REFERENCE ZIRCONIUM
HYDRIDE REACTOR

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

~~This report may not be published without the approval of the Patent Branch, AEC.~~

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

REFERENCE ZIRCONIUM
HYDRIDE REACTOR

By

L. M. MAKI
E. J. DONOVAN, JR.

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

CONTRACT: AT(04-3)-701
ISSUED: JUNE 15, 1969

CONTENTS

	Page
Abstract	7
I. Introduction	8
II. Summary	9
III. Reactor System Design and Performance	16
A. Requirements	16
B. System Design Description	17
1. Overall Reactor Vessel	17
2. Fuel	23
C. Reactor System Performance Requirements	25
1. Core	25
2. Control System	29
3. Reactor Lifetime	31
IV. Reactor System Design Description	37
A. Introduction	37
B. Fuel Design	37
1. Requirements	37
2. Design	38
3. Technology Status	38
C. Reactor Vessel Design	39
1. Requirements	39
2. Design	40
D. Control Component Design	42
1. Control Drum	42
2. Control-Drum Bearings	46
3. Actuators	51
4. Control-Drum Drive System	52
V. Shield Design and Performance	54
A. Introduction	54
B. Typical Requirements	54
C. Description of Typical 4π Shields	55

CONTENTS

	Page
D. Performance	59
E. Shield Materials Technology Status	62
1. Gamma-Shield Materials	62
2. Combined Neutron-Gamma Shield Materials	63
3. Neutron-Shielding Materials.	65
References	66

TABLES

II-1. Reference ZrH Reactor Design Parameters	13
III-1. ZrH Reactor Reactivity Requirements, No Burnable Poison	25
III-2. ZrH Reactor Initial Excess Reactivity with \$10.00 Gd-155 Loading	25
IV-1. Reactor Weight Breakdown.	42
IV-2. Friction and Thermal-Cycle Test Data.	49
V-1. Typical Dose Levels for Manned Space Applications	55
V-2. Shield Hydrogen Loss Through Meteoroid Punctures After 20,000 Hours in Orbit	61
V-3. SNAP Shield Development	64

FIGURES

	Page
II-1. Reference ZrH Reactor Operating Envelope	10
II-2. ZrH Reactor Reference Design	10
II-3. Reactor-Design Comparison, Reference ZrH vs S8DR	12
III-1. Reference ZrH Reactor Design	
a. Side View	19
b. Core at Upper Gridplate	19
c. Core at Lower Gridplate	19
III-2. Reference Fuel Element	24
III-3. Maximum Fuel Temperature Isotherms (600 kwt).	28
III-4. Volume Expansion of BeO (2.9 gm/cc, 0.5% MgO, 20-micron grain size)	29
III-5. Expansion of BeO	30
III-6. Maximum Reflector Temperatures	32
III-7. (Page deleted)	34
III-8. Reactor Excess Reactivity, $T_{IN} = 1100^{\circ}\text{F.}$	36
III-9. Reactor Excess Reactivity, $T_{IN} = 1000^{\circ}\text{F.}$	36
IV-1. Lower Vessel Closure Assembly	41
IV-2. Reference Design Control Drum	44
IV-3. Control-Drum Temperatures	45
IV-4. Development Test Bearing Assembly, Pretest	48
IV-5. Development Test Bearing Assembly, Post-Test	48
IV-6. Upper Control-Drum Bearing	50
IV-7. Lower Control-Drum Bearing	50
IV-8. S8DR Actuator	52
IV-9. Reference Design Actuator Installation	53
V-1. Radiation Tolerance Limits for Typical Components	54
V-2. Reference Design 4π Shield	
a. Configuration	56
b. With Structure	56
V-3. Preliminary Shield Temperatures, 600 kwt.	58

ABSTRACT

This study was performed under contract to the Atomic Energy Commission to establish a conceptual design of a reference zirconium hydride (ZrH) nuclear reactor suitable for relatively high power, long duration space applications. The design is based on the technology of the zirconium hydride moderated, U-235 fueled, NaK-cooled reactor type as typified by the SNAP 8 reactor. The design evolved has a minimum performance capability of 600 kwt for 20,000 hours at 1300°F, is suitable for use with a number of candidate power conversion systems, and can be operated with either a shadow shield or within a fully enclosed (4π) radiation shield.

I. INTRODUCTION

The Zirconium Hydride Space Reactor Program has evolved along a route of ever-increasing power-temperature-lifetime capabilities. The SNAP 10A reactor was designed to provide 40 kw at 1000°F for one year; the interim SNAP 10A/2 reactor was designed for 100 kw, 1200°F, and one year; and the SNAP 8 Developmental Reactor (S8DR) is designed for 600 kw, 1300°F, for 12,000 hr. These reactors were all configured for use with shadow shielding. In the course of various applications studies it became apparent that an increased lifetime capability over that provided by S8DR would be desirable. In addition 4π shielding surrounding the core appears to be necessary for some applications.

In view of the need for increased performance capabilities the AEC authorized the conceptual design of an improved zirconium hydride reactor. The design requirements were that the reactor system is to be:

- 1) Capable of 20,000-hr lifetime at 600 kw and 1300°F;
- 2) Compatible with both shadow and 4π shielding; and
- 3) Based to the maximum extent practical on S8DR.

The study was to be performed in conjunction with the design of a 25-kwe reactor-thermoelectric space power system to assure that the various system integration aspects were considered.^{(1)*}

This report describes the conceptual design of the reference zirconium hydride reactor which resulted from this study.

*Superscript numbers in parentheses are for References at end of this report.

II. SUMMARY

The reference design zirconium hydride reactor is capable of providing 600 kw of thermal power for 26,000 hr with an outlet coolant temperature of 1300°F. The overall performance envelope is shown in Figure II-1, and the reactor design in II-2.

The reactor core contains 295 fuel-moderator elements of the S8DR type.

The fuel material is a zirconium-uranium alloy which has been hydrided to a hydrogen content (N_H) of 6.3×10^{22} atoms/cm³. (The N_H of cold water is about 6.6.) This fuel material is quite stable in the expected temperature and radiation environment, and its performance has been well characterized as a result of the operation of five SNAP reactors plus many in-pile irradiation experiments. The fuel is contained in a Hastelloy-N cladding tube which is coated on the inside with a ceramic material to inhibit the loss of the hydrogen moderator from the fuel rod.

The system, like all preceding zirconium hydride reactors, uses the sodium-potassium eutectic NaK-78 as the coolant. The NaK enters the base of the reactor vessel at 1100°F and flows around "dry wells" which contain the reflector control drums. The coolant flow is turned 180° in direction, passes over the fuel elements, and leaves the reactor vessel at 1300°F. The fuel elements in the core are spaced in a nonuniform hexagonal array to provide each element with a coolant flow approximately proportional to the power generated in the element. The nozzle-to-nozzle pressure drop of the reactor is 0.65 psi with a flowrate of 49,000 lb/hr (this corresponds to 600 kwt with a 200°F ΔT across the core). Type 316 stainless steel is used as a structural material for the reactor vessel and internals.

The major change from previous zirconium hydride reactors lies in the design of the reflector control system. The SNAP 10A and SNAP 8 reactors were designed to operate in a shadow-shielded configuration. The control drums were semicylinders of beryllium, mounted around the core, which could be rotated to increase or decrease the neutron leakage from the system. The reference control system is designed to operate inside a shield which completely surrounds the reactor. Ten cylindrical reflector-poison control

Figure II-1. Reference ZrH Reactor
Operating Envelope

8-AU-13-119-115

Figure II-2. ZrH Reactor Reference Design

AI-AEC-MEMO-12715

drums are mounted in dry wells cooled by the inlet NaK. The control drums will operate at relatively high temperatures (1500 to 1800°F) because the nuclear heat generated in the drums must be transferred by radiation to the walls of the dry well, which are cooled by 1100°F NaK. Beryllium oxide is used as the reflector material in this high-temperature environment. The selected poison material is europium oxide dispersed in nickel, and high-strength niobium alloys are planned for structural use. The overall geometry of the reference system is compared with the S8DR in Figure II-3.

The control drum rotates in a set of ball-and-socket bearings. Like S8DR an aluminum-oxide-coated shaft rotates in a carbon-graphite ball mounted in a socket coated with aluminum oxide. Although the maximum bearing temperature is expected to be below 1350°F, testing of this type of bearing has been conducted at temperatures up to 1500°F. The control drum will be rotated by an S8DR actuator which has been modified slightly for this application.

The shield configuration will be dependent upon the particular mission for which the reactor is to be used. In general the reactor vessel will be surrounded by a gamma shield of molten lead or tungsten. A thin layer of insulation will separate the high-temperature gamma shield from a region of lithium hydride neutron shielding. This type of two-region shield is adequate for applications where the allowable doserates are high or the separation distance from personnel is large (e. g. manned lunar base). In the case of a shaped 4π shield for a manned space station the shielding will be much thicker in the direction of the station. Normally the first two regions will be followed by a gallery containing the power-conversion system or an intermediate NaK-to-NaK heat exchanger. Aft of the gallery will be a second gamma shield of depleted uranium and a second lithium hydride neutron shield.

The lithium hydride shields are cast in stainless-steel containers, strengthened by internal shells and stringers. The shield assembly is normally the heaviest unit in the reactor system and provides the primary structure for the entire system. The weight of the reactor, power-conversion system, and shield is carried through the shield structure to a load ring which is mated to the vehicle.

Some of the more significant design parameters for the system are given in Table II-1. In general the design involves very few deviations from the

10-16-68 UNCL

7759-5202

Figure II-3. Reactor-Design Comparison, Reference ZrH vs S8DR

TABLE II-1
REFERENCE ZrH REACTOR DESIGN PARAMETERS
(Sheet 1 of 2)

<u>General Performance</u>		
Design thermal power		600 kw
Outlet temperature		1300°F
Temperature rise		200°F
Design lifetime		26,000 hr
Coolant flowrate		49,000 lb/hr
Pressure drop		0.65 psi
Operating pressure		35 psia
Peak fuel temperature		1420°F
Peak cladding temperature		1320°F
Short-term reactivity losses		- \$2.16
Long-term reactivity losses		- \$11.59
Burnable poison reactivity gain		+ \$10.00
Reactivity margin		- \$1.50
Required initial excess reactivity		\$5.25
Peak control-drum BeO temperature		1500°F
Peak control-drum poison temperature		1570°F
Peak control-drum cladding temperature		1510°F
Control-drum bearing temperature		1270°F
Control-drum actuator temperature		950°F
<u>Fuel and Core Internals</u>		
Number of fuel elements		295
Type of fuel elements		Modified S8DR
Hydrogen content, N _H		6.3×10^{22} atoms/cm ³
Wt % uranium		10.5
Hydrogen barrier material		SCB-1
Fuel-clad gap		Optimized
Cladding material (reference)		Hastelloy-N
(alternate)		Incoloy-800
Cladding thickness		0.015 in.
Burnable poison material		Gd-155
Active fuel length		16.8 in.
Overall (flat to flat) element length		17.5 in.
Outside diameter of fuel element		0.570 in.
Shock and vibration suppressor		To be determined
Fuel-element spacing		Variable for flow control
(maximum)		0.045 in.
(nominal)		0.030 in.
(minimum)		0.025 in.
Internal reflector material		BeO
Internal reflector cladding		316 stainless

TABLE II-1
REFERENCE ZrH REACTOR DESIGN PARAMETERS
(Sheet 2 of 2)

<u>Fuel and Core Internals (Continued)</u>	
Gridplate material	316 stainless
Core shell ID	11.4 in.
Core shell thickness	0.030 in.
Core shell material	316 stainless
Reactor coolant material	NaK-78
<u>Reactor Vessel</u>	
Outside diameter of vessel	22.0 in.
Vessel wall thickness	3/16 in.
Inside diameter of dry wells	4.62 in.
Dry-well wall thickness	0.065 in.
Overall vessel height	29.3 in.
Number of inlet and outlet nozzles	4 each
Nozzle size	1.5 in.
<u>Reflector-Control System</u>	
Control-drum type	Reflector-poison
Number of control drums	10
Reflector material	BeO
Poison material	60% Eu_2O_3 in Ni
Candidate structure material	Nb-10W
Candidate cladding material	Nb-1 Zr
Drum diameter	4.5 in.
Drum length	19.0 in.
Structural strongback thickness	0.25 in.
Cladding thickness	0.030 in.
Drum bearing-shaft material	Ta-10W
Shaft diameter	0.75 in.
Shaft plating	Al_2O_3
Bearing ball material	Carbon-graphite
Ball diameter	1.375 in.
Bearing socket material	Ta-10W
Socket plating	Al_2O_3
Control-drum actuator type	Modified S8DR
<u>Shield</u>	
High-temperature gamma shield	Molten lead
Lead container material	Croloy 2-1/4
Low-temperature gamma shield	Uranium
Neutron-shield material	LiH
LiH container material	316 stainless

state-of-the-art, as exemplified by the SNAP 8 Developmental Reactor. In those areas where conditions are more severe than those for S8DR, development programs are underway to establish the required capabilities. For example, high-density experimental Eu_2O_3 -Ni dispersions have been fabricated, prototype bearings have been tested at 1500°F for 1000 hr, experimental shock and vibration suppressors for the fuel have been fabricated, and molten lead specimens have been tested for 10,000 hr at 1500°F. The engineering design and development of the reference ZrH reactor will now proceed from the conceptual design phase into detailed preliminary design.

III. REACTOR SYSTEM DESIGN AND PERFORMANCE

A. REQUIREMENTS

This section presents the overall reactor system design and performance requirements. These overall requirements become more detailed as they pertain to specific components and subsystems and are discussed in later sections of this report.

The reactor system will be based to the maximum practical extent on S8DR and other state-of-the-art SNAP technology. In particular the basic S8DR fuel element will be used except that certain modifications necessary for extending the fuel lifetime are allowable. As described in Section III-B, the major modifications consist of increasing the fuel hydrogen content from an N_H of 6.05 to 6.3, the cladding thickness from 0.011 to 0.015 in. and internal clearances between fuel and cladding, and adding a burnable poison on the surface of the fuel or in a foil in the radial-gap region. A final modification may be the substitution of Incoloy-800 cladding for Hastelloy-N, which was chosen for the current reference design.

The reactor system will have a minimum lifetime capability of 20,000 hr at 600 kwt, with 1100°F core-inlet and 1300°F core-outlet temperatures. Under these operating conditions the system reliability objective will be 0.995 with no credit for possible maintenance. The reactor will also be capable of operation under alternate conditions of power, temperature, and lifetime. These conditions are defined by the operating envelope shown in Figure II-1. Within the envelope, reactor operation will not be constrained by the thermal or endurance limits of any reactor system components. The upper extreme of the operating envelope is the current estimate of the maximum allowable control-drum temperature.

At any given power level and temperature the fuel-element design has been optimized to maximize the reactor lifetime. The limit curves of Figure II-1 represent the initial onset of a potential failure mode. In general the limit is considered to be the simultaneous occurrence of β -phase fuel in the worst-case (0.5%) element and a 0.5% probability of sufficient fuel growth to crack the cladding.

The structural design of the reactor system will be based on the most stringent shock and vibration requirements of either a Saturn 1B, Saturn V, Titan 3C, or Titan 3F launch vehicle. During reactor operation in orbit the system will be capable of withstanding the acceleration loads during normal attitude-change maneuvers and vehicle docking. Since the exact requirements for these loads are mission-dependent, for the current design they will be established as ± 1 -g axial and $\pm 1/4$ -g radial acceleration and 3-g shock. The reactor will be capable of operating in a variety of environments ranging from ground test facilities in 10^{-5} torr vacuum to deep-space vacuum. The reactor system will operate in either a shadow or 4π shield environment. Supplemental cooling of the 4π shield to maintain acceptable shield temperature or provide a heat sink for the reactor control drums will be allowed, but only if designs employing inherent shield and control-drum cooling are determined to be impractical.

The reactor system will be capable of automatic startup and unattended operation following the receipt of a single startup command signal. It will be capable of 50 startup and shutdown cycles in which the maximum rate of change of coolant temperature will be $150^{\circ}\text{F}/\text{min}$ from room temperature to 400°F , and $60^{\circ}\text{F}/\text{min}$ to 1300°F .

The inlet-to-outlet nozzle pressure drop through the reactor will be as low as practicable and not exceed 1.0 psi at 49,000 lb/hr NaK flow (600 kwt, $200^{\circ}\text{F} \Delta T$). All pressure-containing members will be designed to conform to the ASME Pressure Vessel Code, Sections III and VIII. The design pressure for the system will be 35 psia.

B. SYSTEM DESIGN DESCRIPTION

This section provides a summary design description of the reference reactor system including fuel elements and integration with a typical 4π shield. Detailed design and performance descriptions of system components are given in later sections.

1. Overall Reactor Vessel

The reference design of the reactor vessel is shown in Figure III-1. In this design the cooling for the shield in a 4π application and for the control

a. Side View

b. Core at Upper Gridplate

c. Core at Lower Gridplate

Figure III-1. Reference ZrH Reactor Design

drums is provided by radiation to surfaces cooled by the inlet NaK. This is done by locating the control drums in dry wells surrounding the core and causing the inlet NaK to flow upward through the cusp-shaped regions outside the dry wells and between the outer pressure-vessel and inner core-can. The NaK enters the vessel through multiple inlet nozzles in the vicinity of the lower control-drum bearings, passes through openings provided in the lower dry-well support ring, and continues upward to the top of the vessel through the cusp-shaped regions described above. The flow then passes to the core inlet-plenum through the fuel-element array and exits through multiple outlets in the core outlet-plenum. The inlet NaK stream is separated from the NaK in the core by means of a 0.030-in. -thick core-can. The core-can is welded directly to the vessel lower structure. The upper end of the core-can is free to move relative to the core upper gridplate to accommodate differential thermal expansion. As the core-can separates the inlet from the outlet NaK a thermal shunt of a fraction of the core power occurs. The quantity of heat transferred was determined to be 6% of the core power under the base point operating conditions. The net result of this effect is to increase the temperature of the NaK entering the core relative to the reactor system NaK-inlet temperature. This effect, plus the heat resulting from control-drum cooling, results in an increase in NaK-inlet temperature of only 18°F. The penalty is manifested in an increase in fuel temperature of 6°F to attain the 1300°F mixed-mean core-outlet temperature.

To meet the power and lifetime requirements a core consisting of 295 fuel elements of the S8DR type was selected.⁽²⁾ The core reactivity is controlled both by Gd-155 burnable poison selectively loaded in the central region of the core and ten poison-backed beryllium oxide control drums. The reference poison material for the drums is 60 wt % natural europium oxide dispersed in a nickel matrix.

The control-drum bearings consist of a tantalum-10% tungsten inner shaft and outer socket rotating against a P5 graphite ball. The shaft and socket are flame-spray-coated with approximately a 0.003 in. thickness of aluminum oxide where they contact the graphite ball. Thrust loads are carried by the upper bearing. This basic bearing type has been used successfully on various SNAP reactors including S8DR with the major exception that the metal parts

were previously Inconel. The tantalum-10% tungsten is a recent improvement, currently under test, which gives superior performance due to a better match of thermal-expansion coefficients.

Cooling is provided in the lower bearing by spring-loading the bearing socket to maintain thermal contact with the NaK-cooled bearing housing. Thermal contact in the upper bearing is provided by a combination of spring-loading the bearing socket into the threaded stainless-steel housing and a jam nut which provides positive loading of the housing into the threaded dry well. The bearing sockets are spring-loaded with Rene-41 Belleville washers bearing against snap rings.

To provide the desired low pressure drop for the core, a variable pitch fuel-element array is used. Previous SNAP 8 reactors used orifice-type devices in the core inlet-plenum to provide the required matching of coolant mass-flow velocity-to-core-power distribution. In this ZrH reactor the flow areas between elements are adjusted to obtain the required matching. The preliminary spacing ranges from 0.045 in. in the center of the core to 0.025 at the periphery.

Neutron-reflector elements are used both in the irregular area defined by the outermost row of fuel elements and core-can, and in the cusp regions between control-drum dry wells. The purpose of the reflector elements is to increase the reactor excess reactivity and hence the allowable poison loading. This in turn minimizes the reactivity-change requirement of the control drums.

As in S8DR the core internal reflectors are D-shaped BeO rods canned in austenitic stainless steel. At locations where the BeO rod dimensions become unacceptably small, solid stainless-steel rods are used. The fixed reflector elements in the flow area between dry wells are of two sizes dictated by the dimensions of the inner and outer cusp regions. These reflector elements are BeO clad in stainless steel for the reference design. Unclad beryllium metal is a backup design approach.

All NaK-wetted materials with the exception of the fuel elements are fabricated from Type 316 stainless steel. The conceptual configuration and sizing of the pressure vessel were based on preliminary detailed stress analyses using the AVCO computer code. Allowable-stress values were based on the ASME Pressure Vessel code, Sections III and VIII.

Both steady-state and startup transient stresses were considered. With the exception of the core outlet-plenum all pressure members are maintained at the NaK-inlet temperature. The outlet-plenum operates at the core-outlet temperature. Differential expansion of the outlet-plenum relative to the remainder of the vessel is accommodated by the inherent flexibility of the design configuration. The structural studies also showed that reduced stresses result when the control-drum dry-well structure is solidly attached to the lower vessel structure rather than freely suspended from the upper head. Hence the solid attachment approach was selected.

In the design of the vessel, x-ray-inspectable butt-welded joints were used exclusively with the exception of the final closure weld. This closure is made at the core outlet-plenum after the fuel elements and lower gridplate are installed. Although the problem of x-ray film and source placement makes x-raying of this joint of limited value, dye-penetrant and ultrasonic methods may be used.

The weight of the reference ZrH reactor as shown in Figure II-2 with the addition of control-drum actuators is 1420 lb. The volume of NaK contained within the vessel is 1.75 ft³.

2. Fuel

The reference fuel element design is shown in Figure III-2. A more detailed description is given in Section IV-B. The basic features consist of the zirconium hydride fuel pin containing 10.5 wt % uranium, a 0.015-in. -thick Hastelloy-N cladding tube coated on the inside with a ceramic hydrogen barrier, and a gadolinium oxide burnable poison vapor deposited on the surface of the fuel rod. The hydrogen density of the fuel rod is 6.3×10^{22} atoms/cc (i.e. $N_H = 6.3$).

The design represents an evolution from the S8ER and S8DR designs. The significant differences between the reference design and the element currently beginning long-term testing in S8DR is an increase in N_H from 6.05 to 6.3 and in cladding thickness from 0.011 to 0.015 in., the use of Gd_2O_3 poison on the fuel rather than Sm_2O_3 in the ceramic, and increased radial and axial clearances. A device to dampen the effects of launch shock and vibration will also be included in the fuel element.

Figure III-2. Reference Fuel Element

The 40% additional fuel elements in the reference reactor result in a lower core-power density than in S8DR for comparable power levels. This results in lower fuel-element temperatures and is the principal method used to extend the fuel life from the S8DR design value of 12,000 hr to the present one in excess of 20,000 (both at 600 kwt, 1100°F inlet, 1300°F outlet). The predominant life-limiting factor is hydrogen loss from the fuel element. This factor represents the major component of core-reactivity loss. Excessive hydrogen loss can also result in a fuel material transformation to the beta phase, which has been associated with accelerated fuel growth.

The hydrogen loss rate is governed by the hydrogen pressure in the gas annulus in the fuel element and by the properties of the ceramic permeation barrier both of which are strongly temperature-dependent. The average fuel temperature in the reference reactor is 1300°F at the nominal operating point as compared with 1330°F for the S8DR at the same operating conditions. This relatively small temperature difference would result in about a 50% change in the hydrogen-dissociation pressure for fuel with an N_H of 6.3.

C. REACTOR SYSTEM PERFORMANCE CHARACTERISTICS

1. Core

a. Nuclear Performance

The core reactivity requirements may be defined as short term (less than three days after startup) and long term. The requirements for the base-point operating conditions before burnable poison addition are summarized in Table III-1.

TABLE III-1
ZrH REACTOR REACTIVITY REQUIREMENTS,
NO BURNABLE POISON

Short-Term: Temperature and power defect, xenon, hydrogen redistribution	\$ 4.21
Long-Term: Fuel depletion, hydrogen loss, fission products	11.59
Estimated Stuck-Control-Drum Worth	1.50
	<u>\$17.30</u>

To minimize the reactivity-control requirements a \$10.00 (hot) loading of Gd-155 poisoning is used to suppress the initial core excess reactivity. The poison burns out during reactor operation and provides compensation for the long-term losses. Essentially no poison remains at the end of the core life. The poison also has a positive reactivity contribution on system heatup. This is due to the change in neutron spectrum with temperature. The positive reactivity from this source is calculated to be \$2.05 leaving a net short-term (startup) loss of -\$2.16. The reactivity requirements for the poisoned core may then be summarized as in Table III-2.

TABLE III-2
ZrH REACTOR INITIAL EXCESS REACTIVITY WITH
\$10.00 Gd-155 LOADING

Total Losses, Unpoisoned Core	- \$17.30
Gd-155 Burnout (Hot)	+ 10.00
Gd-155 Worth, Cold to Hot	+ 2.05
Total	- \$ 5.25
Net Initial Excess	<u>+ \$ 5.25</u>

Thus the reference reactor has a requirement for an initial poisoned excess reactivity of + \$5.25. The reactivity-control requirements for the reactor are discussed in Section III-C-2.

Under power, lifetime, or thermal conditions other than the base point the initial required and total-available excess reactivities will vary to some extent. The net short-term loss is reduced by a reduction in core power level, operating temperature, or core ΔT . Of the long-term losses the hydrogen-loss component is the most significant and has the greatest calculational uncertainty. Because it is related to the hydrogen pressure, fuel-cladding temperature, and time it is increased by higher power level or greater core lifetime. The S8DR test which is to be run at two power levels (600 and 1000 kw) under two sets of thermal conditions will provide normalization points for the hydrogen-loss predictions. The long-term losses caused by fuel burnup and fission-product accumulation are obviously directly related to the total energy output of the core. Based on current calculations the reference 295-element core with the ten poison-back drums has sufficient margin to meet all reactivity requirements within the specified operating envelope shown in Figure II-1.

The average neutron-flux level in the reactor core is 1.8×10^{12} nv thermal and 2.2×10^{13} nv fast (>0.1 Mev). The flux spectrum of the zirconium hydride reactor is somewhat more epithermal than is characteristic of water-moderated reactors. This is particularly significant with regard to the core temperature coefficient of reactivity and the temperature variation of the Gd-155 poison worth.

The overall peak-to-average-power generation in the reactor core is 1.7. The variation of power across an individual fuel element diameter, however, is negligible.

b. Thermal and Hydraulic Performance

The NaK-78 eutectic has been the coolant used in all ZrH reactors designed and built to date. It is used in preference to sodium because its low melting point of about 10°F , compared to 208 for sodium, greatly reduces the problem of maintaining the system in a molten state before startup. The vapor pressure of NaK at 1300°F is 6.4 psi, which makes it compatible with temperature limitations on the ZrH fuel element and externally applied pressure limits on fuel-element cladding.

The distribution of the NaK flow within the core is a significant factor in obtaining maximum performance. Poor distribution would result in the core reaching fuel-element lifetime limitations prematurely. As discussed later in this section, maximum core lifetime can be shown to be attained when the peak temperatures within all fuel elements approach the same value near the end of core life. The required flow distribution is obtained by varying the spacing between fuel elements. Larger spacing is provided in the center of the core to provide a greater mass flow to compensate for the higher power density in the region. The preliminary minimum spacing between central elements is 0.045 in. and is varied to 0.025 in. at the edge of the core. The S8ER and S8DR reactors used a flow-baffle device in the inlet plenums to provide a velocity distribution across the core which achieved about the same result. This method, however, requires a relatively high pressure drop in comparison to the variable-spacing approach. As an example the S8DR nozzle-to-nozzle pressure drop is 4.6 psi compared to 0.65 psi for the reference reactor. The 0.65-psi value allows some margin below the 1.0 specified limit in the event that some flow-baffle or pressure-equalizing device is found to be needed to provide a uniform pressure distribution at the core inlet.

The fuel-pin temperatures within the core are controlled by the local coolant temperature, local power density, temperature "jumps" that occur across the NaK/cladding boundary, cladding wall, gas gap, and the radial temperature profile within the fuel pin. The average difference between the fuel centerline temperature and the local coolant temperature is 150°F, and the nominal peak value is 250°F.

The fuel temperatures in the core vary with lifetime as a result of shifts in the power density and changing conductance across the fuel gas gap. The principal factor in the power shift is the change in position of the control drums. The radial and axial variation of centerline fuel temperatures at beginning and end of life are shown in Figure III-3 for operation at 600 kwt and a 1300°F outlet temperature. The maximum fuel temperature is 1420°F and occurs at the beginning of life.

A consideration in the core thermal performance is the heat flow which occurs through the core-can to the NaK flowing in the control-drum dry-well region. Calculations were made indicating a loss of 35 kwt under core 200°F ΔT

Figure III-3. Maximum Fuel Temperature Isotherms (600 kwt)

conditions. The result of this loss is an increase in the core-inlet temperature and an increase in fuel temperature to attain the required value of mean core-outlet temperature. Of these the increase in fuel temperature is the more significant because of the potential reduction in core life. Calculations show, however, that the maximum fuel temperature is increased only 6°F over the hypothetical case of a perfectly insulated core-can.

2. Control System

The ten control drums provide the \$2.16 reactivity control for reactor startup, \$3.09 for long-term losses, and a margin for shutdown at any point in the reactor life. The control drums have sufficient margin to permit normal operation if a drum should become inoperative in either the most or least reactive position.

The temperature of the BeO in the control drums places an operating limit on reactor power level. This is a result of a temperature-dependent growth which occurs in BeO under neutron irradiation. BeO growth has been studied by General Electric Corporation^(3,4,5) and others. Typical results are shown in Figure III-4. The growth is characterized by two modes: a low-temperature

Figure III-4. Volume Expansion of BeO (2.9 gm/cc, 0.5% MgO, 20-micron grain size)

AI-AEC-MEMO-12715

swelling caused by microcracking, and at higher temperatures a helium-gas-induced expansion. A composite of these two modes is shown in Figure III-5. The clearances between the BeO and drum canning are established to allow for 2 vol % growth. With an 1800°F temperature limit this should be quite conservative.

Figure III-5. Expansion of BeO

Temperatures within the drums are governed by heat-generation rates, drum-material conductance, and radiative heat transfer from the drum to the NaK-cooled dry well. The maximum heat generation occurs at the core midplane; hence this is the location of maximum temperature. A plot of maximum midplane drum temperature as a function of reactor power level is shown in Figure III-6. Also shown on the figure are maximum temperatures within the Eu_2O_3 in nickel-poison material. An upper limit on the poison-material temperatures has not been established but it is assumed to be sufficiently above the BeO limit so as not to be power limiting.

The control-drum bearings are also heated by internal heat generation. The major heating occurs in the dense Ta-10W bearing shaft and socket. Temperatures are minimized by contact pressure between mating parts and the dry-well heat sink. Under 1200-kwt and 1100°F NaK-coolant temperature conditions the maximum bearing temperature is approximately 1310°F. (An experimental bearing of the same basic design has been tested at 1500°F.)

3. Reactor Lifetime

The SNAP core lifetime capability is defined as the point at which a systematic fuel-failure mode becomes imminent or the core has insufficient excess reactivity to maintain rated conditions. Systematic fuel failure is assumed to be imminent when irradiation fuel growth has proceeded to the point of allowable cladding strain. In practice, statistical methods are used to account for the experimental scatter of fuel-growth and cladding strain-to-failure data. A 0.5% fuel-failure limit has been defined as the end-of-life point although capability of significant full-power operation lifetime is known to exist far beyond this point.

SNAP fuel-irradiation experience has shown that two types of fuel growth can occur. One is associated with the appearance of beta-phase fuel material in the ZrH fuel rod resulting from loss of hydrogen through the cladding. The presence of β -phase has been observed to cause rapid fuel growth in a very limited number of observations. The consequences of β -phase are not currently well defined in terms of quantitative effect or in-core environment. Conservatively, the first appearance of β -phase is assumed to be synonymous with cladding failure.

10-16-68 UNCL

7759-5207

Figure III-6. Maximum Reflector Temperatures

AI-AEC-MEMO-12715

The second growth mechanism occurs in fuel of normal hydrogen composition. A significant amount of data exists in this regime and a burnup, temperature, and compositional correlation is available.⁽⁶⁾

The fuel-growth model in current use for SNAP fuel-performance analysis is based on numerous data from several inpile irradiation experiments and postoperational examination of S8ER fuel elements. The model includes correlation with temperature, burnup, and fuel hydrogen content and is applicable to the range of hydrogen contents above the β -phase of ZrH. (Sentence deleted.)

The important consideration in the fuel design is actually the statistical spread of the available data. Clearances within the fuel element are established based on consideration of the statistics of both the fuel-growth and cladding strain-to-failure data. A design limit of 0.5 % failure at end-of-life has been established as acceptable for the reference reactor and is used throughout this report.

The other defined limit on fuel-element life is the appearance of β -phase in the fuel. As mentioned previously the available data on β -phase swelling is limited, hence no correlation currently exists. For this reason the first appearance of β -phase is considered the end of design life although it is clear that this is conservative.

In sizing the radial clearance gap in the fuel element to allow for normal fuel growth an optimum point exists. This is because increases in gap dimensions result in higher fuel temperatures and hence in acceleration of the hydrogen loss and β -phase appearance. In lifetime analyses, discussed herein, the gap has been selected in all cases to represent this optimum point where the fuel-growth and β -phase limits are approached simultaneously. (Sentence deleted.)

(This page deleted, including Figure III-7.)

The lifetime calculations are made with the LOAFER digital computer code. The analytical model provides a two-dimensional thermal analysis of the core, accounting for changes in fuel temperatures due to control-drum rotation, changing dimensions in the fuel/cladding radial clearance due to fuel growth and accumulation of low-conductivity fission gas in the gap. The time and spatial variation of hydrogen loss is then computed using the temperature and time-dependent loss model previously discussed. The first appearance of β -phase is determined from phase-diagram relationships programmed into the code. Finally the core-reactivity change due to hydrogen loss, uranium burn-up, and fission-product poisons is calculated. The reactivity/hydrogen relationship and reactivity/fission-product and burnup relationships are entered as input coefficients.

Figure III-8 shows the reactivity change and fuel-life limits in the core at various power levels under the base-point thermal conditions (1100°F inlet, 1300°F outlet). As indicated the core is reactivity-limited if one drum were to be stuck in the least-reactive position. If all drums are operative the core is life-limited by fuel growth and phase change.

Figure III-9 shows the effect of a 100°F reduction in core temperatures on lifetime. Under these conditions the core is reactivity-limited except for exceptionally high power levels. The core life, however, is much longer than for the case of the 1300°F NaK-outlet temperature.

Figure III-8. Reactor Excess Reactivity, $T_{\text{IN}} = 1100^{\circ}\text{F}$

Figure III-9. Reactor Excess Reactivity, $T_{\text{IN}} = 1000^{\circ}\text{F}$

IV. REACTOR SYSTEM DESIGN DESCRIPTION

A. INTRODUCTION

This section gives the design requirements, descriptions, and technology status of the reactor system, to include the reactor and vessel, fuel elements and control-drum and drive system. The overall system requirements were stated in Section III. They are discussed in this section in more detail, to the extent to which they pertain to the design and performance of particular reactor components.

A major factor in the selection of the current reactor concept is the requirement of operation in either a 4π - or shadow-shield configuration. The 4π -shield application creates the problem of providing adequate cooling of the control drums and gamma-shielding material. The concept of containing the control drums within dry wells which penetrate through the reactor vessel pressure envelope has not been used previously in SNAP reactors. It was selected, however, after trade studies indicated its superiority over other possible methods of accomplishing the control-drum and gamma-shielding cooling function.

B. FUEL DESIGN

1. Requirements

The fundamental requirements for the fuel element are to maintain cladding integrity and control the permeation of the hydrogen moderator from the fuel to an acceptable rate for all power levels and lifetimes within the operating envelope shown in Figure II-1. These two fundamental requirements are related to a degree in that excessive loss of hydrogen leads to the formation of undesirable β -phase ZrH which has been associated with abnormal fuel growth and cladding rupture. Under some operating conditions excessive hydrogen loss may also lead to sufficient loss of core reactivity to cause inability to maintain rated conditions of power and temperature. Thus the element permeation-barrier must be capable of withstanding the stresses of launch shock and vibration and thermal transients of operation.

Other considerations in maintaining cladding integrity are the fuel growth associated with fuel in the normal composition range ($N_H > \sim 5.5$) and the cladding stress-levels associated with core operation.

2. Design

The reference fuel element was shown in Figure III-1. Background information on the evolution from S8ER (which suffered extensive cladding cracking) to the S8DR design is given in Reference 7. The fuel-rod composition and dimensions, with the exception of the hydrogen content, are identical to the S8DR design. The hydrogen content was increased from an N_H of 6.05 in S8DR to 6.3 for the reference design. The additional hydrogen increases the fuel phase-change lifetime.

As in S8DR the fuel rod is contained within a Hastelloy-N cladding tube coated internally with SCB-1 ceramic. The cladding tube has a nominal thickness of 0.015 in. The tube OD is 0.570 in. and the active fuel length 16.83. The dimensions of the diametral and axial gaps are dependent on the core design-point power level and coolant temperatures.

A vibration suppressor is used within the axial-gap space to prevent the fuel rod from damaging the ceramic barrier during launch. The current concept consists of a thin stainless-steel column attached to end fittings contoured to fit the fuel-element closure cap and fuel pin (giving the appearance of a collar button). The device possesses little strength at reactor-outlet temperature conditions and collapses as fuel growth proceeds. Limiting testing has been performed on this device.

The gap volumes within the fuel element are evacuated in the process of making the final electron-beam closure weld. Under operating conditions the gap volumes contain hydrogen which is evolved by the ZrH fuel rod. The gaps also serve to contain and dilute the small percentage of fission gases released in the course of operation.

3. Technology Status

In the course of postirradiation examination of the S8ER core, it was found that 80% of the fuel elements had developed cracks in the cladding. It had been known from fission-product activity in the primary loop that cladding failure had occurred. The number of failed elements was believed to be very small, however, based on quantitative evaluation of the fission-product activity and the absence of any other abnormal core characteristic. The surprising finding,

however, that 80% had sustained cracks led to an exhaustive evaluation of S8ER fuel-design fabrication and operating history. The majority of the cracks were determined to have been basically caused by insufficient clearances within the element to accommodate fuel growth. The S8DR design was reviewed in light of S8ER findings. The S8ER data and subsequent inpile fuel irradiation provided a vastly improved understanding of the fuel-growth behavior. Extensive stress analyses and cladding-material irradiation tests were also performed. The ceramic-barrier technology was also markedly advanced. All these results have been factored into the S8DR design which is currently beginning nuclear operation.

C. REACTOR VESSEL DESIGN

1. Requirements

The pertinent design requirements for the reactor vessel are as follows:

- 1) Contain the reactor coolant during test, checkout, launch, and operation in space with a core-inlet temperature of up to 1150°F, a maximum core-outlet temperature of 1300°F at a design pressure of 35 psia;
- 2) Attach to the shield and provide support for the core and reflector assemblies;
- 3) Maintain integrity when subjected to the most stringent shock and vibration requirements of Saturn 1B, Saturn V, Titan 3C, or Titan 3F launch vehicles with the reactor system cold.
- 4) Maintain integrity in space orbit with the reactor operating or shut down when subjected to the accelerations associated with normal attitude-change maneuvers and vehicle docking.
- 5) Operate without maintenance or repair during the reactor lifetime of 20,000 hr at 600 kw with up to 1150°F core-inlet and 1300°F core-outlet temperatures and alternate conditions of power and lifetime.
- 6) Allow maximum use of previously qualified components.

2. Design

The 295-fuel-element reactor vessel with the ten poison-backed control drums was shown previously in Figure III-1a. Cross-section views of the reactor are shown in Figures III-1b and c. All pressure-containing members of the reactor vessel are made of 316 stainless steel. The vessel consists of an outer cylindrical 3/16-in.-thick shell closed at the upper end by a convex spherical dished head and at the lower end by the inlet-annulus which is joined to an extension of the outlet-plenum. The outlet-plenum is closed by a concave spherical dished head. The upper vessel-head is penetrated by ten dry wells open to the ambient (space) environment at the upper end and closed at the lower end to exclude the reactor coolant.

The vessel configuration provides plenums at the two ends of the reactor core. The plenum at the upper end of the core is bounded by the central portion of the upper vessel-head, the inner portion of the dry wells and the upper gridplate. This plenum receives coolant flowing upward and inward from the dry well and reflector region and distributes it downward through the upper gridplate to the core.

The outlet-plenum is formed by an extension of the core vessel which extends to the lower concave dished head. In the current concept four radial pipes below the inlet-annulus provide a return path for reactor coolant to the power-conversion equipment. The number of inlet and outlet pipes could be adjusted to meet the requirements of any particular mission adaptation.

The reactor vessel upper-head is made from a vacuum-cast ring forging machined to the spherically dished configuration shown. The dry wells and central portion of the head are welded in place. The dry-well-to-head forgings are accessible to x-ray inspection. The upper-head has a handhole to permit access for the upper-gridplate installation. The closure weld for the handhole is made after gridplate installation and is not readily accessible for x-ray inspection as now designed. It is expected, however, that with further design effort this weld will be made accessible for this inspection technique.

The lower vessel closure shown in Figure IV-1 consists of a completely welded subassembly of the inlet-annulus, the core-can, the outlet-plenum with coolant-outlet nozzles, and the concave spherical dished-head closure.

Figure IV-1. Lower Vessel
Closure Assembly

The function of the core-can is to separate the NaK flow in the core region from the inlet NaK in the dry-well region. The loading of fuel and the internal reflector elements is made through the dished-head closure. All welds except this closure may be readily inspected by x-ray techniques.

The dry wells, joined to the upper-head and to the lower-gridplate support provide the primary support for the gridplates which in turn support the core. This construction results in a very rigid structure which is expected to meet launch-load requirement with little or no modification. The dry wells also support the NaK-cooled control-drum bearing housings. This provides a structure which assures that the required dimensional relationship within the core and between the core and control drums remains fixed throughout launch and operation.

The hole pattern in the core gridplates provides the variable spacing between the 295 fuel elements and the locating holes for the 42 internal reflector elements. Twenty-four of the internal reflector elements are solid 316 stainless steel rods. The remainder are BeO clad in stainless steel. The gridplates also have an average of two coolant holes per fuel element.

The upper gridplate is bolted to bosses provided on the control-drum dry-well assembly. The core-can (Figure IV-1) forms a slip joint at the periphery of the upper gridplate. The lower gridplate is installed after the fuel and internal reflector elements have been loaded into the core. It is captured between the clamping ring and the core-can extension shown in Figure III-1c. The bolts extend through the core-can extension into the control-drum dry-well assembly.

A weight breakdown of the complete reactor system including control components and fuel is shown in Table IV-1.

TABLE IV-1
REACTOR WEIGHT BREAKDOWN

Item	Weight (Lb)
295 fuel elements	236
Lower gridplate	17
Upper gridplate	13
Dry wells	108
Outer vessel with nozzles	162
Internal reflector elements	15
External fixed reflectors	68
Inner core-can	6
Control drums	436
Bearings and housings	66
Control-drum actuators	150
NaK	94
Miscellaneous	50
Total	1422

D. CONTROL COMPONENT DESIGN

1. Control Drum

The control drums are used to regulate the excess reactivity of the reactor by reflecting or absorbing neutrons which leave the core. The requirements, technology status, and physical description of the reference control drum are described below.

a. Requirements

The control drum is supported by outboard bearings and will be capable of being rotated about its axial centerline by a rotary actuator. The drum has to withstand the expected environmental conditions and meet the operational requirements from factory to end-of-life in orbit.

During launch the control drums will be mounted on the reactor, and will be capable of withstanding Saturn- or Titan-class vehicle launch loads. The ambient pressure during launch will vary from sea level to the vacuum of outer space. The temperature may vary from 30 to 200°F.

During operation in space the drums will be capable of operating at a reactor power level of 1200 kw, a neutron flux of 2.5×10^{13} nv (>1.0 Mev) reactor coolant-inlet temperature of 1100°F, and pressure of 10^{-9} torr or lower, for 20,000 hr. During space operation, the drums will be capable of withstanding 50 reactor startups and shutdowns. The minimum temperature of the reactor coolant in space will be 50°F.

The drum or drum-bearing assembly will be capable of being assembled into a dry well from the top of the reactor, and will be removable at any time before launch. All drums will be interchangeable.

The calculated design reliability of the drum will be at least 99.9 with a confidence level of 90%.

The drum or drum-drive system will contain one limit switch for the neutron-absorber "full-in" position.

The following will be considered in detail during design: margin of safety, thermal stresses, fatigue, launch and thermal-induced deflections of the drum and dry well, drum and dry-well clearance, combined loads, thermal expansion, and irradiation effects.

b. Technology Status

The SNAP 10A reactor contained four control drums supported and driven in a manner similar to the ZrH reactor drums. A flight version of SNAP 10A was launched into orbit and successful operation in space. A ground-test version of SNAP 10A was successfully operated on the ground for 10,000 hr. The

SNAP 8 reactor contains six control drums which are supported in a similar manner. The ZrH reactor drums will not be unbalanced to the extent that the SNAP 10A and SNAP 8 drums were.

c. Reference Drum Description

Figure IV-2 shows the design of the reference control drum. It is a circular cylinder 4-1/2-in. in diameter by 19 in. long. Shafts are located at each end of the drum upon which self-aligning bearings are mounted. The bearing towards the top of the reactor carries both axial and radial drum loads while the lower bearing supports radial loads only. Tantalum with 10% tungsten is used for the shaft material. The shaft journal surfaces are flame-plated aluminum oxide.

Figure IV-2. Reference Design Control Drum

The expected operating temperatures of the drum are shown in Figure IV-3.

The control drum consists of five main parts:

- 1) End-plates to which the bearing shafts are fastened;
- 2) A curved structural member or strongback welded at each end to end-plates;
- 3) Crescent-shaped neutron-absorber blocks (60 wt % Eu₂O₃ in a nickel matrix) bolted to the strongback;
- 4) Neutron-reflector blocks (BeO) bolted to the strongback; and
- 5) A low neutron-absorber cladding enclosing the BeO and the strongback.

	POWER (kwt)	MAXIMUM BeO	MAXIMUM POISON	MAXIMUM CAN
POISON-IN	600	1500	1570	1510
POISON-OUT		1450	1420	1385
POISON-IN	1200	1730	1830	1710
POISON-OUT		1700	1640	1580

8-M22-048-23A

Figure IV-3. Control-Drum Temperatures

Candidate materials for the drum structural members (end-plates and strongback) are:

- 1) Columbium with 10% tungsten;
- 2) Columbium with 1% zirconium;
- 3) Tantalum with 10% tungsten;
- 4) Hastelloy-N; and
- 5) Inconel-600.

The cladding is made of a low neutron-absorber material such as columbium with 1% zirconium, Hastelloy-N, or Inconel-600. Sublimation of the materials while in space is a consideration.

Both the reflector and absorber blocks are made of short blocks fastened at the midplanes to prevent the differential expansion between the various materials from bowing of the drum. The cladding is fastened at one end-plate only to prevent cladding thermal changes from causing bowing.

Belleville springs are used with the BeO mounting bolts to accommodate the differential thermal expansion and irradiation growth of the BeO. Since one end of the cladding is not sealed the irradiation-generated oxygen, hydrogen, and helium gases can escape from the drum. Thermal-cycle growth of the neutron-absorber material (Eu_2O_3 in nickel) will be determined and accommodated as required. Clearance is provided between the BeO and the inside diameter of the cladding to allow for irradiation swelling of the BeO. Differential thermal expansion between mating parts is also accounted for.

2. Control-Drum Bearings

Each control drum is supported by a pair of self-aligning bearings. The axial weight of the drum is supported by the upper bearing and the radial load by both bearings.

a. Requirements

The bearings will support the radial and thrust loads of the control drums under all expected environmental conditions from assembly to end-of-life.

The bearings will support the drum and withstand the shock and vibration loads encountered using Saturn- or Titan-class launch vehicles without auxiliary support devices during launch. Temperatures may vary from 30 to 200°F, and pressure from sea level to the vacuum of outer space.

During space operation of 1200 kw, bearings will operate at minimal torque levels in a neutron flux of 2.5×10^{13} nv (> 1.0 Mev) at coolant-inlet temperatures to 1100°F for 20,000 hr. Bearings will be able to operate through 50 startups and shutdowns and some 200 bearing cycles. Bearings will also operate at coolant minimum temperature of 50°F and during space maneuvers applying acceleration forces of 1-g axial and 1/4-g radial.

The design of the bearings must consider required margin of safety, thermal stresses, fatigue, launch and thermal-induced deflections, and effects of irradiation on bearing performance.

b. Technology Status

Design of control-drum bearings has evolved through SNAP 10A, SNAP 2, and SNAP 8 reactor systems. The basic design is a shaft-ball-socket type of self-aligning bearing. Bearings in SNAP 10A reactors have operated successfully for 10,000 hr under ground-test conditions of 10^{-5} torr in a radiation environment at 800°F. These same bearings operated during the SNAP 10A reactor space mission. S8DR reactor development has included some 50,000 hr testing of bearings at 1150 and 1250°F at 10^{-5} torr.

A prototype high-temperature ZrH reactor bearing has completed a 2000-hr test at 10^{-9} torr performing satisfactorily during some 1050 hr at 1500°F and through 27 thermal cycles. Figures IV-4 and -5 show pre- and post-test condition of the development test bearing assembly. The material selections have been substantiated by compatibility tests at 2000 hr and 1450°F, and 5000 hr and 1350°F. Friction-coefficient studies have included numerous 1250°F, 1000-hr tests at 10^{-6} torr. Table IV-2 gives a summary of tests conducted to date.

2 20 68 UNCL

7696-5555A

Figure IV-4. Development Test Bearing Assembly, Pretest

27 THERMAL CYCLES BETWEEN 70°F AND 1500°F
2000 hr AT 10⁻⁹ torr - 1050 hr AT 1500°F

8-19 68 UNCL

7696 5573A

Figure IV-5. Development Test Bearing Assembly, Post-Test

AI-AEC-MEMO-12715

TABLE IV-2
FRICTION AND THERMAL-CYCLE TEST DATA

Sliding Friction Data				
Couple	Operating Temperature (°F)	Hours at Temperature	Vacuum (torr)	Maximum Friction Coefficient at Temperature
Al ₂ O ₃ vs P5	1250	405	10 ⁻⁷	0.12
	1250	1080	10 ⁻⁷	0.29
	1600	24	10 ⁻⁸	0.26
Al ₂ O ₃ vs AXF-5Q	1600	24	10 ⁻⁸	0.26
K-162B vs P5	1250	1080	10 ⁻⁷	0.35
	1600	16	10 ⁻⁸	0.43
K-162B vs AXF-5Q	1600	16	10 ⁻⁸	0.70

High-Temperature Bearing Thermal-Cycle Test	
Hours at Vacuum (10 ⁻⁹ torr)	2300
at 1500°F	1050
Thermal Cycles (70°F-1500°F)	27
Bearing Oscillation Cycles	180
Maximum Friction Torque (in. -lbs) at 10 ⁻⁹ torr	
at 70°F	22.2
at 800°F	15.0
at 1500°F	9.5

c. Reference Design

Figures IV-6 and -7 show the reference design bearings. Both upper and lower bearings use bearing shafts of Ta-10W, approximately 3/4-in. diam, flame-plated with Al₂O₃. The shaft rotates in a solid P5 carbon-graphite ball 1-3/8-in. diam. The balls in turn rotate and/or oscillate within Al₂O₃ flame-plated Ta-10W socket housings. The upper bearing carries the drum axial load of 60 lb through Al₂O₃-coated Ta-10W thrust washers. The radial load is supported equally by both the upper and lower bearings.

Figure IV-6. Upper Control-Drum Bearing

Figure IV-7. Lower Control-Drum Bearing

Ball-to-socket and shaft-to-ball bearing clearances are maintained throughout the operating temperature range with consideration for the effect of reactivity changes caused by movement of the control drums within the clearances. Belleville washers are used to provide spring-loading of the bearing sockets against the bearing brackets to promote effective cooling of the bearings.

3. Actuators

Each control drum will be gear-coupled to an actuator which will on command provide rotary step-motion in either direction, and will lock the drums when stepping is not commanded.

a. Requirements

The actuator will be mounted to the reactor outer housing, and will be gear-coupled to the control-drum upper shaft extension. In this location it must be capable of withstanding the expected launch and operating environments. Temperature of the actuator is expected to run approximately 300°F lower than the drum shaft to which it is coupled.

The actuator torque requirement will depend upon the gear ratio selected but it is expected the needed torque, as fixed by operation during space maneuvers, will be somewhat higher than the S8DR requirement. The step size will be larger, approximately 4°.

The actuator must be designed to operate on a dc voltage, less than 56 volts, available from the converter.

The mechanical configuration and overall envelope of the actuator will be similar or identical to the S8DR actuator, shown in Figure IV-8. A change of the rotor, endbell, and pinion material may be made to improve the structural integrity, particularly with regard to impact resistance. The phase and brake-coil conductor material will be selected to enable higher conductor current densities and therefore high torques since past operation has been well below magnetic saturation. The choice will be based on coil development work performed since the S8DR actuator design.

11-15-67 UNCL

7695-55516

Figure IV-8. S8DR Actuator

b. Technology Status

The basic mechanical features and performance of the proposed actuator design have been demonstrated by the operation of a number of S8DR actuators beyond 10,000 hr at 1025°F. A smaller number have operated at 1125 for over 5000 hr. Extensive testing of conductor materials and insulation to 1400°F at constant temperature and under thermal-cycle conditions has been performed. Many of the features of the proposed design were included in the SNAP 10A actuator, which operated successfully in space after being launched into orbit as a part of the SNAP 10A reactor control system.

4. Control-Drum Drive System

The control-drum drive system consists of the control drums, bearings, S8DR-type actuators, and gears previously discussed in this section. The relationship of the actuator pinion gear to the drum gear is maintained by means of a housing integral with the actuator motor.

Figure IV-9 shows the control-drive system applied to the reference reactor in a shadow-shield configuration. An internal toothed gear is secured directly to the control-drum upper shaft extension. The actuator output pinion gear meshes with the internal tooth gear providing a one-to-six gear ratio. The gear housing, which supports and positions the actuator, is located by and secured to the upper control-drum bearing housing. This housing includes the mounting location for the full-out limit switch. The limit-switch cam is attached to the internal tooth gear. The switch-cam angular position is adjustable to provide adjustment of the limit-switch setpoint.

This system achieves a degree of compactness consistent with the requirements of the shadow-shielded application. The same drive system is applicable to the 4π -shielded reactor system with the addition of an extension shaft, flexible couplings, and a bearing.

Figure IV-9. Reference Design Actuator Installation

V. SHIELD DESIGN AND PERFORMANCE

A. INTRODUCTION

The specific design and performance requirements placed on the reactor shielding are highly dependent on the mission application. For this reason this section will discuss requirements for typical applications, factors which have a general influence on design, a description of the design and performance of a typical 4π shield and finally, the state-of-the-art shield technology status.

B. TYPICAL REQUIREMENTS

Shields basically fall into two classifications, those for manned applications and those for instrument applications. An instrument-rated shield typically provides a specified level of attenuation in a shadow-cone volume containing the payload. Allowable dose levels are governed by the nature of the payload. Radiation tolerance levels of various electronic components which would frequently make up a payload are shown in Figure V-1. Because there would seldom be a need for shielding in directions other than the payload a shadow shield is generally all that is required for an instrument payload.

7-D31-273-44

Figure V-1. Radiation Tolerance Limits
for Typical Components

For manned-space-station applications there may or may not be a need for dose attenuation in directions other than the space station. Frequently there will be a need to attenuate dose to allow reactor operation during a rendezvous operation with another vehicle. This results in shields providing tailored doserates in two or more zones. Such shields are referred to by the fraction of the total solid angle which is shielded. A 4π shield provides some attenuation in all directions.

Typical dose levels for manned shields are shown in Table V-1.

TABLE V-1
TYPICAL DOSE LEVELS FOR MANNED SPACE APPLICATIONS

Type Shield	Dose Level
Manned Shadow	2 to 3 mr/hr
4π Payload	2 to 3 mr/hr
Parked vehicle area	2 r/hr*
Rendezvous zone	100 r/hr

*Dictated by scattering to the payload

Shielding requirements for a lunar base application differ significantly from those of a space station principally because the practical separation between man and reactor is greater.

Other considerations significant in the reactor shield design are the weight, total system reliability requirement, and launch load structural requirement. In a space application, reductions in weight can generally be obtained by increasing the reactor/payload separation distance. The weight savings in the shield must be balanced against the weight of additional structure and the reduction in system reliability resulting from devices which must mechanically extend and retract the reactor power system.

C. DESCRIPTION OF TYPICAL 4π SHIELDS

Because of the structural and thermal interaction between the reactor and 4π shielding the integrated combination of both is defined as the reactor system. A typical 4π arrangement is shown in Figure V-2. This conceptual configuration was derived for the Saturn S-IVB Workshop and is discussed from that standpoint in Reference 1.

a. Configuration

8-J1-099-1

10-16-68 UNCL

8-JY22-099-39A

b. With Structure

Figure V-2. Reference Design 4π Shield

AI-AEC-MEMO-12715

The gamma shield immediately surrounding the reactor is molten lead contained in a 2-1/4 Croloy container. The choice of lead proves to be advantageous over other gamma-shield materials such as tungsten and depleted uranium because of its low secondary-gamma production in the high neutron-flux environment adjacent to the reactor. The gamma shield is followed by a lithium hydride neutron shield which is thermally insulated from the gamma shield to maintain temperatures less than 1100°F (the LiH melting point is 1270°F). Shielding thickness and contours are optimized to produce the minimum shield weight to attain the required dose levels. The shield shown is sized for a maximum dose of 3 mr/hr in the workshop, up to 2 r/hr in the rendezvous vehicle parking area, and 100 r/hr in the rendezvous maneuvering zone. The greatly reduced neutron and temperature levels at the gamma shield below the equipment gallery permit the use of depleted uranium. This gamma shield provides the added shielding required for the reactor source plus shielding of the workshop from the induced radioactivity in the primary-loop NaK coolant. The final shielding region is lithium hydride.

The major source of heat generation in the shielding is from core gamma-ray absorption in the lead shield. Approximately 10 kwT of energy are deposited at a 600-kwT reactor power level. The generated heat is radiated from the shield to the reactor vessel where it is transferred to the NaK coolant. The lesser amounts of heat generated in the remaining shielding regions are conducted to the shield surfaces where the heat is radiated to space.

The shield provides the structural support for the reactor. Launch loads are carried through aerospace-type shell-and-stringer structures contained within the cast lithium hydride shields. The gallery region is bridged by structural panels bolted in place after the equipment in the gallery is assembled and operational. Reactor loads are carried to the shield structure through support lugs located at the lower edge of the reactor vessel in the vicinity of the NaK-inlet lines. The cast lithium hydride is supported within the shield structures by a combination of mechanical bonding to the internal structural members and by transferring loads through the shield casings. Gamma shielding is supported directly on the primary structural members.

10-16-68 UNCL

7759-5212

Figure V-3. Preliminary Shield Temperatures, 600 kw

D. PERFORMANCE

As a static device the principal performance characteristic of a shield is the operating temperature. In the 4 π reference reactor system design the shield immediately below the reactor vessel is considered to be the most critical from the thermal standpoint. This is the result of the following considerations:

- 1) The heat-generation rate is relatively high because of the close proximity of the reactor.
- 2) The heat sink for the lead gamma shield is the 1300°F reactor outlet-plenum.
- 3) The LiH heat sink is limited to the side of the shield because of the presence of the gallery.
- 4) The internal structural shells within the shield impose thermal barriers to the heat flow.

A simplified spherical model of the shield was used to perform a preliminary thermal calculation. The gallery region within the structural panels was assumed to be adiabatic as was the joint between this shield and the top shield. Heat transfer across the internal structural panel was assumed to be by radiation only. The shield dimensions used in the study were larger than those of the S-IVB workshop design, which tends to make the results conservative. Results at 600 and 1200 kw are shown in Figure V-3. They are considered to be promising since the maximum LiH temperature of 1000°F is safely below the LiH melting point of 1270°F. This melting point is known with reasonable certainty to be the only upper limit on shield temperature. Possible low temperature limits in an irradiation environment and lead gamma-shield temperature considerations are discussed below in the technology status section.

LiH as an ionic-type compound has a low dissociation pressure relative to most other hydrides. At the melting point it is 27 mm Hg. Nevertheless the small but finite dissociation pressure would result in hydrogen loss through a weld leak in the shield casing or through a meteoroid puncture. Of these the meteoroid puncture is considered to be the more significant because

of the potential for one or more large punctures. Results of a previous study of this problem are applicable to the current design and are described below.

Hole diameters from meteoroid impact were conservatively assumed to be equal to twice the penetration depth in a semi-infinitely thick target, based on experimental hypervelocity impact data. A spalling factor of 1.75 was used to define the plate thickness which would just allow penetration for a meteoroid of given mass. The semi-infinite target-penetration depth was taken from a semi-empirical equation in the NASA-417-5 specification.

Having established a relationship between hole diameter, shield casing thickness, and particle mass, the mass-frequency distribution of Whipple was used to obtain the hole-size frequency distribution. Flowrates through the resulting punctures were then calculated as a function of the LiH plateau dissociation pressure within the shield which is given by

$$P \text{ (mm Hg)} = \exp \left(\frac{-18,950}{T \text{ (}^\circ\text{K)}} + 22.9 \right).$$

The proper temperature to use in the above equation is the maximum LiH temperature, since this will establish the pressure level within the shield.

The resulting equation for the fraction of hydrogen loss is given by

$$F = \frac{7.6 \times 10^{-12} A \theta^2}{W t^2 T^{1/2}} \quad (P_{\text{mm Hg}})$$

where

A = shield casing surface area in m^2 ,

θ = time in orbit-seconds,

W = original mass of hydrogen grams,

t = stainless-steel casing thickness in cm, and

T = hydrogen temperature in $^\circ\text{K}$.

Inherent in the equation are four very conservative assumptions. First, the hydrogen-loss model is based on a continuum flow of gas through an ideal orifice with a coefficient of 1.0. Second, it is assumed that no diffusion is required to transport the hydrogen from the hottest part of the shield to the

holes in the cool outer surface. Third, no plugging of the holes occurs. Fourth, the 1963 Whipple distribution overestimates the number and size of micrometeoroids.

Using this expression an evaluation was made of the hydrogen loss from the upper shadow shield for the S-IVB workshop applications. LiH temperatures and casing thickness were taken as variables. The exposed surface area of the casing was taken as 2.3 m^2 . Results are shown in Table V-2.

TABLE V-2
SHIELD HYDROGEN LOSS THROUGH METEOROID PUNCTURES
AFTER 20,000 HOURS IN ORBIT

Max LiH Temperature (°F)	Shield Casing Thickness (in.)	Fraction of H ₂ Lost (%)
1000	0.040	over 80
	0.080	47
	0.120	21
1100	0.040	over 80
	0.080	over 80
	0.120	over 80
900	0.040	36
	0.080	8.7
	0.120	4.0
800	0.040	5.1
	0.080	1.3
	0.120	0.56

Experimental data⁽⁸⁾ indicate that the hydrogen loss rates for a given hole size and the micrometeoroid flux may each be overestimated by a factor of 3 to 5. The hydrogen-loss values in the table, therefore, are probably high by a factor of 10 to 25.

Examination of Table V-2 indicates that at LiH temperatures in the 1000 to 1100°F range, the shield casing thickness may be dictated by meteoroid considerations rather than structural requirements. It should be noted that minor losses of hydrogen may be accommodated by increasing the initial thickness of

the shield. The trade-off between this approach and adding increased thickness to the shield casing must be made on an individual case basis. Another approach in accommodating LiH temperatures in the range above 1100°F is to segregate the high-temperature material within a gastight inner shell. The same shell could double as a structural member. This arrangement would complicate the shield fabrication process although it should be noted that a similar two-region shield was developed for the SNAP 8 DRM system.^(9,10)

E. SHIELD MATERIALS TECHNOLOGY STATUS

1. Gamma-Shield Materials

a. Liquid Lead

Corrosion studies conducted at Al of liquid-lead and lead-bismuth alloy versus 1020 carbon steel, Type 446 stainless steel, and Croloy 2-1/4 for 10,000 hr at 1000, 1300, and 1500°F were used to define material combinations and maximum operating conditions for advanced-reactor shielding. These tests established that in a static-system liquid lead does not corrode any of the three structural materials. Evidence of decarburization which was observed was not believed to significantly detract from the performance of the structural alloys.

Croloy 2-1/4 was selected as the better structural material, not on the basis of corrosion resistance to lead but on its better mechanical properties.

Lead-bismuth alloy was found to be corrosive at 1300 and 1500°F and was relegated to systems operating at lower temperatures.

b. Depleted Uranium and Alloys

A review of the literature on uranium and its alloys revealed that unalloyed uranium should not be used in most gamma-shield applications, and that a stabilizing additive such as molybdenum should be present. Uranium alloyed with 2 to 8 wt % molybdenum alloy was selected as the reference material, with the alloy content dependent on operating temperature and severity of service.

Vendor information indicates that massive slabs of uranium alloy probably could be fabricated to the sizes visualized for future shields, although it is possible that tailored shaping of the shield may preclude the use of monolithic slabs, resulting in the use of several pieces mechanically joined together.

c. Tungsten and Alloys

Review of the available literature on tungsten and alloys has revealed that a number of materials are available for gamma shielding; for example, the W-Ni-Cu and W-Ni-Fe alloys are suitable for low-to-medium temperature application, and pure W or W-Mo alloys may be used for high-temperature application.

Tungsten-7.5Ni-2.5 Cu alloy was selected as a backup for lead in the current reference design from trade-offs of properties, temperature application, and cost. Unfortunately tungsten fabrication today is limited to relatively small shapes, and future shields will have to be designed to utilize blocks or slabs mechanically joined together. It is not likely that massive monolithic shapes will be available in the near future unless extensive fabrication development programs are undertaken.

2. Combined Neutron-Gamma Shield Materials

Recently a new family of materials which function as both neutron and gamma shielding⁽¹¹⁾ was conceived. The materials investigated to date are aggregates of crushed zirconium hydride or tungsten cemented together with a matrix of cast lithium hydride (LiH). This concept is not limited to these materials, but could be adapted to any gamma-shielding material which does not react chemically with LiH.

The majority of the work has been limited to laboratory-size specimens, but one large demonstration item (200 lb of W-2Mo) has been fabricated.⁽¹¹⁾ Specimens of LiH/W have been prepared with a density of 8.5 gm/cc (0.68 wt% H), and it is believed that densities of 12.0 gm/cc (0.30 wt % H) are possible. Similarly LiH/ZrH compositions of 2.9 gm/cc (3.5 wt % H) have been prepared, but densities of 3.7 gm/cc (2.6 wt % H) or greater appear possible.

These materials not only have the properties of combined neutron-gamma shielding, but also have the very important advantage that they may be made into large monolithic bodies whereas tungsten or zirconium hydride cannot be made into large shapes. This one advantage plus the use of lower cost, granular, or chunk material may encourage the use of these materials in future shield designs.

TABLE V-3
SNAP SHIELD DEVELOPMENT

Shield	LiH Weight (lb)	Shape	Special Features
SNAP 10A test specimen	200	Cylindrical	First large casting
SNAP 10A cast shield	200	Conical, S10A casing	
SNAP 8 two-region test specimen	450	Cylindrical	Two-region casting
SNAP 8 DRM-1	780	Conical	Two-region shield
SNAP 8 thermal-test shield	240	Conical, S10A casing	Measured bulk thermal conductivity
S10A DRM-1C shield	200	Conical, S10A casing	Internal stiffening structure
SNAP 2 developmental shield	300	Conical	Internal ribs and reactor supports
SNAP 2 DRM-1 shield	300	Conical	Same as S2 developmental shield
Lithium-metal-back- filled shield	240	Conical	SNAP 8 thermal-test shield backfilled with lithium
SNAP 8 experimental shield No. 1	200	Conical S10A casing	Internal tubular structure
GST experimental shield No. 2	200	Conical, modified SNAP 10A	Pipe penetrations and tungsten aggregate
S8DR shield	780	Conical, modified S8DRM-1	One-region, instrumented for nuclear testing

3. Neutron-Shielding Materials

Lithium hydride has been selected as the primary neutron-shielding material for space reactor application after many years of development studies and testing.^(10,12) This selection was based on its excellent high-temperature stability (melting point $\sim 1270^{\circ}\text{F}$, dissociation pressure at the melting point about 27 torr), attractive shielding properties, radiation stability, and lack of corrosiveness to most common structural alloys at shield operating temperatures.

Many other materials have been evaluated currently and in the past but none has been found to possess such an attractive combination of properties plus ready availability and relatively low cost ($\sim \$10/\text{lb}$).

A fabrication and casting development program at AI starting in 1962 has produced shields and test specimens as shown in Table V-3. Some of these hardware items have been used to develop new fabrication techniques while others such as the S8DRM, the S8DR, and the two S2 shields were fabricated to system requirements.^(8,9)

To date, these shields have been limited to about 700 lb of LiH, restricted not only by system requirements, but also by equipment limitations. Design studies have shown that future shields will require 1000 to 2000 lb of LiH and be larger in diameter and thinner than previously fabricated. It is believed that with larger casting equipment, techniques can be developed to produce LiH shields conforming to these requirements.

REFERENCES

1. "Reference ZrII Reactor-Thermoelectric System," (CDI), AI-AEC-MEMO-12717, to be published.
2. "Advanced SNAP Reactor Capability and Program Study," (CRD), NAA-SR-MEMO-12522, Volume I, October 20, 1967.
3. "Nuclear Fuels and Materials Development," U.S. AEC Division of Reactor Development, G.E. NMPO-TID-11295, February 1961
4. "Fifth Annual Report, High Temperature Materials Programs, Part A," General Electric Co., GEMP-400A, February 28, 1966.
5. "Sixth Annual Report, High Temperature Materials Programs," General Electric Co., GEMP-475A, March 31, 1967.
6. K. R. Birney, "An Empirical Study of SNAP Reactor Fuel Irradiation Behavior," (CRD), NAA-SR-12284, August 10, 1967.
7. L. D. Swenson, "S8DR Core Performance Evaluation." (CRD), NAA-SR-12482, August 9, 1967.
8. F. H. Welch, "Properties of Lithium Hydride for SNAP Shielding Applications," (CRD), NAA-SR-9400, Volume III, May 1, 1967.
9. F. H. Welch, "Fabrication of Shadow Shields for SNAP Applications," (CRD), NAA-SR-9400, Volume II, September 30, 1965.
10. F. H. Welch, "Testing and Examination of SNAP Shadow Shields," (CRD), NAA-SR-9400, Volume V, August 31, 1967.
11. F. H. Welch, "Lithium Hydride Technology IV. A Novel Neutron-Gamma Shielding Material for SNAP Applications," NAA-SR-9400, Volume IV, May 25, 1967.
12. F. H. Welch, et al, "Lithium Hydride Technology, Properties of Lithium Hydride and Corrosion Studies, NAA-SR-9400, Volume I, May 8, 1964.