are identified, such as heat source, pumps, compensators, radiators, power conversion systems, and their physical relationship. The specific schematics included are:

- 1) NASA Lunar Base - Direct Radiating (Figure 59) and Compact Converter (Figure 60)
- 2) Organic Rankine - Terrestrial and Marine (Figure 61)
- 3) Hardened Missile Base (Figure 62)
- 4) NASA Space Station - Reactor Thermoelectric (Figure 63) and Reactor Brayton (Figure 64)
- 5) Applications Technology Satellite (Figure 65)
- 6) Brayton Marine Application (Figure 66)

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Figure 59. Schematic of Direct Radiating System for NASA Lunar Base

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Figure 60. Schematic of Compact Converter System for NASA Lunar Base

Figure 61. Schematic of Organic Rankine System for Terrestrial and Marine Applications

Figure 62. Schematic of Reactor Thermo-electric System (Natural convection) for Hardened Missile Base

Figure 63. Schematic of Reactor Thermoelectric System (One quadrant) for NASA Space Station

Figure 64. Schematic of Reactor Brayton System (36.4 kwe to TCD) for NASA Space Station

Figure 65. Schematic of Reactor Thermoelectric System for Applications Technology Satellite

Figure 66. Schematic of Reactor Brayton System for Marine Application

- 7) NASA Space Base — Reactor Brayton System (Figure 67), Reactor Mercury Rankine System (Figure 68), Reactor Organic Rankine System (Figure 69), and Reactor Thermoelectric System (Figure 70)
- 8) NASA Space Station — Reactor Brayton System (Figure 71)

NOTES:

- A. AUXILIARY HEAT-REJECTION LOOP NOT SHOWN.
B. OPERATING TEMPERATURE = 80°F IN AND 240°F OUT.
C. HEAT REJECTED = 17.6 kw/t

6531-5049

Figure 67. Schematic of Reactor Brayton System for NASA Space Base

Figure 68. Schematic of Reactor Mercury Rankine System for NASA Space Base

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Figure 69. Schematic of Reactor Organic Rankine System for NASA Space Base

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Figure 70. Schematic of Reactor Thermoelectric System for NASA Space Base

6531-5040

Figure 71. Schematic of Reactor Brayton System for NASA Space Station

VIII. AEROSPACE SAFETY

A. PROGRAM DEFINITION

Before a nuclear reactor is launched into a satellite orbit about the earth, the hazards to the population must necessarily be considered in the event that the reactor reenters the earth's atmosphere, and possibly impacts with the ground. In addition, safe prelaunch handling must be assured.

The objectives of the Aerospace Nuclear Safety Program were to evaluate the nuclear hazards associated with the construction, launch, operation, and disposal of SNAP systems, and to develop methods and designs to assure their radiological safety.

In assessing the safety inherent in the use of SNAP reactor systems, it is important to consider not only the conventional problems associated with conducting space launches, but also possible nuclear problems unique to reactor space power systems. Potential nuclear safety problems for reactor systems can arise only following criticality, and the subsequent generation of fission products.

Accidental criticality may present problems during the prelaunch activity associated with preparation of the system for its space mission. Techniques were developed and successfully used in the SNAP Reactor Program for all safety problems associated with the fabrication, testing, transportation, handling, and launch of one nuclear-powered system.

Postoperational disposal of the orbiting reactor may also pose problems of public safety. Present space reactor systems are maintained subcritical until a satisfactory space orbit has been achieved. A reactor operating in space does not constitute a hazard; but, upon return to earth, the fission products which have been generated could compromise public safety, unless appropriate steps are taken. Techniques were developed to safely dispose of the generated fission products.

In general, there are three techniques for safely disposing of these fission products. The first of these techniques is fission product decay. In any space application, fission products will have decayed to a safe level, if reactor

operation is terminated within 5 years and the fission products are allowed to decay for 50 years or more. The second technique is to recover the reactor, and use standard reprocessing and disposal techniques. The third technique is to dilute the fission products to background or acceptable levels, in either the atmosphere or the ocean.

Exposure to the public can result from external or internal radiation sources. External sources result from direct radiation from reactor debris and from reactor operation on the earth's surface; internal sources are caused by inhalation of fission or activation products released to the atmosphere, and by ingestion of these products released to drinking water or available through the food chain. The three alternative disposal modes lead to somewhat different types and levels of exposure.

A typical disposal mode events tree for a SNAP mission is shown in Figure 72. Results of aerospace safety studies were used to quantitatively assess a mission's safety. Hazards were assessed by summing the probability of exposure for each of the consequences of an event, times the probability of the event, for all events of all possible disposal modes. Probabilities were evaluated from the results of detailed analytical and experimental studies of various aspects of accidents and disposal modes.

The definition of analyses and experiments, and the evaluation of disposal modes and exposure probabilities, were refined iteratively throughout the program. Projects are summarized in the next section, followed by an evaluation of the program.

B. PROJECTS

1. Reactor Disintegration

One method of eliminating any significant hazard from a SNAP reactor reentering the atmosphere would be for the core to disassemble at a high enough altitude for the fuel elements to ablate and disperse in the upper atmosphere. An analytical and experimental program was conducted to determine conditions and designs which would effect disassembly, using the reentry heating available during orbital decay.

6531-4510

Figure 72. Events Leading to Potential Exposure (NASA Space Station)

The study of reactor reentry behavior (temperature response to aerodynamic heating, mode of failure of the vessel, and release of the fuel rods) consisted essentially of an analytical approach, at first. Studies were made of the aerodynamic and inertial characteristics of the SNAP 10A Agena D reactor and vehicle, the reentering satellite trajectory, reentry attitude and oscillation, stagnation and local aerodynamic heat rates, flow regimes, and failure modes. Thermal networks were used to describe the temperature response of the reactor when subjected to aerodynamic heating. Preliminary studies led to the conclusion that the available reentry heating during orbital decay was adequate to effect complete reactor meltdown and fuel ablation of SNAP 10A. This conclusion was based on theoretical aerodynamic calculations.

Subsequently, experimental data were obtained on the aerodynamic force and heating coefficients for the SNAP 10A System. Wind tunnel tests were run to determine local heating coefficients at various angles of attack and at various simulated altitudes. A flight test (RFD-1) of a full-scale mockup of the SNAP 10A reactor was conducted in an actual reentry environment, to obtain aerodynamic data and confirm the theoretically predicted response of the reactor to reentry. A NASA Scout booster rocket launched a nonradioactive SNAP 10A model into a suborbital flight path, with an apogee of 490,000 ft. Much valuable data was obtained by telemetry during the descent. A study of aerodynamic heating in nonequilibrium flow, and in transition regimes between free molecule and equilibrium flow, was initiated because of the results of this test.

In addition to these experimental data, the analysis was improved through the use of more refined methods for calculating aerodynamic heating rates to the reactor. Based upon these experimental data and on the improved analysis, the conclusion was drawn that the reactor vessel walls would melt away, allowing the fuel rod core assembly to fall free, at an altitude of not less than 237,000 ft.

A hyperthermal wind tunnel (RADE) experiment was conducted, using half-scale models of the SNAP 10A and SNAP 8 reactors. Aerodynamic heat fluxes were measured, and ablation and disintegration sequences photographed in detail. Ablation to the point of fuel element release was not achieved. The tests were made to provide data for verifying and improving calculational models

for complicated SNAP reactor geometries. Results indicated that considerable welding between the upper grid plate and the fuel element bundle, and between fuel elements, probably would occur on reentry. Analytical models were developed, and measured parameters calculated. Accurate predictions of the pre-ablation temperature response of reentering SNAP reactors were indicated by the good agreement with experiment.

Reentry aerodynamic heating distributions for SNAP-8 were obtained, based on hypersonic shock tunnel experiments. Results formed the basis for heating distributions used in analytic models. A highly instrumented one-fifth scale model was used. Heat transfer correlations derived from the data provided a relatively complete heat transfer distribution over the SNAP 8 configuration for 0 to 70° angles of attack.

An analysis of the thermal behavior of a SNAP-8 reactor system reentering the earth's atmosphere was made. Analysis was made for a spinning reactor with the neutron shadow shield attached, and for a separated reactor tumbling randomly. The heating for the latter reentry mode was less than for the former. In both cases, however, it was concluded that the SNAP-8 reactor vessel would not ablate sufficiently to release fuel elements at altitudes necessary for complete fuel ablation.

In direct contrast to the situation for long-orbital-decay missions, where the disintegration of the core by reentry heating is desirable, core survival may be desirable for short-orbital-decay missions, to avoid the scatter of fuel elements over a wide area. Tests were made on the ablation of LiH to evaluate its effectiveness as a protection for the top surface of the reactor core vessel, to assure core survival. Measurements of heating rates and ablation were made in an arc-heated wind tunnel. Using the measured data, calculations of LiH ablation were made with the Thermal Analyzer Program. All studies indicated that LiH would provide effective reentry protection.

2. Fuel Rod Reentry Burnup

To determine the conditions of reentry which would cause meltdown and dispersion of a fuel rod, it was necessary to understand the characteristic behavior of the fuel material when it is subjected to a reentry environment. First,

analytical descriptions were formulated, and used to predict the expected behavior of a fuel rod for various reentry trajectories. However, the formulations were complex, and required a number of approximations to the physical phenomena involved (i. e., the aerodynamic heating, the material thermodynamics and metallurgy, and the fuel rod dynamics). The refinement and verification of these analytical approaches required experimental support.

Initial tests were made in the North American Aviation* arc-heated, hyperthermal wind tunnel. Overall fuel rod behavior was studied, as well as the separate effects of cladding and fuel oxidation. The cladding was observed to fail early in the heating sequence. The remaining fuel material was observed to form cracks and heat without ablation for a period, then to melt, flowing viscously back, and ablating rapidly from the leading edge. Oxidation appeared to cause the flow to be less viscous, and the ablation to occur faster.

Measured temperature histories were compared with results of computations using a simple thermal approximation to the fuel material. Reasonably good agreement was obtained. By making preliminary assumptions with regard to reentering rod trajectories and motions and other simplifying assumptions, the arc-jet behavior was extrapolated to the case of reentry. The calculations indicated that fuel rod released at 241,000 ft would melt down on reentry.

Advanced arc-jet tests were performed, to provide experimental support for SNAP fuel element reentry burnup analytical programs and associated fuel particle disintegration studies. Data were obtained on the transient temperature and hydrogen effusion behavior, surface catalytic effects on heat flux, and ablating particle size distributions and compositions. The large number of specimens tested, and the flow conditions and configurations used, allowed several confident correlations and conclusions to be made.

3. Critical Configurations

Ocean-impact disposal modes and land-impact disposal modes, or handling or launch incidents under the proper set of circumstances, would result in water reflection and/or flooding of the SNAP reactor core. Consequently, a program

*Now Rockwell International Corporation

was conducted to determine the reactivities of SNAP 10A and SNAP 8 reactor cores under such conditions, and to test the effectiveness of devices designed to maintain subcriticality.

Initial experiments were carried out with a SNAP 10A core immersed in water. Test apparatus and equipment were designed, and tests conducted in the SNAP 2 Environmental Test Facility. The excess reactivity of a water-reflected bare core was determined to be $\$6.4 \pm 0.6$. The additional worth of internal water was estimated to be $\$7.2 \pm 0.7$. Simple boron poison sleeves surrounding the core were found to render the water-reflected dry core subcritical, but were not adequate to prevent criticality in the water-flooded-and-reflected case.

In subsequent tests, it was demonstrated that:

- (1) A combination void and poison sleeve (designated a shipping sleeve), surrounding the bare core, was capable of maintaining SNAP 10A and SNAP 8 reactor cores subcritical when they were immersed in, and internally flooded by, water.
- (2) Combination void and poison filler blocks that filled the four control drum voids and covered two radial beryllium faces of the reactor were capable of maintaining a beryllium-reflected SNAP 10A reactor subcritical when it was immersed in, but not internally flooded by, water.

4. Reactor Transient Behavior

Destructive nuclear excursions of SNAP reactors are not entirely precluded under certain launch accident or reentry impact conditions. Analysis to determine the maximum reactivity accident was performed, supported by a program of transient behavior tests.

A NaK-filled SNAP 10A/2 type reactor was submerged in water, and a transient initiated by a step insertion of $\$3.60$ excess reactivity. The transient was terminated by core disassembly. Called SNAPTRAN-3, the test was performed at the National Reactor Testing Station by Phillips Petroleum Co. The test was instrumented to record fast and thermal neutron and gamma ray fluxes, temperatures, energy depositions, vessel strain, displacement, acceleration, pressure, noise, photographs, and fission-product-release activity.

Reactivity as a function of time was obtained from power history (energy probe) data, by solving the reactor kinetics equations. Continuous energy deposition was indicated, up to fuel disintegration or massive mechanical disassembly. Temperatures measured at disassembly ranged from 1800 to 2000°F. Photographs showed the vessel to bulge at the center, becoming circular as expansion proceeded. Radial and axial temperature distributions were obtained from fitted energy-probe data. The energy-probe data were integrated to obtain the total core energy release of 32.5 ± 3.6 Mw-sec. A fuel-moderator temperature coefficient of $-0.22\text{¢}/^\circ\text{F}$, up to $\sim 1400^\circ\text{F}$, was derived from the data. A reactivity-feedback coefficient of expansion was derived, using expansion photographs. The value was $-\$6.65/\text{in.}$, up to 0.12 in. expansion, and $\$11.05/\text{in.}$ thereafter.

Specific results were calculated, using theoretical models. Good agreement was obtained for power, energy, inverse period, reactivity, and maximum radial expansion as functions of time. Using improved analytical models based on SNAPTRAN-1 and -2 experimental results, the temperature coefficient was calculated to be $-0.24\text{¢}/^\circ\text{F}$, in good agreement with the experiment.

A series of experiments was performed to determine the fission product release from SNAP 10A/2 fuel, in order to evaluate the radiation hazards from nuclear incidents. Both clad and unclad samples were used. Samples irradiated for the equivalent of 1-year operation at 60 kw were subsequently heated to 2000 and to 3500°F. Temperatures were maintained for a few minutes for melts, and 8 hr for non-melts. All fission products were collected. Activity of isotopes of interest was determined by radiochemical analyses. Fission product activities were measured with a scintillation detector.

Data indicated that product gas release was relatively constant during a run; it was not evolved in a large burst when the temperature was raised. Release fractions were obtained for the main isotopes of interest. Using the maximum excursion temperature from SNAPTRAN-3, these results indicated that releases from clad samples would be several orders of magnitude less than maximum theoretical releases for postulated accidents.

5. Mechanical and Thermochemical Incidents

Accidents, such as impacts, chemical interactions, explosions, and fires, could be postulated during various handling, transportation, and launch situations. To obtain data and information which could be analyzed for evaluating the associated hazards, a series of tests were conducted at Holloman Air Force Base, New Mexico. Full-scale mockup reactors and systems were subjected to conditions similar to the various postulated accidents. Tests were carefully selected to give a statistically representative sampling of credible conditions.

The reactor, with and without NaK, was exposed to a liquid oxygen spray. No damage occurred. A suspended reactor was subjected to an over-pressure of 400 psi for 28 msec, caused by an explosion of TNT. The reflectors were blown off, and the bottom of the reactor vessel deformed inwardly. A reactor immersed in water underwent simulated rupture of the NaK nozzles. Two explosive pulses were followed by a gradual pressure buildup. The core vessel was not damaged.

The reactor assembly was dropped side-on, tail-on, and nose-on onto concrete from 100 ft. Reflector drums were severed. No nuclear hazards were indicated. The reactor was subjected to a simulated launch-abort fire. The reflector separated. The vessel remained intact. A fire test on control-drum lockout pins indicated they would not fail during a free fall to the pad during a launch fire.

Three reactor assemblies, and one unit including the converter and shield, were impacted, at different attitudes, into an 8 by 8 by 16 ft water tank at 400 to 600 ft/sec velocities, simulating a fall into water from orbit apogee. The reflector separated, and other damage occurred in some cases, but disassembly was not sufficient to prevent criticality in water. A reactor was impacted onto concrete at 560 ft/sec. The reactor was completely destroyed. A reactor assembly was impacted at terminal velocity into compacted soil. The reflectors separated, but the core vessel survived.

The tests indicated that no significant nuclear hazard would occur in postulated accidents, unless the reactor were immersed in water.

6. End-of-Life Shutdown

In many aerospace reactor missions, a reliable shutdown device was required to reduce the potential radiological hazards from a reentering reactor. From early studies, it was concluded that:

- 1) From potential radiological consequences of reactor operation, orbital shutdown was always desirable, and, in some cases, necessary for safety in reentering missions in which controlled reentry or complete burnup and dispersal in the upper atmosphere was not achieved.
- 2) Shutdown was not a natural consequence of reactor operation or space environments, so that a highly reliable shutdown mechanism was needed.
- 3) The incorporation in the reactor of radioactive tritium, which decays with a half-life of 12.26 yr to helium-3, a strong neutron absorber, provided an inherently reliable mechanism for shutdown of the SNAP aerospace reactors.
- 4) In orbit lifetimes greater than ~ 5 yr, the tritium shutdown mechanism was practically as effective in reducing radiological hazards from reentering SNAP reactors as prompt shutdown at the end of a 1-year mission.
- 5) The tritium-helium shutdown mechanism would essentially eliminate radiological hazards from reactor missions in orbits of >400 years.
- 6) If tritiding were feasible, the tritium could be incorporated in the present SNAP reactors, with a minimum of design modifications or reactivity penalties.
- 7) Reactivity penalties of the tritium system would not significantly affect reactor performance for most missions.
- 8) Incorporation of tritium in a reactor would not significantly increase the radiological hazards of reactor handling after fabrication.

- 9) From preliminary development considerations, incorporating tritium as a tritide appeared feasible, but experimental information was needed on the tritiding process, to demonstrate the technique.

Later, detailed studies were made of pre-orbital radiological safety, and the feasibility of incorporating tritium into SNAP fuel. The radiological safety study concluded that hazards were not materially increased by the addition of tritium to a SNAP reactor.

7. Disposal Mode Studies

Methods for safely disposing of SNAP reactors after space operation were evaluated. Three principles were employed: decay, dilution, and recovery. Three basic methods had evolved for implementing the disposal principles: (1) boost to a long-lived orbit, (2) destruct, and (3) controlled reentry. The value of each of the methods depended on the characteristics of the reactor.

Boost to a sufficiently long-lived orbit would allow fission products to decay until no reentry hazard would exist. If a reliable end-of-life shutdown device were used, orbit lifetimes as short as 50 years could be adequate. For shorter-lived orbits, reentry radiation hazards are functions of the reactor power history, shutdown mechanism, and orbit lifetime.

C. PROGRAM EVALUATION

The Aerospace Safety Program provided substantial support for the assurance of radiological safety in the use of SNAP systems. The results provided a firm basis for the evaluation of the probability of potential radiological exposure for particular SNAP missions.

The mechanical and thermochemical incidents experimental program provided assurance of safety in handling and prelaunch activities which did not result in immersion of the reactor in water. Use of a shipping sleeve, designed and tested in this program, provided safety under combined water reflection and flooding of the reactor core during prelaunch handling. Techniques were developed and successfully used for all safety problems associated with the fabrication, testing, transportation, handling, and launch of one nuclear-powered SNAP system.

The mechanical and thermochemical incidents experimental program also provided assurance of safety upon intact impact of a reactor reentering the atmosphere at terminal velocities which did not result in water immersion. Experiments to develop assurance of water-immersion subcriticality, through dispersal of fuel elements on impact, did not find a solution short of major design modifications. Design recommendations were made. Single-channel boiling experiments showed that water accumulation in an impact basin or crater provided a mechanism for steady-state operation, for which the radiation exposure to approaching humans would have to be controlled. Experiments which more closely mocked up the reactor were needed, to better define this mechanism. The likelihood of such operation, however, was found to be extremely small.

Experimental and analytical searches for intrinsic water subcriticality methods identified potential combinations of rare earth isotopes which would: (1) provide the needed negative reactivity on immersion, (2) allow normal power operation, and (3) affect the temperature defects and coefficients in a tolerable manner. Establishment of the feasibility of an intrinsically water subcritical (IWS) system for any particular mission would require additional experiments.

Confident prediction of the maximum radiological hazard from a nuclear excursion of a SNAP 10A or SNAP 8 system on earth was provided by results of the series of transient-effects experiments and fission product release experiments.

Disposal of an orbiting reactor with an accumulation of fission products was studied in terms of decay, dispersal, and recovery. For longer-lived orbits, the potential of guaranteeing decay through an intrinsic shutdown mechanism was developed. The feasibility of the tritium-helium system was experimentally demonstrated. Its pre-orbital safety was established. The desirability, feasibility, or practicality of this end-of-life shutdown device for shorter-lived orbits was not demonstrated. Boost to longer-lived orbits was assumed possible, but no feasibility studies were made in this program.

The altitudes necessary for fuel element release, to result in adequate burnup and dispersal, were established. Considerable experimental support

for core-vessel ablation and fuel element ablation determinations was provided. The elimination of significant hazards by adequate dispersal was determined analytically. Better definition of the requirements for dispersal could be obtained from large-particle cloud dispersal experiments, and further transition-flow-region heating and particle ablation measurements. Adequate dispersal was predicted for the SNAP 10A reactor system, but was negatively indicated for the SNAP 8 reactor. An investigation of the feasibility of assisting dispersal with a chemical explosive was begun, and further tests planned.

For low-orbit missions, the possibility of assuring intact recovery of the core by using a LiH shield was established experimentally. Targeting reentry by deboost was assumed to be possible, but its feasibility was not studied in this program.

A procedure for evaluating the overall radiological overexposure hazard for a mission was developed. Probabilities, based on the reliabilities of disposal systems, were determined. The hazardous consequences of each event of each disposal mode were found for the possible modes of exposure. Radiological exposure limits were examined, and found to be adequate and appropriate to SNAP systems.

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