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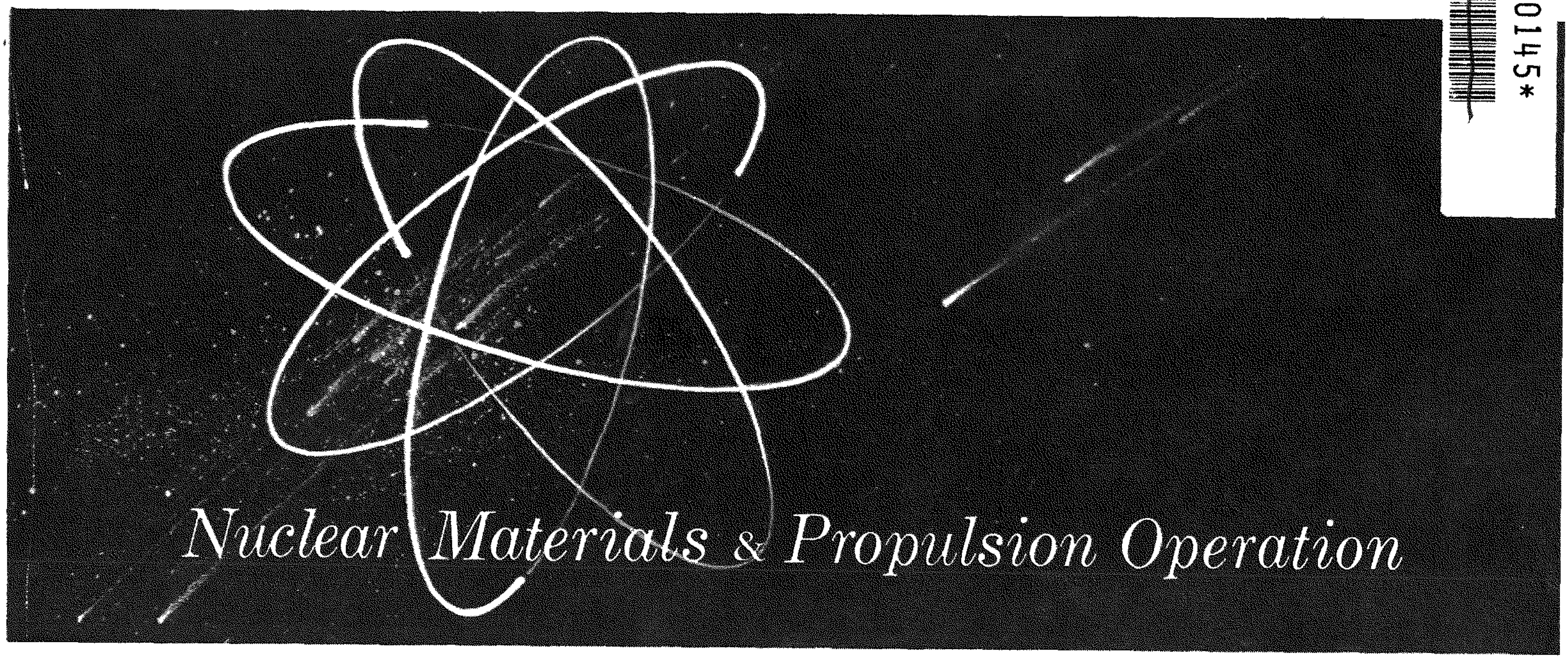

PRESENTATION TO THE ATOMIC ENERGY COMMISSION AND THE AIR FORCE, JUNE 14, 1962

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GE-NMP-62-40

PRESENTATION TO THE ATOMIC ENERGY COMMISSION AND THE AIR FORCE, JUNE 14,1962

OCTOBER 1, 1962

NUCLEAR MATERIALS and PROPULSION OPERATION
FLIGHT PROPULSION LABORATORY DEPARTMENT

GENERAL ELECTRIC

Cincinnati 15, Ohio

DEPARTMENT OF ENERGY DECLASSIFICATION REVIEW	
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NAME: Ted Davis	5 CLASSIFICATION CANCELED
	6 CLASSIFIED INFO BRACKETED
	7 OTHER (SPECIFY):

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INTRODUCTION

This volume contains the charts and backup material presented to the Atomic Energy Commission and Air Force on June 14, 1962.

General Electric's Nuclear Materials and Propulsion Operation (formerly the Aircraft Nuclear Propulsion Department), during its work on the development of a nuclear power plant for manned aircraft, has made highly significant contributions and has developed outstanding capabilities in the following areas:

Reactors
Materials
Testing
Nuclear Systems

These technological capabilities are uniquely applicable to the development of compact, high-performance nuclear power plants.

1. NMPO - ANPD STATISTICS

1 1 REACTORS

1.1.1 R-1 Reactor

The R-1 reactor was designed for the P-1 power plant (first power plant studied for nuclear flight). This was to be a water-moderated reactor utilizing metallic fuel elements. The shield for this reactor was primarily water, supplemented by a lead and steel gamma shield at the forward end.

When the P-1 program was discontinued, the reactor and propulsion components were under final design and development; fuel element development was proceeding satisfactorily; exploratory critical experiments had been performed; and a full-scale shield mockup had been constructed.

1.1.2 HTRE No. 1

In the absence of a specific power plant objective (after the P-1 cancellation), it was decided that the most effective way to provide direction to component development would be to perform a series of nuclear experiments using reactor types with potential application to aircraft propulsion systems These experiments were designated Heat Transfer Reactor Experiments (HTRE).

The first experimental reactor, HTRE No. 1, was an air-cooled water-moderated reactor using nickel chromium concentric ring fuel elements containing uranium dioxide dispersed in nickel chrome. This reactor was first operated at full nuclear power in 1956, in the Core Test Facility (CTF) at the Idaho Test Station. The original test objective was 100 hours operation at full power; the actual operation time was 150 hours at full nuclear power. During operation, fuel element temperatures reached 1800°F and reactor discharge air temperatures were approximately 1400°F. During this experiment, the first known operation of a high-temperature, gas turbine engine from a nuclear heat source was accomplished.

1 1 3 HTRE No. 2

The second experiment, HTRE No. 2, was conducted to test a variety of metallic and ceramic reactor components. For this experiment, the HTRE No. 1 reactor was modified to accommodate large test specimens.

Thirteen different inserts were tested in the HTRE No. 2 reactor for a total of over 1300 hours. The majority of these insert tests were of ceramic fuel element assemblies in which fuel element temperatures reached 3000^{o}F with an exit air temperature of 2200^{o}F.

HTRE No. 2 operation verified the use of hydrided zirconium as a metallic reactor material. During the ceramic insert tests, the integrity of clad materials to prevent water vapor corrosion was verified. Valuable data was collected on fission fragment release, deposition on ducting and other components, filtration, and atmospheric diffusion.

1.1.4 HTRE No. 3

The third experiment, HTRE No. 3, differed from the first two reactors in three basic ways: (1) it had a solid moderator, (2) the reactor was horizontal, and (3) the single reactor supplied heat for two turbojet engines operating simultaneously. The use of lightweight hydrided zirconium as a moderator in place of water, reduced the complexity of the reactor. The HTRE No. 3 experiment was conducted to provide the technical information needed for the design of a ground test prototype power plant and to evaluate and further develop materials.

The HTRE No. 3 reactor was successfully operated for 150 hours with fuel element temperatures of 1800^{o}F and exit air temperatures near 1400^{o}F.

1.2 FUEL ELEMENTS

During GE-ANP operations, approximately 700 metallic fuel cartridges were fabricated. These fuel cartridges were tested for a total of approximately 20,000 hours.

Approximately 1,000,000 ceramic fuel elements were fabricated. Elements of this type were tested at temperatures to greater than 2400^{o}F for a total of approximately 17,000 hours in-pile, and about 70,000 hours non nuclear.

1.3 NMPO TECHNICAL PERSONNEL

At the present time, there are approximately 480 people at Evendale. Of these 160 are technical personnel, including 18 PhD's. They have an average of 15 years experience each and an average of 7 to 8 years each on the project. Approximately 80 are in fundamental materials and 80 in systems, design, and application work.

At the Idaho Test Station there are 33 technical personnel, of which 4 hold PhD's, out of a total employment of 127.

2. CURRENT TECHNICAL PROGRAM

This section describes the nature of the work under the High Temperature Materials Program sponsored by the Atomic Energy Commission. This program is divided into two sections, metallurgy and ceramics. For this presentation it will be regrouped into specific categories rather than discuss each of the 18 separate jobs. Progress reports pertaining to this program are issued monthly and annually in the GEMP series.

IRRADIATION EFFECTS

BeO

The objective of this job is to determine the effects of irradiation on BeO as a function of composition, density, and grain size of the composite body. The determination of the mechanism of growth and subsequent recovery on annealing are also being studied in the program. The most notable observation to date is the influence of preferred orientation on the radiation behavior. Certain compositions can be fabricated with a high degree of preferred orientation and exhibit less damage.

Heat Resistant Alloys

The objectives of this work are to study the nature and causes of the loss in life-time properties of the precipitation-hardened alloys, such as Inconel-X and A-286. A very marked reduction in the stress rupture life of these alloys has been observed in all of the precipitation-hardened alloys studied so far. Solid solution alloys, such as Hastelloy-X, do not show a loss in the stress rupture life. The mechanism of damage is not completely understood; however, it seems to be concentrated in the grain boundary region. The damage, at least in part, is caused by thermal neutrons.

METALLIC FUEL ELEMENTS FOR OXIDIZING ATMOSPHERES

The objective of this work is to study the properties of various oxidation-resistant alloys for fuel element application. This includes the Fe-Cr-Al-Y family of alloys and their compatibility with UO_2. The reaction between aluminum in the alloy and UO_2 to produce uranium metal which migrates to the surface has been largely eliminated by small additions of chromium oxide to the UO_2 to form a barrier and prevent reaction. This job also includes a study of the noble metals and their compatibility with UO_2.

Control Materials

The objective of this work is to provide high temperature control rod materials for various nuclear applications. Earlier studies concentrated on extending the high temperature properties of Eu_2O_3 by using alloys other than Nichrome. However, the reaction of the Eu_2O_3 with yttrium in the Fe-Cr-Y alloy has prevented its use with the Fe-Cr system in the higher temperature regions. More recent work has included a more detailed study of the Eu-O phase diagram and the compatibility of EuO_x with refractory metals, such as tungsten, columbium, and molybdenum.

Shield Materials

The early portion of this work consisted of determining the effects of irradiation on the shield materials that were developed under the GE-ANPD Program, beryllium plus boron and beryllium oxide plus boron carbide. More recent work has included characterization and evaluation of various hydrides containing small quantities of boron.

Materials Properties for Direct Conversion

The objective of this work is to determine properties of materials used in direct conversion devices, such as thermal emf, electrical conductivity, thermal conductivity, and thermal expansion. The main feature of this program is to measure these properties in the higher temperature regions where we have unique capability, i.e., 2000° to $3000^{\circ}C$.

Component Development

The objective of this work is to develop specialized components that provide greater reactor safety and environmental sensing capability at higher temperatures. Some of this work is applicable to nuclear rockets and nuclear space applications as well as civilian applications. Specific components which are being developed are: (1) electromagnetic drive mechanism for control rods; (2) static switching devices for reactor safety; (3) capacitance temperature sensing elements; and (4) high temperature nuclear sensors, such as an a-c ionization chamber which is readily adaptable to telemetering from space.

CERAMIC FUEL ELEMENT TECHNOLOGY

This work includes six items dealing with the development of ceramic fuel elements that are dispersions of stabilized fuels in a BeO matrix. It is an extension of work started in the GE-ANPD Program and now is applicable to the PLUTO Program and the long range programs in the civilian power field.

BeO Matrix

The high thermal conductivity and the good moderating properties make BeO an excellent material for high-temperature gas-cooled reactors. The BeO matrix is being used by the PLUTO Program and in a number of gas-cooled reactors for civilian applications.

Stabilized Fuels

The use of stabilized fuels as dispersions in BeO has been studied in GE-ANPD since 1955. The PLUTO Program has recently adopted this concept of fuel stabilization. Fuel stabilization enables one to add a UO_2 solid solution in air without appreciable volatilization of the UO_2 as UO_3 gas, and prevents a large volume change that occurs on oxidation of UO_2 to U_3O_8 which virtually destroys a BeO matrix containing a dispersion of UO_2. The critical factor in stabilizing UO_2 is to add significant quantities of Y_2O_3 in solution so that the net oxygen-to-metal ratio is less than 2. When the fluorite solid solution is less than 2, it becomes oxygen deficient which apparently improves the stability with regard to volatilization. The earliest stabilized fuels used only Y_2O_3 and UO_2 in solid solution; however, these compositions lead to the formation of the eutectic with BeO at 2870°F and, therefore, are limited to temperatures below the eutectic. Late in the GE-ANPD Program it was found that adding small additions of zirconia to the urania-yttria solid solution produced even better fuel retention and no eutectic reaction was observed at 2870°F. Our present work is aimed at characterizing this latter composition in BeO for use at temperatures of 3000°F and higher.

Coated Particles, Ceramic Coatings on Ceramics, Fission Product Transport

These three items are corollary programs to study the fission product transport in dispersed oxide systems and to develop suitable techniques of preventing fission product escape. One method being investigated to prevent fission product release is to coat the dispersed fuel particle itself, and another is by vapor deposition of the dense ceramic coatings on the fuel element surface.

UO_2-Y_2O_3 System

The latter items is an outgrowth from the studies on the BeO fuel element where it was found that the specific compound of the composition of $UO_3 \cdot 3Y_2O_3$ was extremely stable with respect to volatilization in air to temperatures as high as 2000°C. Furthermore, it was found that dense bodies of this material could be cycled between oxidizing and reducing atmospheres without loss of fuel or strength of the body. This property is unusual in high-temperature fuel element materials and offers unique application in the aerospace field where both atmospheres may be encountered. This composition would be used in fast reactor applications. Its properties and fabrication techniques are being studied in detail.

GRAPHITE FUEL ELEMENT TECHNOLOGY

This work is being performed in co-ordination with the Los Alamos Laboratory and includes the studies of fabrication techniques and high-temperature properties of graphitic systems. Principal studies to date have included the evaluation of urania-zirconia-carbide dispersions in graphite and a detailed study of the curing and sintering process in which furfuryl alcohol has been used as a binder. The alcohol polymerizes and carbonizes without an unusually high release of permanent gases. Detailed studies are being carried out on the rheological properties of the furfuryl alcohol system on heating to high temperatures to achieve the optimum conditions for carbonization.

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FAST NUCLEAR REACTOR SYSTEMS

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3. 710 FAST-SPECTRUM, REFRACTORY-METAL REACTOR

The objective of advanced concept work during the last several years has been to find a reactor system that would be best qualified for various applications to propulsion and mobile power generating equipment. In addition to the usual requirements for high efficiency and low weight, these systems were to have long life, multiple restart and part loading operating capability. The fast-spectrum, refractory-metal reactor appears to offer the best potential for meeting the requirements of all systems studied and known to date.

Moderated and unmoderated reactors utilizing materials having metallic and others having basically ceramic characteristics were studied.

Advantages of the fast-spectrum, refractory-metal reactor result from both the use of the fast-neutron spectrum and the use of metallic materials as listed in the chart.

The fast-spectrum, refractory-metal reactor has been studied by GE-NMPO and its parent organization for several years. Three reports summarizing the essential results have been issued:

GE-NMP 61-13 - "Proposal for the Study of the Application of Small Nuclear Rockets to Maneuverable Orbital Vehicles"

GE-NMP 61-15 - "Development of Fast-Spectrum, Refractory-Metal Reactors for Propulsion and Power Generation"

GE-NMP 62-14 - "Proposal for the Preliminary Systems Design of a Nuclear Rocket Engine for Application in Maneuverable Orbital Vehicles"

A major effort was made to supply answers to all questions raised during presentations to Government agencies or industrial companies. During these activities no barrier problems have been discovered. The only disadvantage that can be listed to date is that no fast-spectrum, refractory-metal reactor has been built and the engineering problems common to a new development have to be solved.

ADVANTAGES OF FAST SPECTRUM REFRACTORY METAL REACTORS

FAST SPECTRUM

- SMALLEST POSSIBLE CORE SIZE.
- SMALLER, LIGHTER SHIELDS THAN FOR LARGER CORES.
- HIGH TEMPERATURE STRUCTURAL MATLS (LOW NEUTRON ABSORPTION CROSS SECTION).

METALLIC MATERIALS

- FABRICATING TECHNIQUES HIGHLY DEVELOPED, CLOSE TOLERANCES POSSIBLE.
- ADEQUATE DUCTILITY
- LOW VAPOR PRESSURE AT HIGH TEMPERATURES.
- NON-REACTIVE WITH HIGHEST MELTING FUEL-UO_2.

AS-25

The evaluation of any application of nuclear power must include a detailed concept of the fuel element used in the reactor. A matrix-type fuel element is proposed which consists of a metal can with front and rear header plates. The element is pierced by a large number of small hydraulic diameter tubes. The entire structure is brazed tight or electron-beam welded. The interstices between the tubes and the tubes and can are filled with a mixture of metal and nuclear fuel. Several different manufacturing processes are being investigated for making these elements. Many of these elements are clustered to form a complete reactor core. Typical data for a fuel element are: 1 inch to 1-1/4 inch across flats, 91 to 169 tubes of 0.035 inch or larger hydraulic diameter, 12 inch lengths or larger depending on reactor application, $60UO_2$ - 40W (volume %) as filler material.

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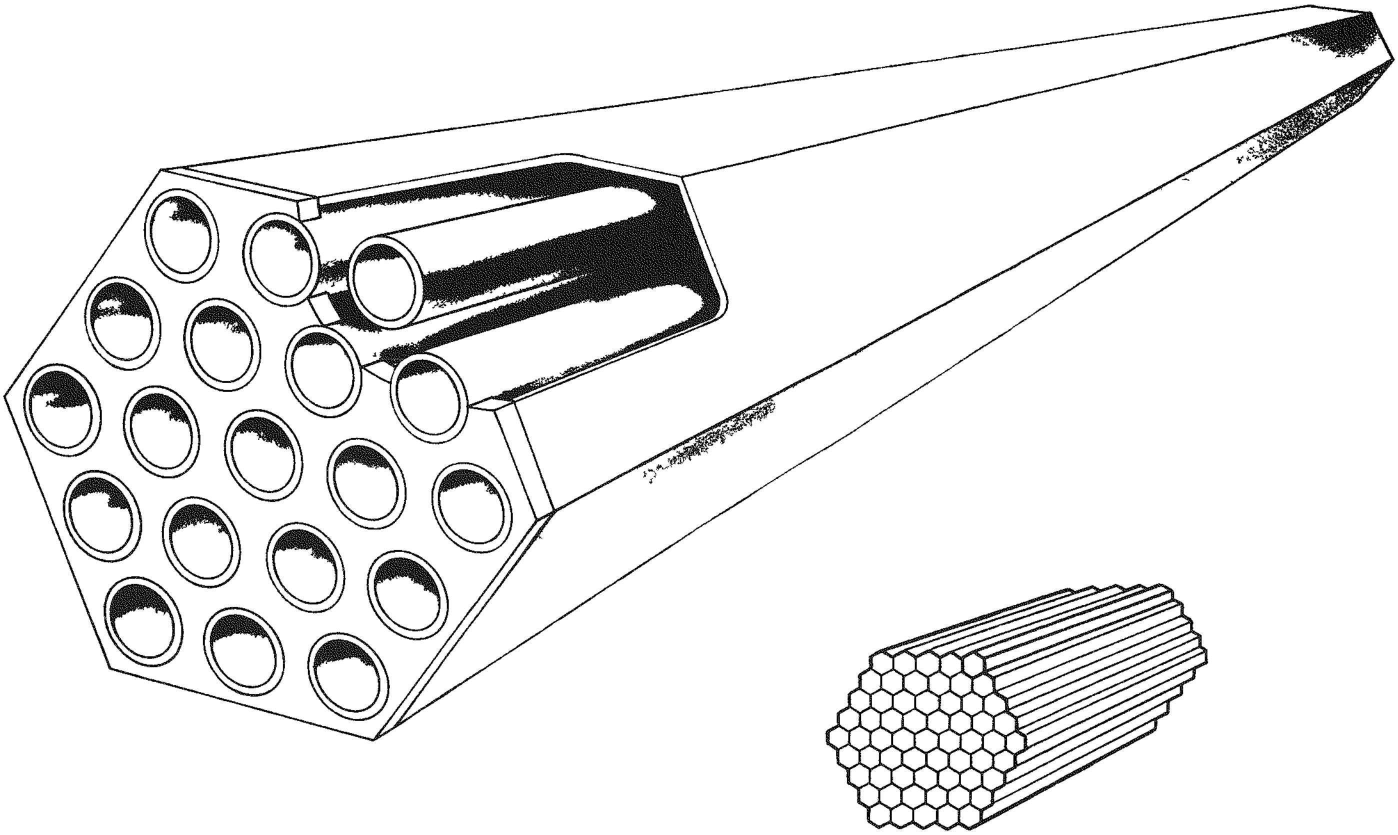
FUEL ELEMENT CONFIGURATION



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In order to maintain thermal stresses in a fuel element at low values, it is essential to use materials that have a high thermal conductivity and good match of thermal expansion coefficients. The addition of W to the fuel UO_2 results in improved thermal conductivity over that of bare UO_2. Furthermore, a composition of $60UO_2$ - 40W (volume %) has nearly the same thermal expansion coefficient as that of Ta. The match of the thermal expansion coefficients is not quite as good for W - 25Re, however, this alloy has improved strengths. Pure W shows a poorer match with the filler material.

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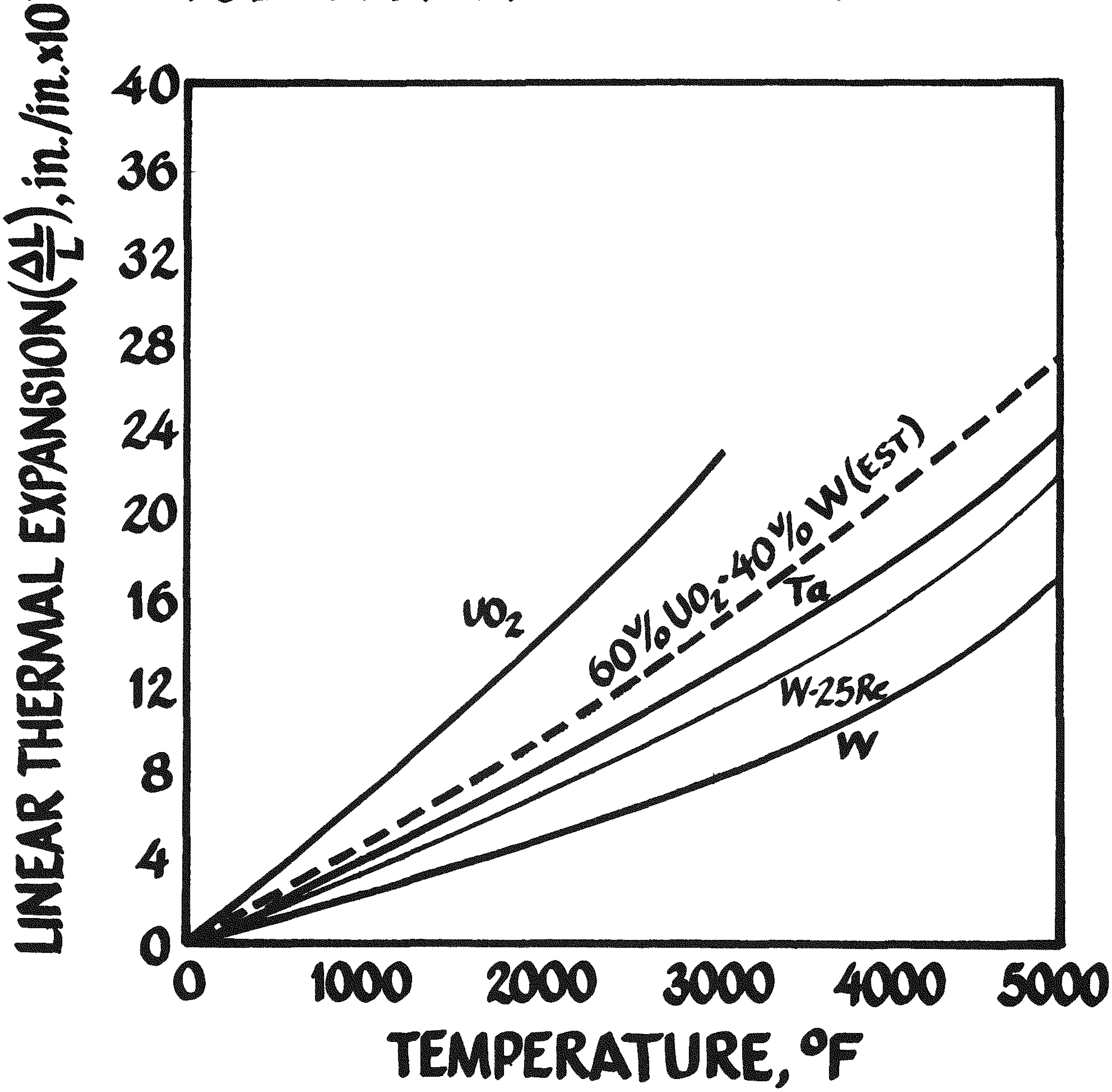
THERMAL EXPANSION OF REFRACTORY FUEL ELEMENT MATERIALS



AS-91

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A typical arrangement of a fast-spectrum, refractory-metal reactor is shown in cross-sectional view. The arrangement is a concept of a reactor of 220 MW (equivalent to approximately 10,000 pounds thrust in nuclear rocket application). It has 151 fuel elements surrounded by a metal oxide reflector that contains 6 control drums which utilize boron carbide sections as poison material.

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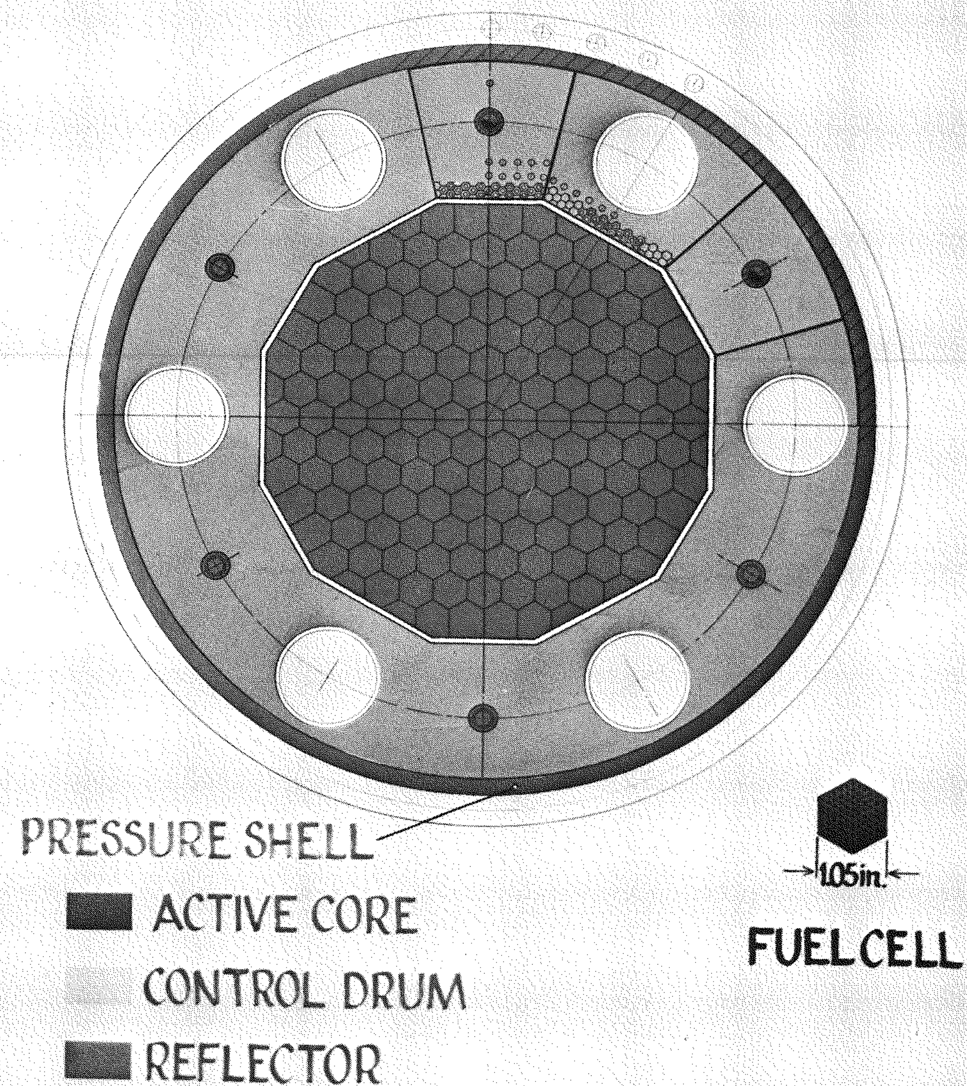
REACTOR ASSEMBLY - END VIEW



CONFIDENTIAL RESTRICTED DATA AS-23

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The side view of the reactor assembly of a 220 MW reactor reflects the objective to keep the number of innovations in the design to a minimum. Accordingly, the coolant flow scheme, support and control arrangements are similar to those used in other designs. The over-all dimensions of the reactor pressure shell are approximately 24 by 24 inches

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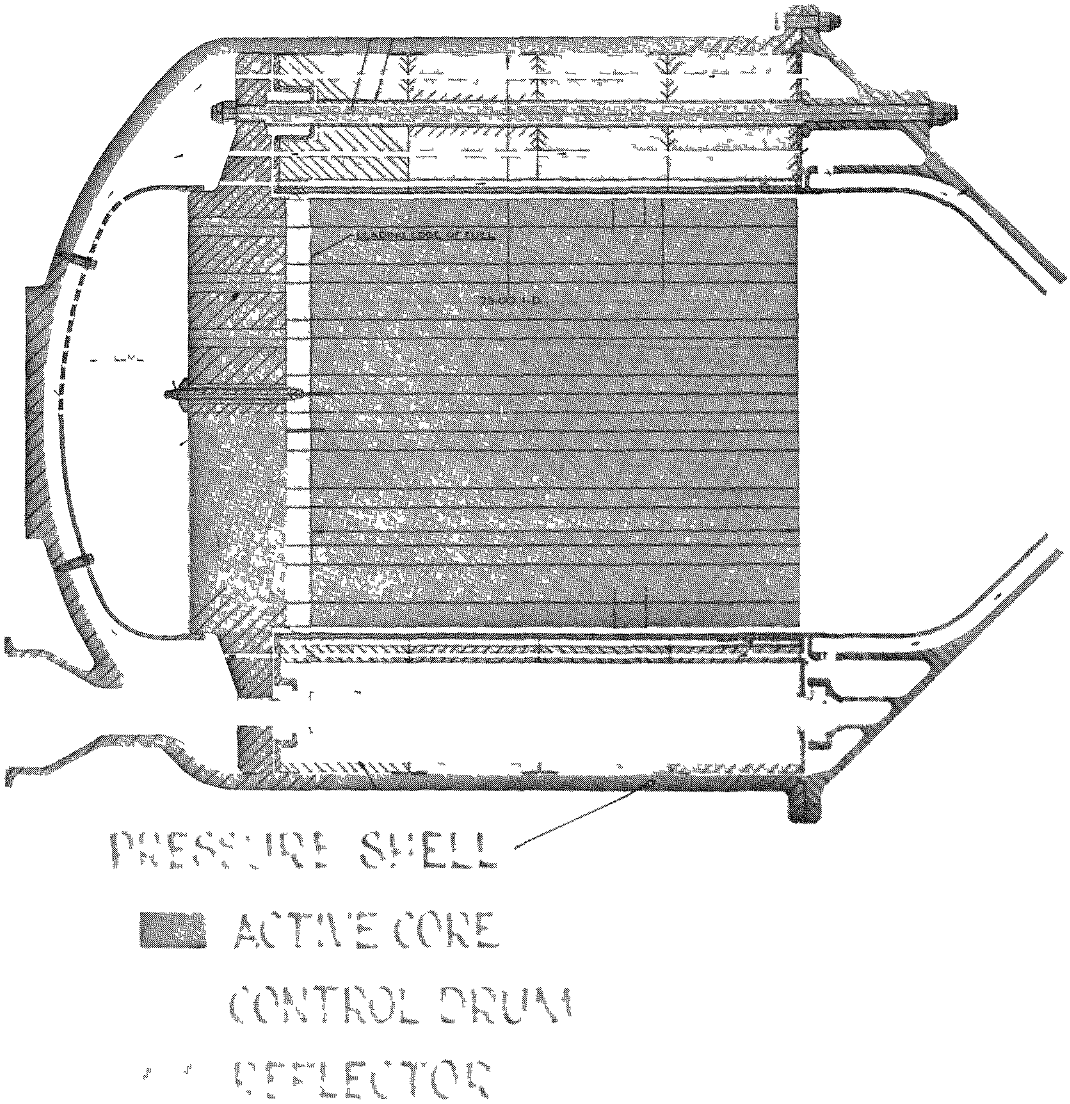
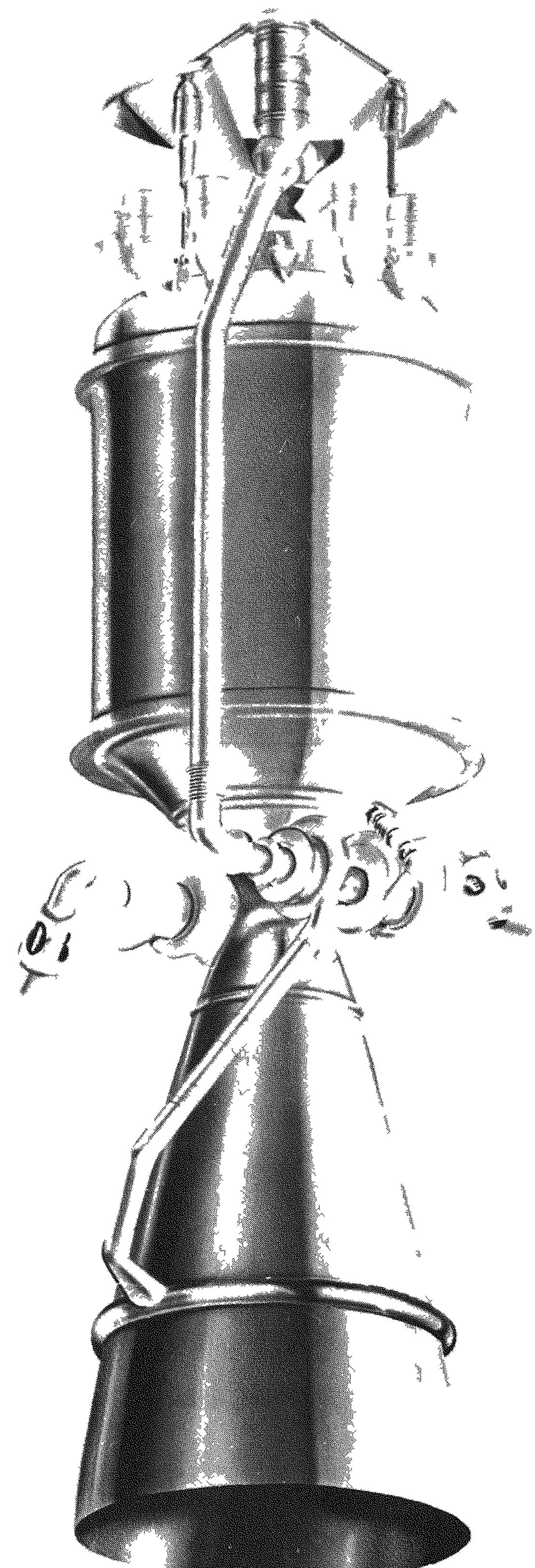
REACTOR ASSEMBLY - SIDE VIEW



AS-22

The attached chart presents one concept of a nuclear rocket arrangement. Several other concepts are being studied, one of which locates the control actuators at the lower end of the reactor and the turbopump system at the upper end.

It shows the propellant flow circuitry external to the reactor from the connection, to the propellant tank, to the turbopump, and to the nozzle cooling manifold. After passing through the nozzle cooling channels and the reflector, the propellant is heated to its maximum temperature in the reactor and discharged through the nozzle. A small amount of propellant from the reactor discharge is mixed with propellant extracted at the end of the nozzle cooling channels and is used to energize the turbine of the propellant pump system.



Total weight is shown as a function of thrust for nuclear rocket engines utilizing fast-spectrum, refractory-metal reactors and graphite reactors. At present GE-NMPO studies have extended over the range of 10,000 to 200,000 pound thrust for the fast-spectrum, refractory-metal system. The data for graphite reactors are based on those available approximately one year ago (GE-ANPD Program) with some extension of these data prepared by GE-NMPO.

The total weights include the reactor (core, reflector, pressure shell, core support, and actuators), nozzle, propellant feed system (turbopumps, turbines, pipes, and valves), thrust structure, and vector control system.

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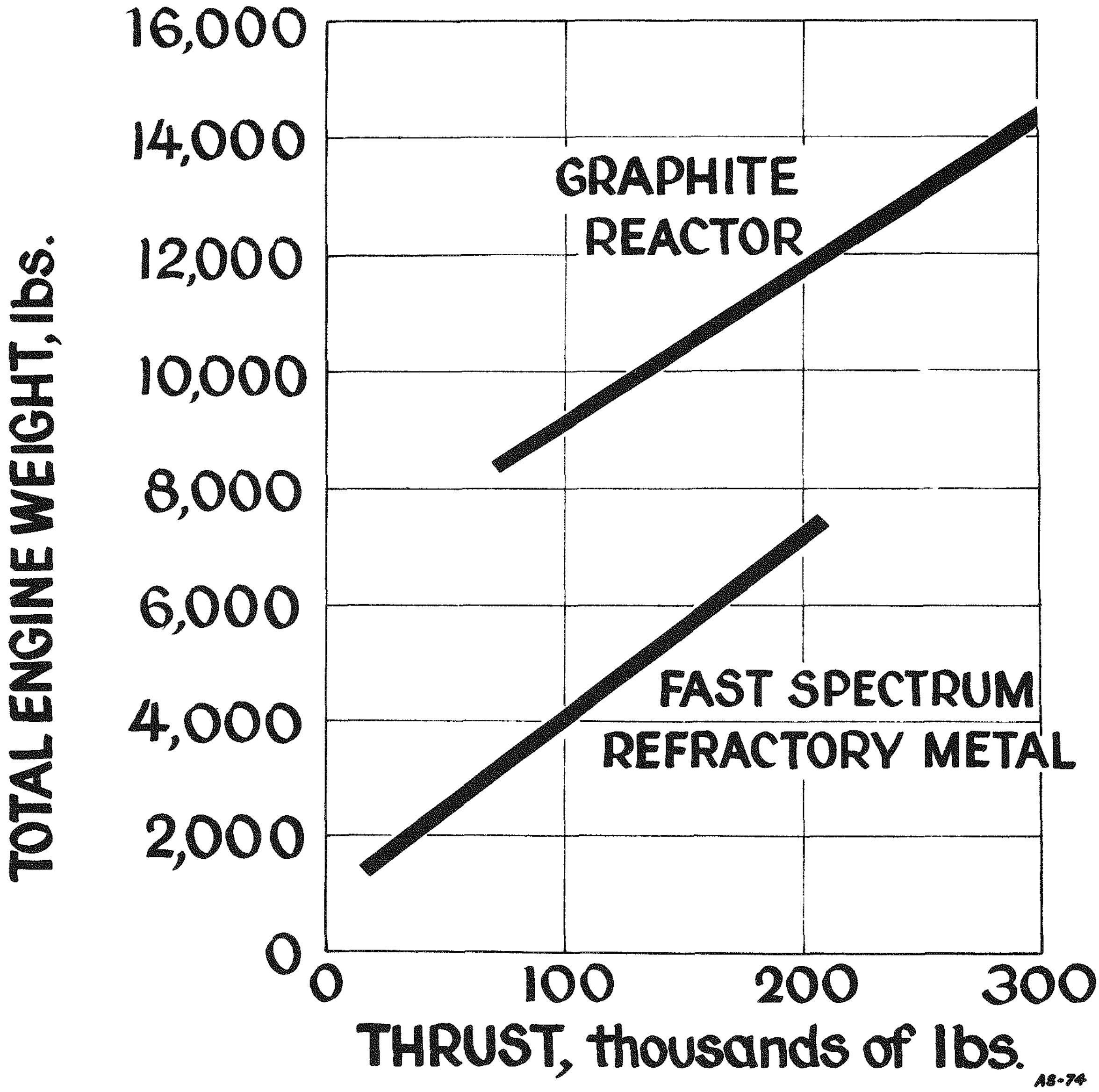
NUCLEAR ROCKET ENGINES THRUST VS TOTAL ENGINE WEIGHT



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Characteristic data are presented for a 10,000-pound-thrust nuclear rocket engine. The values of hydraulic diameter and discharge pressure can be varied to a substantial degree without a major effect on the over-all engine weight. The specific impulse at 4040°F reactor discharge temperature includes an allowance for turbopump drive and represents the value which is expected to be reached in earlier development. Development potential for Ta is considered on the basis of the use of Ta-W alloys to obtain higher strengths. W-Re is expected to be available in shapes suitable for reactor application in approximately two years. The use of pure W will depend on the availability of this material with adequate ductility in the future.

CHARACTERISTICS OF 10,000-POUND-THRUST NUCLEAR ROCKET

REACTOR		
CORE DIAMETER		12 in.
CORE LENGTH		12 in.
HYDRAULIC DIAMETER		0.036 in.
DISCHARGE PRESSURE		900 lb
SPECIFIC IMPULSE (4000°F)		820 sec
I_{sp} DEVEL. POTENTIAL FOR	Ta	860 sec
	W-Re	890 sec
	W	920 sec
ENGINE WEIGHT		1300 lb

AS-24-A

A brief application study was made of a nuclear rocket engine of 10,000 pounds thrust for use in a maneuvering orbital vehicle. It is indicated that for the assumed concept a velocity requirement of 36,900 feet per second would be needed to make a maximum rendezvous and to bring the vehicle back to earth. The term "rendezvous" in this study represents matching perfectly the position, time, and velocity of the two vehicles involved. The lower part of the chart shows the capability of space vehicles propelled by nuclear and chemical energy. It can be seen that the nuclear vehicle can meet the velocity requirement but that the chemically propelled vehicle is deficient by a substantial margin.

The data are based on the following assumptions:

Nuclear rocket engine weight	1920 pounds
Shield weight for 1 hour operation and 24 hours total crew exposure	2080 pounds
Nuclear vehicle tank weight	12% of propellant weight
Specific impulse of nuclear rocket engine	800 and 900 seconds for the respective velocity increment values separated by a slant
Chemical engine weight	500 pounds
Specific impulse of chemical rocket engine	420 seconds

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OVERALL SYSTEM CAPABILITIES

NUMBER OF VEHICLE RINGS	1
VEHICLES PER RING	3
PARKING ORBIT, miles	5,000
MAX. RENDEZVOUS TIME, hr	22
VELOCITY REQUIREMENTS, ft/sec	
Transfer 300-to 5000-mile orbit	7,500
Maximum rendezvous	16,500
Re-entry	12,900
TOTAL	36,900

CAPABILITY, ft/sec	NUCLEAR	CHEMICAL
SATURN-C3		
5,000-LB PAYLOAD	44,200/48,000	27,300
15,000-LB PAYLOAD	34,000/37,000	20,200
NOVA		
5,000-LB PAYLOAD	54,500/59,400	31,700
15,000-LB PAYLOAD	50,100/54,500	28,700

AS-19

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The study was extended to determine the velocity requirements for a mission with two maximum rendezvous'. It can be seen that depending on the velocity increment capability of the individual space vehicles, 6, 9, or 12 vehicles will be needed to perform this type of mission and that the nuclear powered vehicle has the capability to do most of these missions while the chemically powered vehicle is deficient. It is also important to note that a small increase in velocity increment capability (~ 2,000 ft/sec) has a major effect on the economy of the over-all concept; namely the number of vehicles needed for the lower velocity increment capability is about 30 percent higher than for the higher velocity increment value.

OVERALL SYSTEM CAPABILITIES (CONT'D)

NUMBER OF VEHICLE RINGS	2	3	4
VEHICLES PER RING	3	3	3
TOTAL NUMBER OF VEHICLES	6	9	12

PARKING ORBIT, MILES 5000

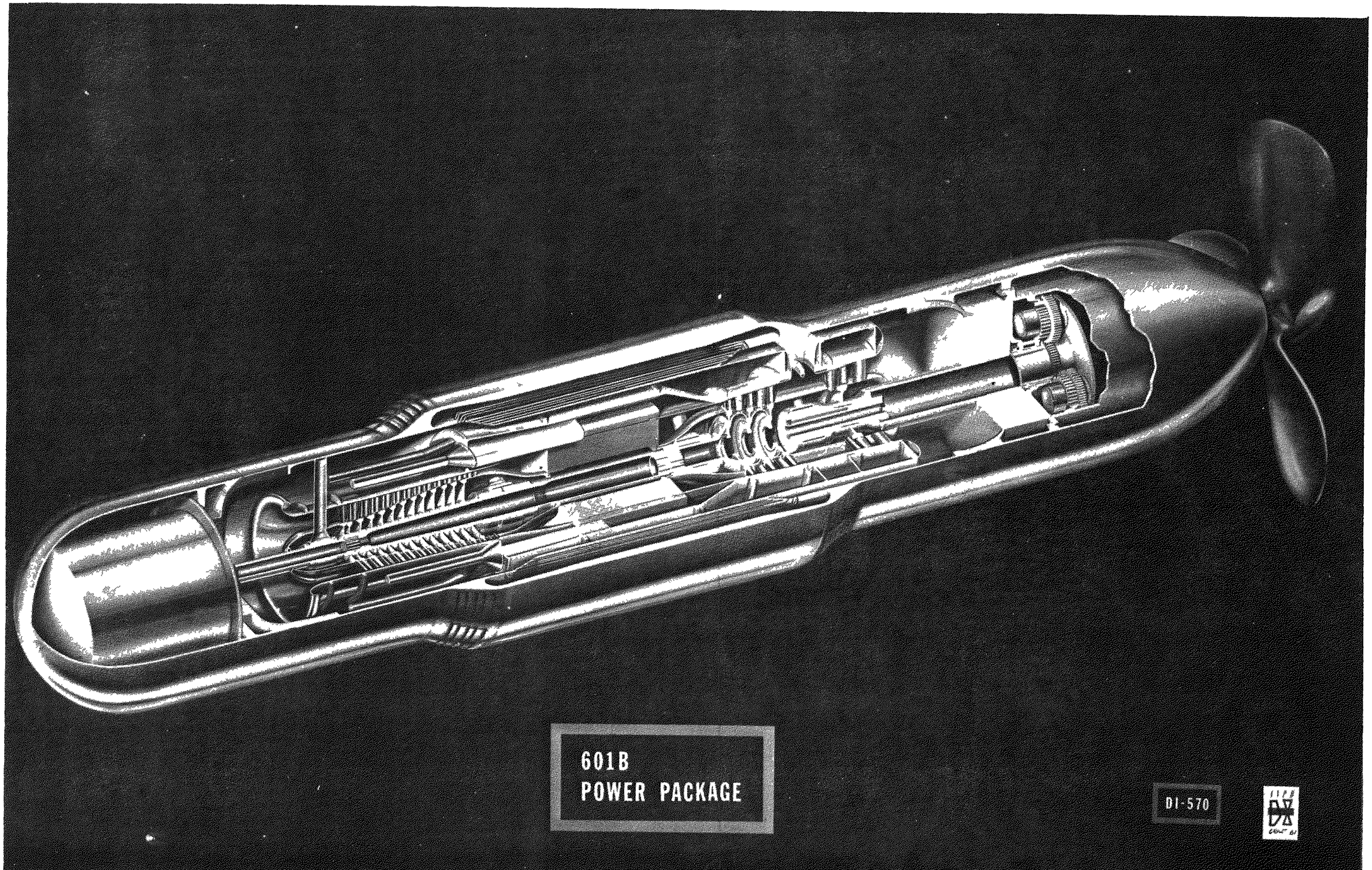
MAXIMUM RENDEZVOUS TIME, hr 22

VELOCITY REQUIREMENTS, ft/sec			
Transfer 300 to 5000 mi. orbit	7,500	7,500	7,500
First maximum rendezvous	11,200	9,500	8,800
Return to parking orbit	11,200	9,500	8,800
Second maximum rendezvous	11,200	9,500	8,800
Re-entry	12,900	12,900	12,900
TOTAL	54,000	48,900	46,800

CAPABILITY, ft/sec	NUCLEAR	CHEMICAL
SATURN-C3		
5000-lb PAYLOAD	44,200/48,000	27,300
15,000-lb PAYLOAD	34,000/37,000	20,200
NOVA		
5000-lb PAYLOAD	54,500/59,400	31,700
15,000-lb PAYLOAD	50,100/54,500	28,700

AS-20

Another attractive application of the fast-spectrum, refractory-metal reactor is in the area of marine propulsion. Small closed-cycle power plants can be mounted outside of the hull of a ship, thereby utilizing the water as part of the required shielding. Shown is a gas cycle power unit with compressor, reactor, turbine and heat exchanger for waste heat rejection to the water. The working fluid is passing through the components in the sequence of the listed components. Power from the turbine is transmitted through a gear set to the propeller. The recommended working fluid is neon because it is chemically and nuclearly inert, has properties sufficiently similar to those of air so that air turbomachinery technology can be applied, and does not diffuse through metals, which is essential for providing long life, maintenance free operation.



601B
POWER PACKAGE

Characteristic data of a 15,000 shaft horsepower marine propulsion unit show a specific weight of 45,000/15,000 or 3 lbs per shaft horsepower. Depending on the requirements of the installation,additional shielding material might be needed which could result in a specific weight of approximately 4 lbs per shaft horsepower. The data were generated on the basis of utilization of a Mo turbine with a capability of 2400°F turbine inlet temperature.

SUMMARY OF PROPULSION POWER PLANT DATA

GENERAL (INCL. PRESS. SHELL & STRUCTURE)	
POWER PLANT O.D.	60.0 IN.
POWER PLANT LENGTH	~360 IN.
POWER PLANT WEIGHT	45,000 LB
INVENTORY	~450 LB U_{235}
OPERATING LIFETIME	5,000 HR
OPERATING POWER	52.0 MW
POWER OUTPUT	15,000 SHP
TURBOMACHINERY	
NEON FLOW RATE	120 LB/SEC.
CPR	4.5 : 1
COMPRESS. INLET PRESS.	100 PSIA
REACTOR	
CORE O.D.	22.0 IN.
CORE I.D	10.0 IN
CORE LENGTH	21.0 IN.
DESIGN POINT OPERATING CHARACTERISTICS	
CORE INLET TEMP.	1215 °R
CORE EXIT TEMP.	3050 °R
PRIMARY TURBINE INLET TEMP.	2860 °R
CORE WALL TEMP. (MAX. AVG.)	3300 °R

1465 Bob Schneithorst

A smaller unit, which could be used as a marine power plant or as a portable electric power plant submerged in water, has been evaluated. The specific power for this unit would be approximately 6.7 lbs per horsepower.

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SUMMARY OF ELECTRICAL POWER PLANT DATA

GENERAL (INCL. PRESS. SHELL & STRUCTURE)

POWER PLANT O.D	45.0 IN.
POWER PLANT LENGTH	~220.0 IN.
POWER PLANT WEIGHT	25,000 LB
INVENTORY	~250 LB U_{235}
OPERATING LIFETIME	5000 HR
OPERATING POWER	12.0 MW
POWER OUTPUT	3750 SHP

TURBOMACHINERY

NEON FLOW RATE	30 LB/SEC
CPR	4.5:1
COMPRESS. INLET PRESS.	100 PSIA

REACTOR

CORE O.D.	13.0 IN.
CORE I.D.	6.0 IN.
CORE LENGTH	12.0 IN.

DESIGN POINT OPERATING CHARACTERISTICS

CORE INLET TEMP.	1215°R
CORE EXIT TEMP.	3050°R
PRIMARY TURBINE INLET TEMP.	2860°R
CORE WALL TEMP. (MAX. AVG.)	3300°R

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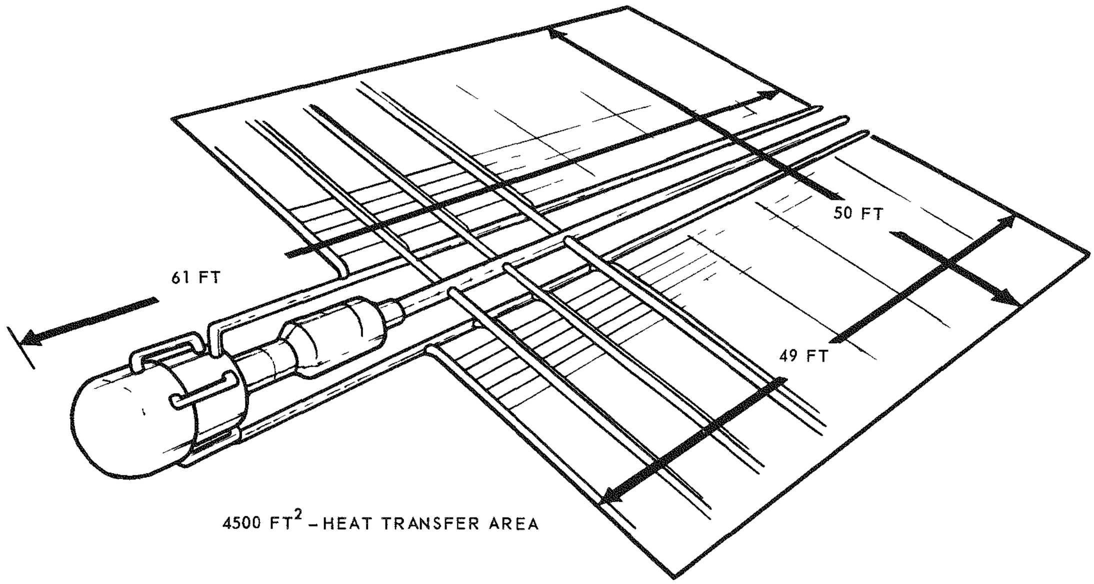
A further application of the fast-spectrum, refractory-metal reactor may be in space power units. A scoping study was made of a 1 MW a space power unit. In this preliminary study, no attention was given to the mechanical arrangement which would meet the requirements of the launching operation. The conceptual drawing shows the relative size of the reactor, turbomachinery, and generator in relation to the radiator. For simplicity a flat plate radiator of tube and fin construction and of approximately square dimensions was assumed.



Conceptual design of a one-megawatt space power plant

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The conceptual arrangement of reactor turbomachinery and generator for a gas cycle space power unit is shown and the flow path of the working fluid is indicated.

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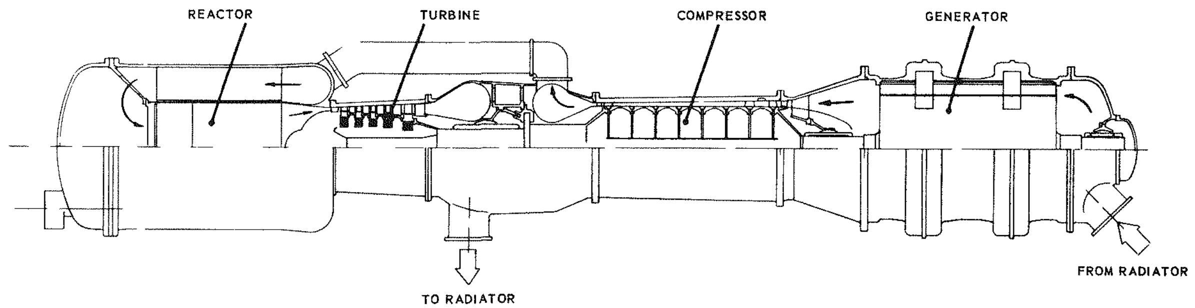

Schematic arrangement of space power plant

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Estimated weight of a 1 MW power unit is shown as a function of turbine inlet temperature. Shown also is the radiator weight which constitutes the major portion of the power unit weight. On the basis of development work on the fast-spectrum, refractory-metal reactor at GE-NMPO and of cooled turbine development at GE-FPLD, it is expected that the technology needed to obtain a specific weight of less than 10 lbs/kw will be available in 1966.

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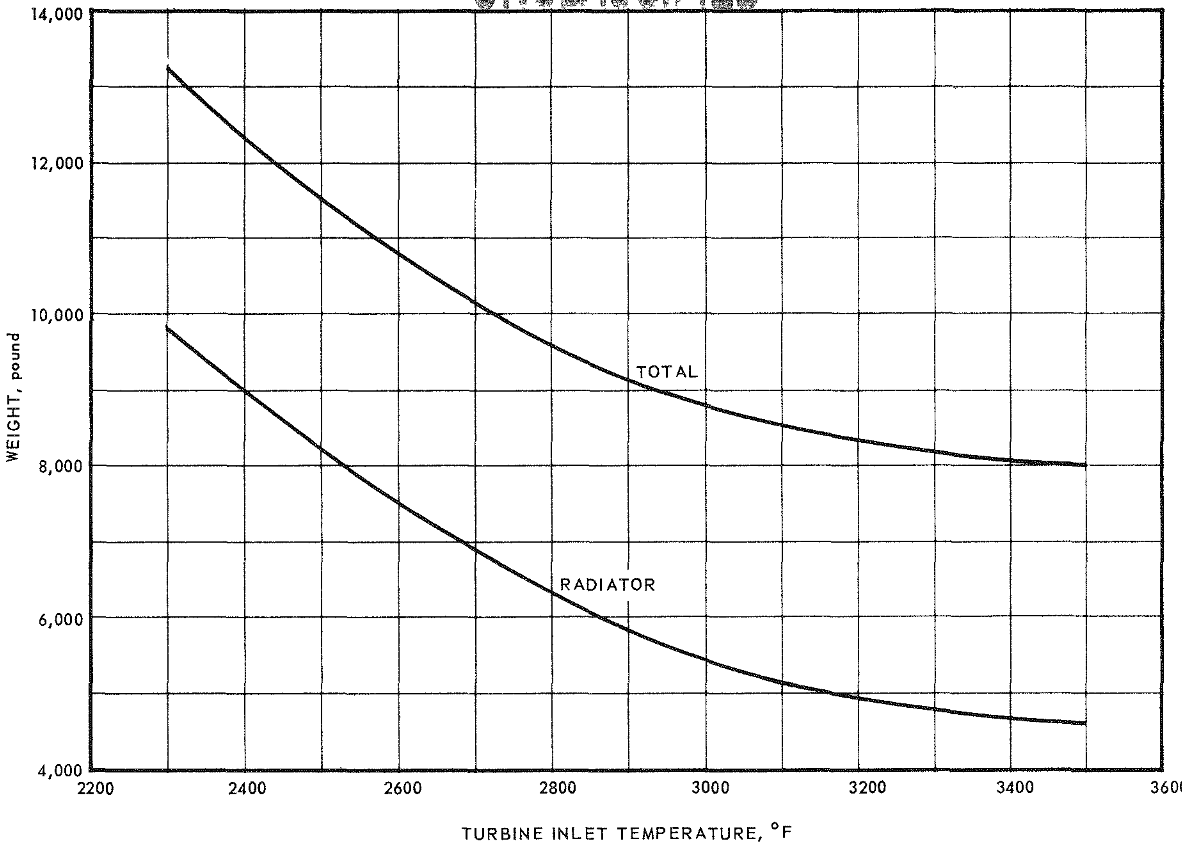

– Weight as a function of turbine inlet temperature

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Estimated weights of the individual components comprising a space power unit are given for several turbine inlet temperatures. Since the present studies have not included mechanical design considerations particularly for storing of the power unit in a launch vehicle, a contingency of 20% on the radiator weight was included to give a second set of total power unit weights. It is believed that these weights represent realistic estimates and that they can be realized in an early state of development.

TABLE

ONE-MEGAWATT POWER UNIT WEIGHTS[a]

Component	Turbine Inlet Temperature, °F 2300	2800	3100	3500
Radiator	9,800	6,300	5,200	4,600
Ducting	480	320	280	240
Reactor	1,310	1,310	1,400	1,500
Turbomachinery	180	190	210	230
Generator	1,000	1,000	1,000	1,000
Reactor controls and structure	130	130	130	130
Structure	300	300	300	300
Total weight	13,200	9,550	8,520	8,000
Total weight including 20% contingency for radiator weight	15,160	10,810	9,560	8,920

[a]Not including shield and system control weight. All weights are in pounds.

4. 710 DEVELOPMENT PROGRAM

4.1 INTRODUCTION

The development program of propulsion and mobile power units utilizing the fast-spectrum, refractory-metal reactor contains two major elements: (1) reactor technology oriented work and (2) applications oriented work. The work for reactor technology development at present is not tied to any specific application. However, the reactor tentatively selected for initial tests is of a size that would provide information readily adaptable to several applications presently under consideration. The applications oriented work may include several specific studies and designs. This work should be initiated at the earliest possible time in order to provide important information for guidance of the reactor technology oriented work. Under the presently proposed program, reactor testing would be accomplished in approximately 4 years. If the work concerned with special applications is carried out simultaneously, a prototype unit for a specific application could be ready for test in approximately 5 years. Considering a substantial reliability program, a propulsion or mobile power generation unit could be available for operational tests in approximately 6-1/2 years.

DEVELOPMENT PROGRAM PHASING

REACTOR TECHNOLOGY

PHASE I
REACTOR DESIGN STUDY

PHASE II
TEST REACTOR FINAL DESIGN

PHASE III
REACTOR TESTING

APPLICATIONS

PHASE I — DESIGN STUDIES OF PROPULSION AND POWER GENERATION UNITS FOR SPECIAL APPLICATIONS

PHASE II — FINAL DESIGN OF PROPULSION AND POWER GENERATION UNITS FOR SPECIAL APPLICATIONS

PHASE IV — CONSTRUCTION AND PROOF TESTS OF PROTOTYPE PROPULSION AND POWER UNITS

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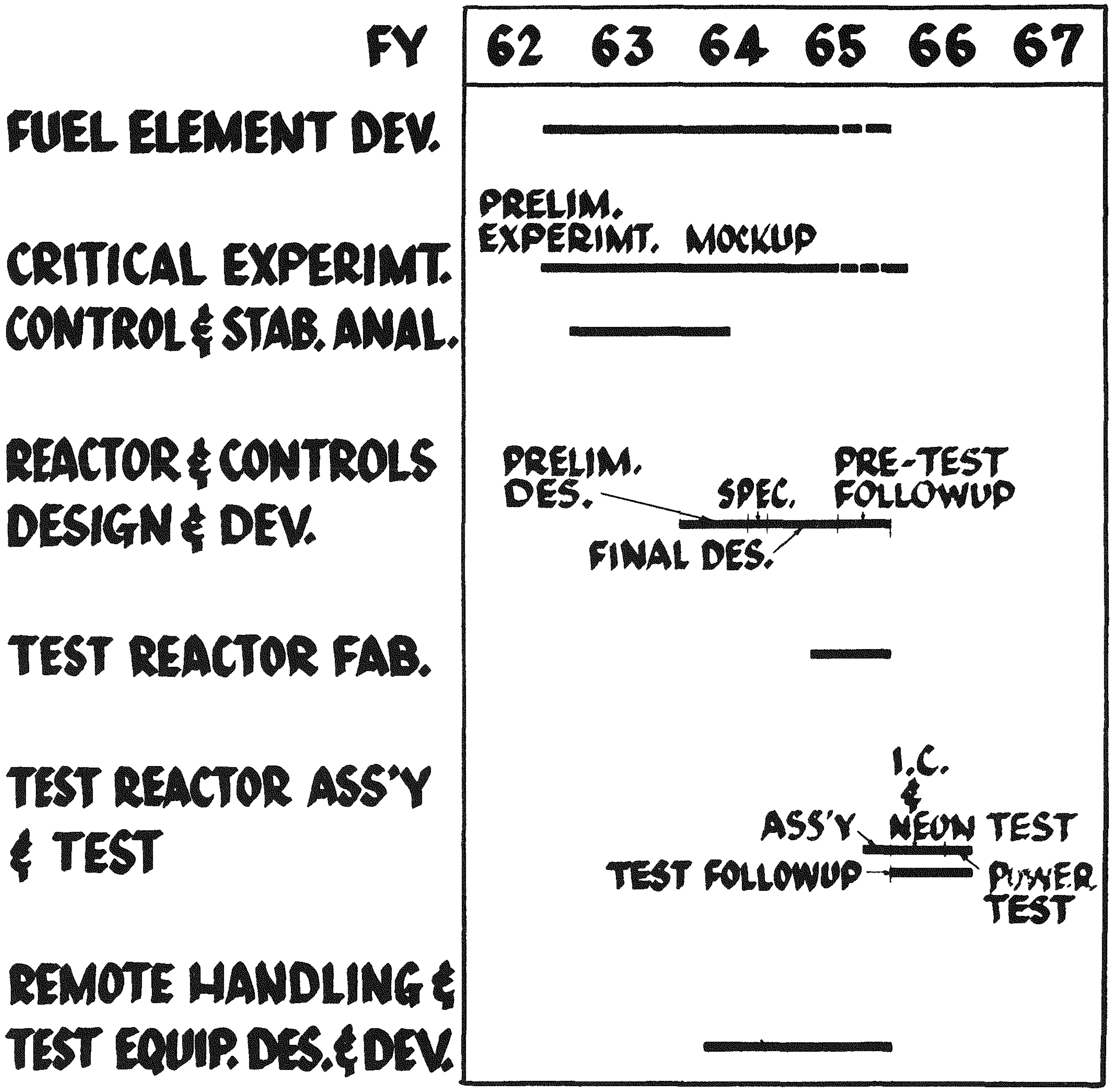
The schedule for the 710 reactor development shows that during Fiscal Year 1963, the work will be concentrated on fuel element development, basic critical experiments, and control and stability analysis. Reactor design will commence in FY 1964 with testing starting in FY 1966. Initial tests will be made in a closed loop arrangement with neon as working fluid. These tests will be at low power level (approximately 10 MW) and high temperature to verify the stability and temperature capability of the reactor. These tests will be followed by open loop tests with hydrogen as coolant to verify the full power capability of the test reactor.

SCHEDULE FOR 710 REACTOR DEVELOPMENT



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4.2 PHASE I EFFORT TO BE ACCOMPLISHED IN FISCAL YEAR 1963

4.2.1 Fuel Element Development

The fuel element shown in the chart is representative of that to be used in the fast reactor system. It consists of a forward attachment section, an integral forward section, and the fueled core region. The reflector and fueled regions are contained within tubes for coolant flow and an external clad. The precise dimensions of the element are dependent on the specific application. Studies to date indicate that the coolant passage tubes will be in the order of 0.035 to 0.100 inches inside diameter and the element from 0.75 to 1.25 inches across the flats of the hex.

Three fabrication techniques are currently being considered:

1. Filling the enclosing hexagonal clad with a powdered matrix material with die pressure and subsequent end welding, brazing, and autoclave sintering.
2. Pre-sintered compacts in wafer form which are machined and assembled with the tubes and hexagonal clad and then autoclaved.
3. Coat small hexagonal tubes with the matrix material stacking into an assembly and autoclave.

Present fabrication effort will concentrate on method number 2, for an element with 0.052-inch tubes and 1.03-inch hex. Concurrently, materials and tools necessary for development will be purchased. Testing will be initiated and stress analysis effort will begin. Ten fuel elements are scheduled for completion during Fiscal Year 1963.

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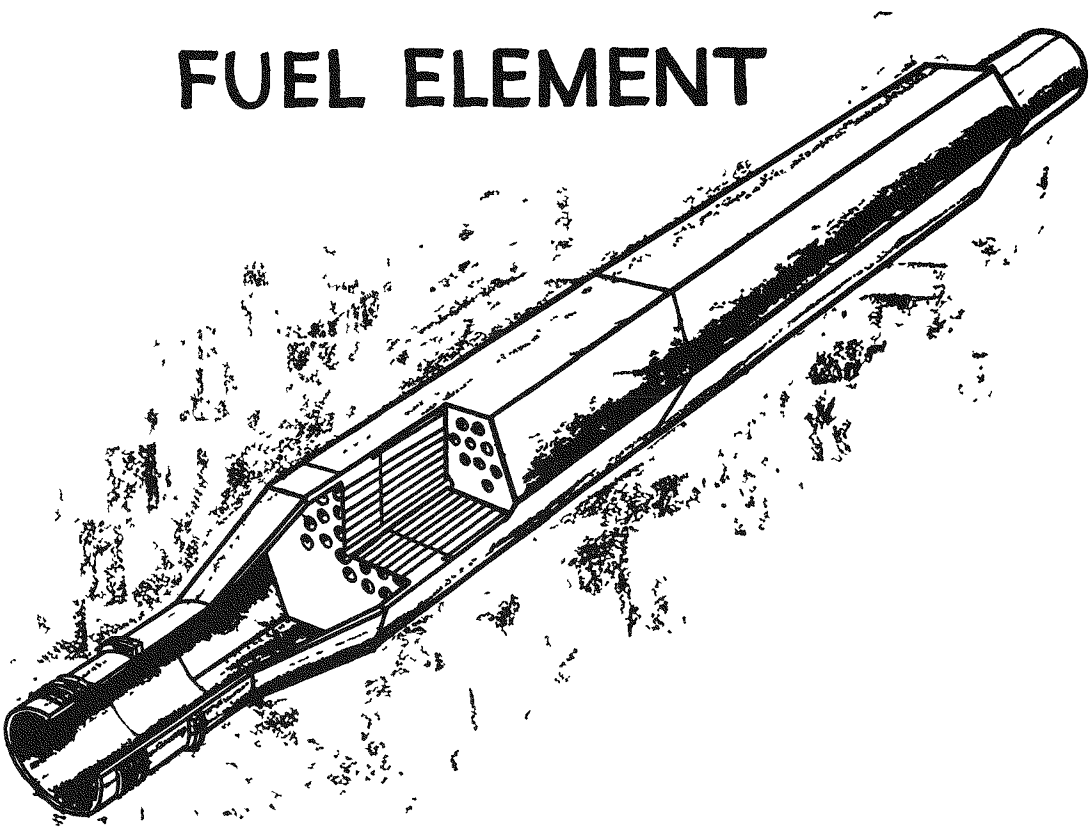
FUEL ELEMENT



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4.2.2 Materials

METALLIC FUEL ELEMENTS FOR NON-OXIDIZING ATMOSPHERES

This work, also part of the High-Temperature Materials Program, is more directly applicable to the fast reactor nuclear rocket which werare discussing today. Principal studies in this category are aimed at understanding phenomena and determining the properties of cermets containing a refractory metal, such as uranium, tungsten, tantalum, columbium, and molybdenum, with uranium dioxide.

High-Temperature Chemistry of UO_2

UO_2 has the highest melting point of the uranium compounds. Of the refractory metals listed, only tantalum, tungsten, and molybdenum have higher melting points than UO_2. In the temperature region above 2000°C, uranium dioxide tends to become oxygen deficient and to volatilize. Since the refractory metals can become partial reducing agents in these temperature regions, it is important to know what degree of oxygen deficiency is needed for a particular refractory metal to be compatible with UO_{2-x}. This program includes compatibility studies of UO_2 with the refractory metals mentioned in the form of encapsulated components. In addition, the high-temperature chemistry of UO_2 itself is being studied in order to understand the disproportionation of UO_{2-x} on cooling and the volatilization and vapor products when sintered by itself or in conjunction with refractory metals. Some work is being carried out on solid solutions of UO_2 in an effort to increase the melting point of the oxide.

High-Temperature Properties

The high-temperature properties of refractory metals for composites of refractory metals with UO_2 are also being studied. A few of the properties are listed in the items below.

The stress rupture properties of the refractory metals are being determined and most recent data on the W-Re and Ta-W alloys are shown in Figure 1. These measurements show that there is no effect of electron beam welding on the stress rupture life of the alloys. Similar data are shown in Figure 2 for molybdenum and Mo-Re alloy at a lower temperature. Table 1 compares the elongation properties of these alloys under various conditions of stress and temperatures.

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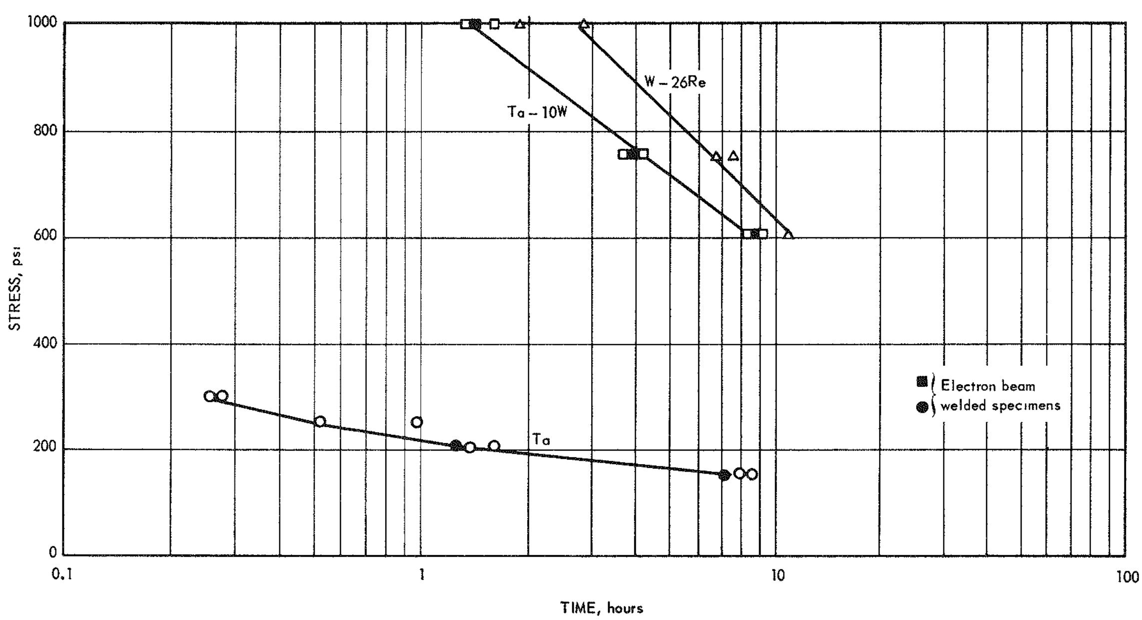

Fig. 1 – Rupture life of W – 26Re alloy and Ta – 10W alloy compared to tantalum at 2600°C in hydrogen

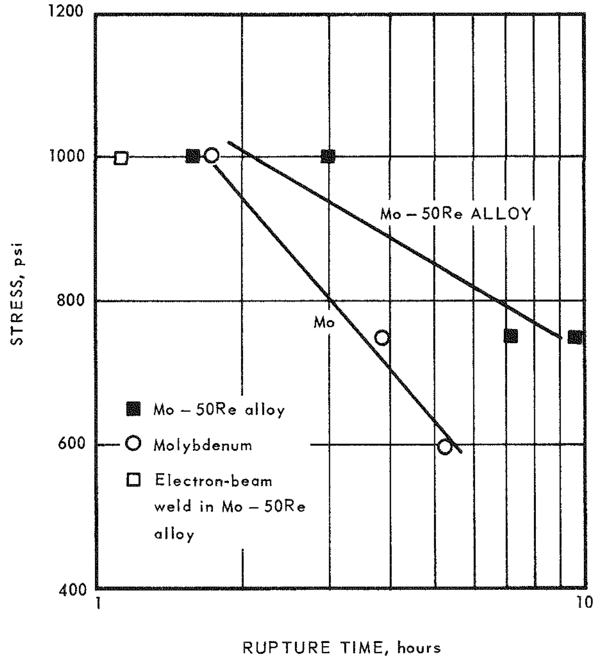

Fig. 2 - Rupture life of Mo – 50Re alloy compared to molybdenum at 2200°C in hydrogen

TABLE 1

COMPARISON OF STRESS RUPTURE PROPERTIES

Material Tested	Test Temperature, °C	Stress, psi	Rupture Time, hr	Time To Effect Indicated Strain, hr	
				2% Elongation	5% Elongation
W - 26Re	2600	750	6.8	3.2	6.8
Ta - 10W	2600	600	9.2	0.36	0.93
Ta	2600	150	8.1	0.41	1.3
Mo - 50Re	2200	750	9.6	0.48	1.9

It is noted that the Ta and Mo alloys are more ductile than the W-Re alloys. Metallographic examination indicates that the mode failure of the W-Re system is different from the other alloys. The next two figures, Figures 3 and 4, show the thermal emf and the electrical resistivity of UO_2 at 2200°C. Figure 5 is a plot of the thermal expansion of UO_2 to 2200°C. In addition, the properties of UO_2-refractory metal capsules are being determined after heating for long periods of time in hydrogen or an inert atmosphere. These studies are aimed at defining the limits of temperature and time of these composites and determining what changes are occurring internally in the capsule when heated both isothermally and by a reactor flux.

Joining Technology

The objective of this work is to develop techniques of joining and encapsulating refractory metal systems. In the early work, brazes were developed for tungsten, molybdenum, columbium, etc., with a specific braze being used for each refractory metal. This technique tended to decrease the maximum temperature to the eutectic temperature of the braze composition; therefore, other techniques were studied. Two of the techniques which have been particularly successful have been diffusion bonding and electron beam welding. The diffusion bonds can be made by heating the metal interface under vacuum to temperatures between 3000°F and 4000°F under very light loads. Complete loss of the original interface is achieved.

Fabrication Technology

Fabrication methods already exist for preparing and encapsulating refractory metal-UO_2 composites. However, further work in this area is being carried out in the High-Temperature Materials Program to improve the fabrication techniques and to optimize processing conditions for making refractory metal-UO_2 fuel elements. Techniques are being studied on ways of making oxygen deficient UO_2 that is compatible with the refractory metal. Also under study, is the technique of achieving a continuous metallic matrix so that the core material can provide load bearing capability. The sintering conditions, such as atmosphere, and temperature, are being studied in order to achieve high densities without appreciable loss of fuel. Further studies are being carried out to determine the optimum conditions for achieving the core and clad bond between the various refractory metals.

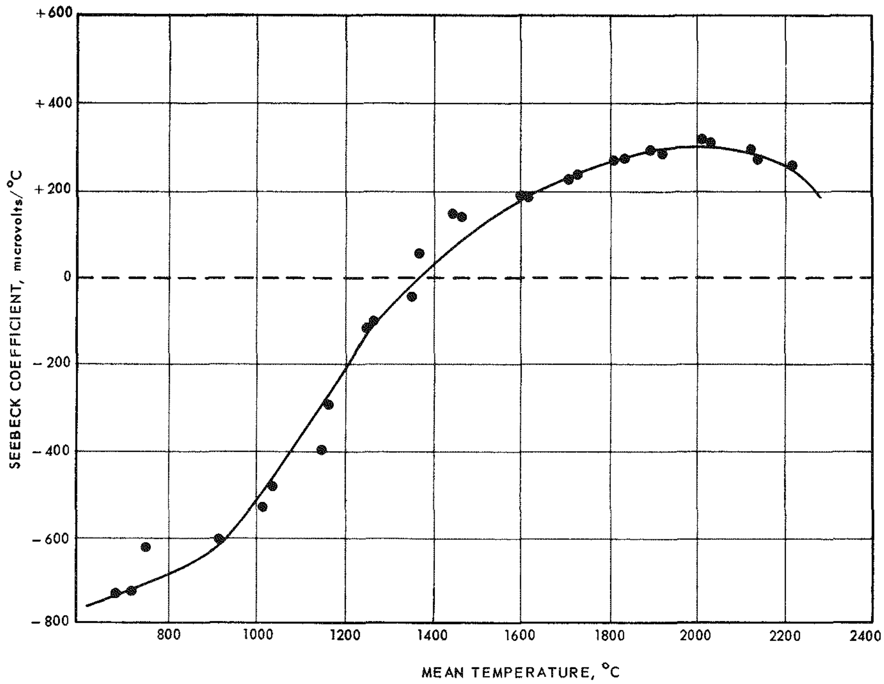

Fig. 3 — Seebeck coefficient of a hydrogen-sintered UO_2 specimen (GE-NMPO) as a function of temperature

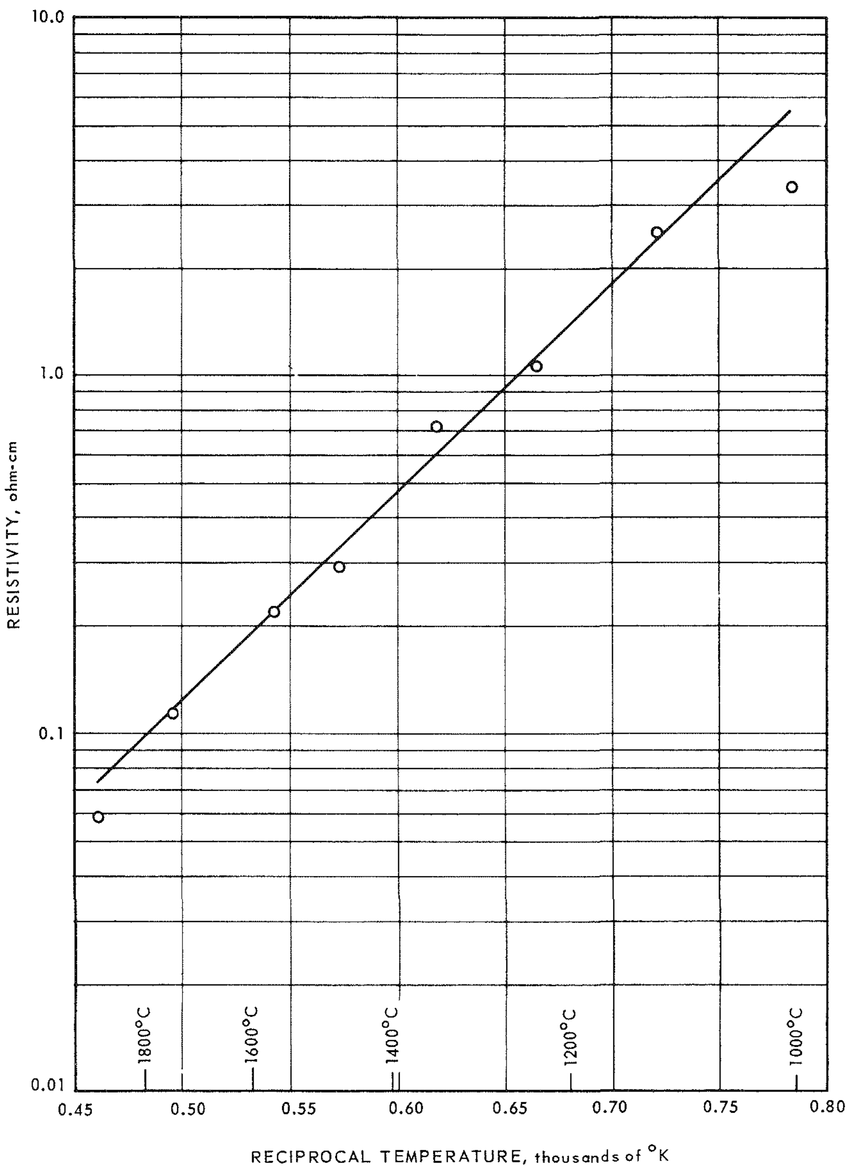

Fig. 4 – Temperature dependence of the electrical resistivity of a hydrogen-sintered UO_2 specimen

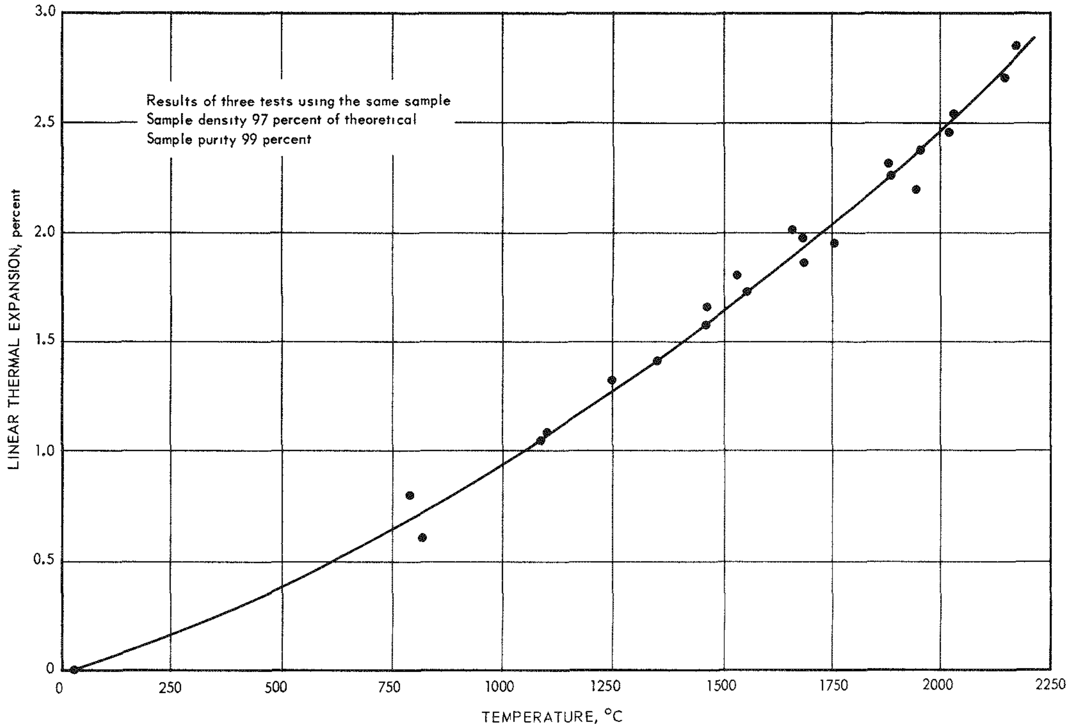

Fig. 5 – Thermal expansion of UO_2

4.2.3 Critical Experiment

The SMR facility will be used for the critical experiment for the 710 reactor development.

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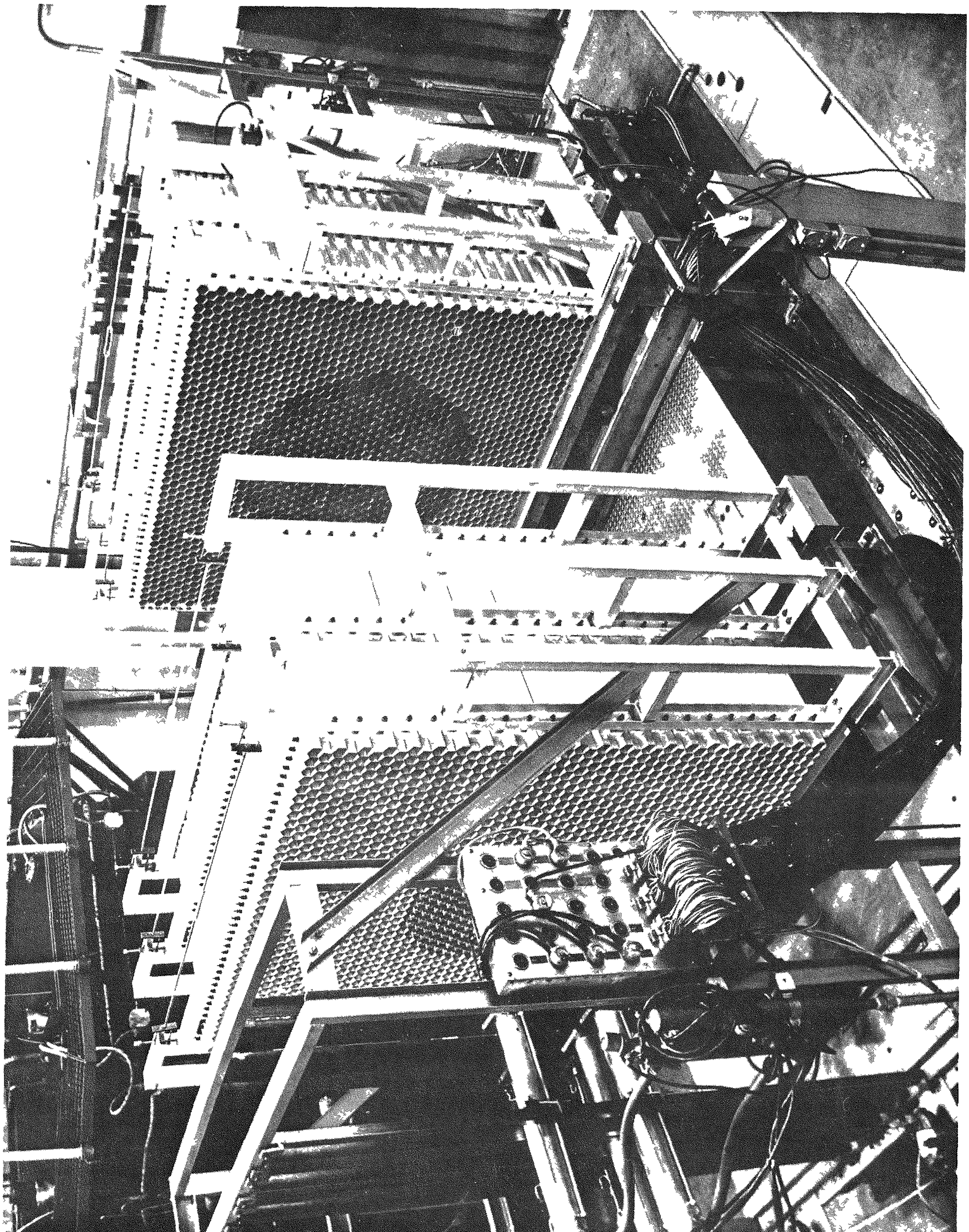

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The purpose of initial experimentation is to provide basic information to establish the validity of analytical environment which is a close simulation of that expected in the 710 reactor. It also must be analytically tractable; i.e., the experiment must be translatable with few compromises to the mathematical model used in analyses. The program, therefore, evolves as shown on the chart.

1. Information to be obtained will be the critical mass of the reactor and the reactivity of a system with a given inventory of uranium 235.
2. Evaluation of absorber materials in the reflector. This phase of the experiment is to provide meaningful information upon which the control system evaluation can be made.
3. Evaluation of material danger coefficients. From these data, predictions of the neutron spectrum and variations of the spectrum within the reactor can be confirmed.

4. & 5. Provide an indication of the fission rate throughout the reactor plus the deposition of prompt and decay gamma energy in various parts of the reactor system. These data can be directly compared with analytically predicted quantities and consequently will provide extremely clean cross-section and spectral correlation.

6. These techniques will be used to provide an evaluation of the ratio of the effective delayed neutron fraction to the neutron lifetime. These data are of prime importance for a kinetic analysis of the 710 reactor as well as a confirmation of the multi-group analysis techniques.
7. To provide preliminary information on the effects of the addition of hydrogenous material in and around the reactor.
8. This information will be used to evaluate the change in spectrum near the core reflector interface caused by the presence of moderating material in the reflector. It will also provide information as to the magnitude of the fission power peak to be expected at the interface.

CRITICAL EXPERIMENT

1. Critical mass

2. Control worth

3. Material danger coefficients

4. Secondary heating

5. Fission power

6. Rossi-α, Noise Analysis

7. Effects of gross hydrogen addition

8. Moderating reflector

4.2.4 Control and Stability Analysis Discussion

The control and stability analysis work that will be performed in Fiscal Year 1963 will be used to provide information and reactor design guidelines, as early as possible, of the effects of structural arrangement on the stability and temperature coefficient of the reactor. The primary objective of this analysis is to evaluate the interaction of components within the reactor on the temperature coefficient and the time constants of the temperature coefficient. With understanding early in the design effort, it would, therefore, be possible to provide control over the temperature coefficient and its time constants rather than accepting that which results from a design involving no consideration of the temperature coefficient.

The expected sequence for this work will follow that shown on the chart.

The first step will be to establish a reference arrangement. This reference arrangement is not a reactor design; it is merely an arrangement of material to provide a reasonable approximation to a reactor configuration which could eventually be built and tested.

The second step is that given this reference arrangement; determine the temperature levels as a function of operating history and operating power level of the fuel element, forward support, reflector material, and the pressure shell. These items constitute the major factors which presently appear to affect the temperature coefficient of the 710 reactor.

Third step will be to determine the reactivity as a function of temperature of the above items as they expand because of thermal expansion and any effects such as Doppler broadening of the resonances in the material to be used.

Fourth step will be to determine the time constants for each of the above effects and then provide a kinetic relationship among these items which can then be used basically as a transfer function on the reactor system for three operating modes. These modes are: startup, steady-state operation, and power modulation and shutdown.

Fifth step - With the above established and known plus the determination of corrective measures available and their efficiency, it will then be possible to establish the requirements for the control system. This is not a control system design. It is the basis from which a control system design would evolve.

CONTROL AND STABILITY ANALYSIS

1. Establish reference arrangement

2. Determine temperature levels of:

   a. fuel element

   b. forward support

   c. reflector

   d. pressure shell

3. Determine Δk as f(T) of above forex expansion and Doppler broadening

4. Determine kinetic relations for 3 operating modes

5. Establish control system requirements

4.3 TEST REACTOR DESIGN AND DEVELOPMENT

Preliminary design of the test reactor is planned to begin July, 1963, and continue for 10 months. Final design can be complete by January, 1965. Fabrication of the test reactor with suitable spares can be complete July, 1965. Design effort is to include:

1. Performance maps
2. Safety evaluation
3. Fabrication drawings
4. Aftercooling requirements
5. Handling requirements
6. Test installation and maintenance instructions
7. Test limits and procedures.

Development effort during the same period will include tests to establish the following information:

1. Complex flow passages
2. Dynamic and static loads on intricate structures
3. Thermal stresses during transient conditions
4. Friction and wear at high temperatures.

Design and development of test and handling equipment will include:

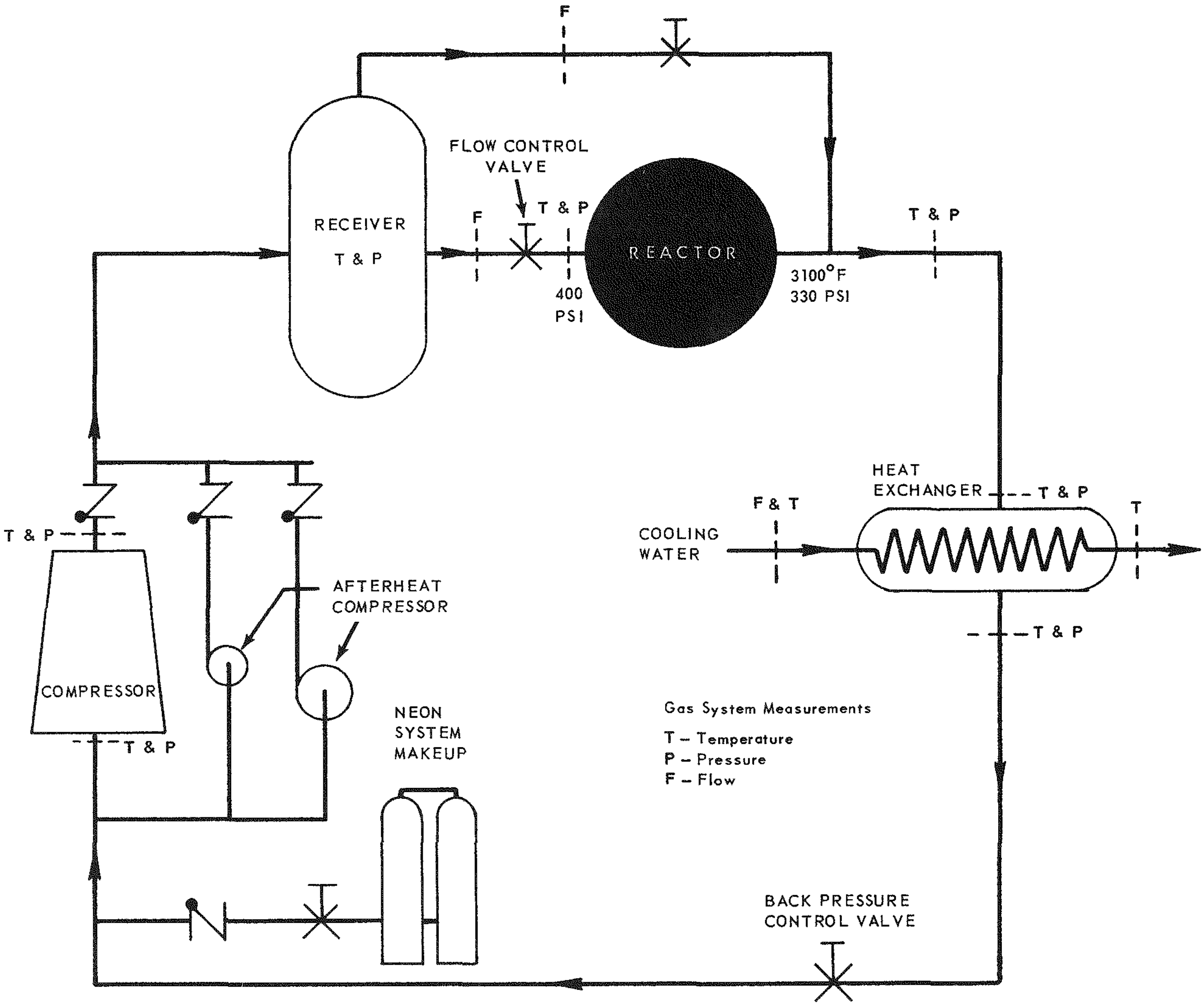
1. Hydrogen and neon test dolly
2. Storage tanks and piping
3. Neon heat exchanger
4. Handling equipment, tools, and fixtures.

4.4 REACTOR TEST PROGRAM

4.4.1 Neon Closed-Loop Test

The purpose of the test loop shown in the chart is to prove the feasibility of the reactor over the whole range of closed cycle operation.

In this system the neon is pumped by a motor-driven centrifugal compressor into a receiver maintained at 400 psi. From the receiver, it flows through the reactor core where it is heated. The hot gas is then cooled in a neon-water heat exchanger before being returned to compressor suction. For afterheat removal, a smaller compressor forces circulation until the heat-generation rate is low enough to be dissipated by convection cooling. The flow rate and reactor back pressure are controlled by remotely operated valves before and after the reactor. Gas from the receiver is bypassed around the core into the annulus to cool the outer pressure pipe. Small ports in the thin inner wall permit the gas to enter and cool the reactor effluent stream. The boundary layer of cool gas maintains the temperature of the liner at a safe level.

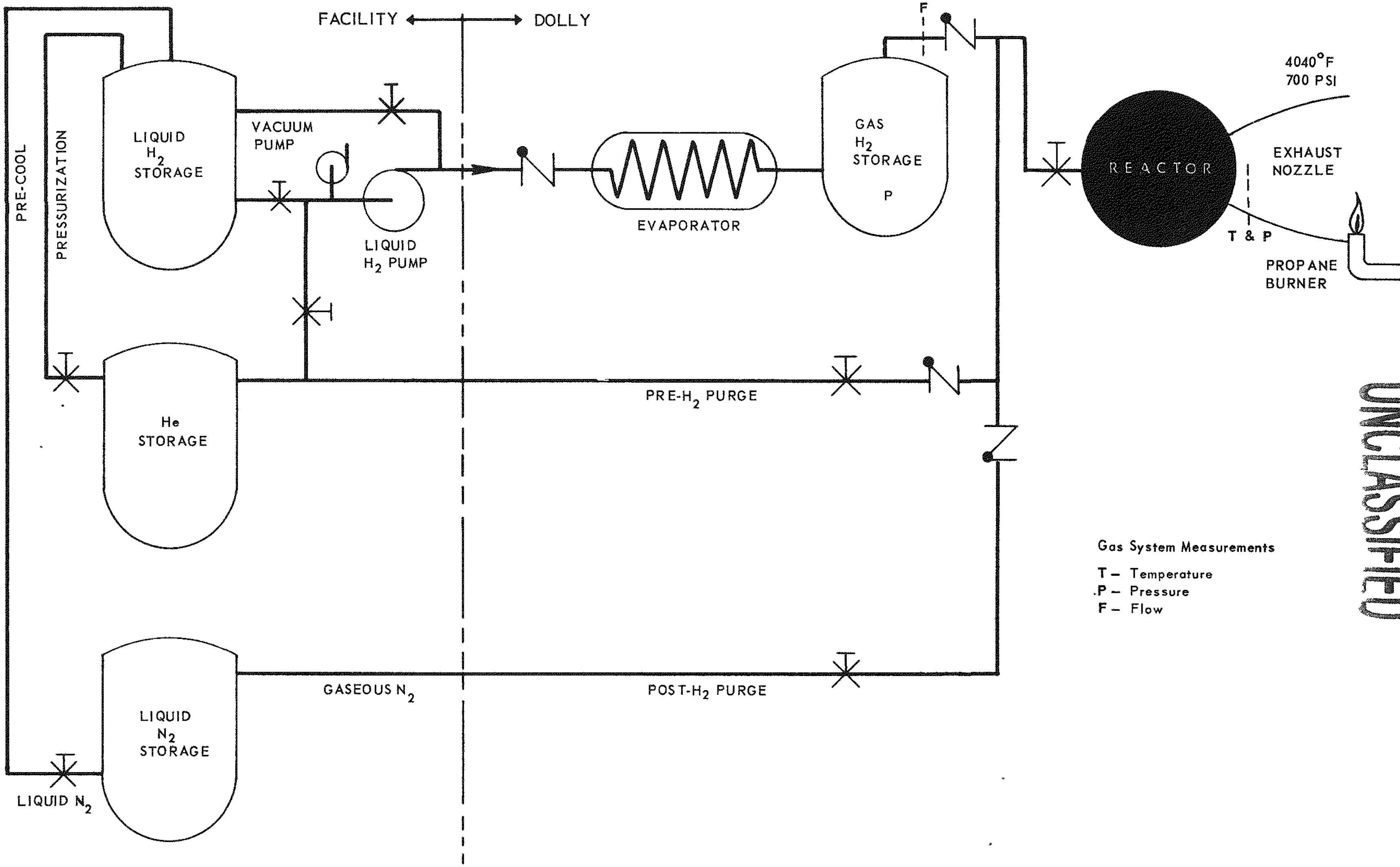


–Neon closed loop

4.4.2 Hydrogen Open-Loop Test

The open-loop test will be used to test the power generating characteristics of the reactor.

The hydrogen test loop is shown in the chart. Liquid hydrogen will be pumped at the desired rate of flow, 11.8 pounds per second at full load, by a liquid hydrogen pump. The liquid hydrogen enters the heat exchanger where it evaporates into a gas. The gas enters a small receiver and passes through a control valve into the reactor. At full power, 11.8 pounds per sedond of gas will flow through the reactor core where it will be heated to about 4040°F, requiring a reactor power of 220 megawatts.



– Hydrogen open loop

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