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OCTOBER - DECEMBER 1962
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NAA-SR-8092
SNAP REACTOR
SNAP PROGRAM
M-3679 (28th Ed.)

GROUP I

SNAP 8 QUARTERLY PROGRESS REPORT

OCTOBER - DECEMBER 1962

(Title Unclassified)

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Edited by

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The previous SNAP 8 Quarterly Progress Report, NAA-SR-7792, covered the period July-September 1962. It was published in three volumes.

Vol. I — Reactor Engineering and System Tests,
C. E. Johnson (SRD)

Vol. II — Fuel Element Development and Fabrication,
G. W. Meyers (SRD)

Vol. III — Component Development, O. P. Steele, III (CRD)

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I. PROGRAM OBJECTIVES

SNAP 8 is a nuclear power supply system designed to provide 30 or 60 kw of electrical power on space vehicles. It is based on the SNAP 2 concept of a NaK-cooled uranium-zirconium hydride reactor coupled to a mercury Rankine cycle power conversion system. The SNAP 8 system is being developed jointly by the Atomic Energy Commission (AEC) and the National Aeronautics and Space Administration (NASA).

The objectives of the SNAP 8 development work under this contract are to (1) develop the SNAP 8 reactor, including its associated controls, shield, and safety features; and (2) test the complete SNAP 8 system, i. e., reactor coupled to power-conversion equipment. The power-conversion equipment is being developed by the Aerojet General Corporation (AGC) under contract to NASA.

The general objectives and operational constraints are shown in Table I.

TABLE I.
SNAP 8 GENERAL OBJECTIVES

Planned application	Electrical propulsion
Test vehicle	Atlas-Centaur
Safety criteria	Subcritical during launch Orbital startup Orbital shutdown at end of power-conversion system (PCS) life backed up by reentry shutdown mechanism
Life	10,000 hr
Power conversion	Mercury Rankine cycle Turboalternator being developed by Aerojet under contract to NASA
Electrical power output	30 kw (capable of extension to 60 kw)
Reactor thermal power	300 kw (capable of operation at 600 kw)
Control mode of operation	System operates at full power. Electrical power not used by payload is dissipated in parasitic load. The reactor control system maintains the reactor coolant outlet temperature around 1300°F by stepping reflector control elements in or out as required.
Reactor weight objective	500 lb
Shadow shield weight objective	1000 lb

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The shadow shield is designed to reduce the direct dose at the payload to 5×10^{12} nvt from fast neutrons and 9×10^6 rad from gammas. The dose due to scattering from the radiators or other structures is assumed to bring the total dose at the payload to 10^{13} nvt and 10^7 rad. The payload is assumed to be 8 ft in diameter and located 15 ft from the reactor.

II. PROGRESS SUMMARY

A. FLIGHT REACTOR DESIGN

S8ER critical experiments were completed and results analyzed; subsequently the reference design of the SNAP 8 nuclear system was updated by increasing the active length of the fuel element from 16.0 inches to 16.5 inches. Corresponding length changes have been made in core vessel and reflector assembly.

In order to decrease anticipated fuel cladding creep, the core vessel internal pressure has been increased to 40 psia from 25.

B. FUEL DEVELOPMENT

Successful development of the nucleated glass coating, SCB, has progressed rapidly. The process is so promising that development of final closure seals has been accelerated. An inpile permeation experiment is now scheduled to obtain data for consideration in the design of the S8DS-1 fuel elements.

C. COMPONENT DEVELOPMENT

Tests are continuing on the investigation of bearing material capabilities for the SNAP 8 environment. Cermet surfaces with Molykote X-15 dry lubricant applied have survived a 1487-hr test at 8×10^{-7} Torr and 750 to 930°F.

Development of components comprising the reflector drive, safety, sensor, and actuator systems is continuing.

The LiH test casting shield has undergone vibration and shock tests and visual examination indicates no external structural damage.

D. SYSTEM TESTS

1. SNAP 8 Experimental Reactor (S8ER)

All heat transfer and service system equipment outside of the S8ER containment vessel have been installed and satisfactorily checked out.

The core vessel to be used for S8ER power tests was fabricated and welded into its final location in the system.

As a result of critical experiment analyses, another 1.30 in. of shim was added to the beryllium control drums prior to power operation. Thus the

as-built S8ER reflector-control drums are approximately 5 in. thick and have a radius of 4.65 in.

Permeation testing of fuel elements after completion of the critical experiment showed that the hydrogen barrier coating on the inside of the fuel cladding had not been degraded by the handling processes involved.

2. SNAP 8 Developmental Reactor Mockup (S8DRM-1)

All fabrication drawings were released during this quarter and fabrication of many items was completed.

The flight type controller and the temperature sensor switches were eliminated from the program when it became obvious that additional component development was required to meet the 10,000-hr endurance requirement in a radiation environment. Elimination of these items will not seriously affect the test program.

3. SNAP 8 Ground Prototype (S8DS-1)

The design of the nuclear shakedown heat transfer loop is nearing completion. Fabrication of the diffusion cold traps, NaK filters, and NaK tank liquid level sensing devices was completed. Design criteria for the reactor have been completed. Preliminary design studies are in progress.

4. SNAP 8 Flight System (S8FS-1)

Design of the S8FS test facility is continuing. Rough grading and site excavation have been completed.

III. DESIGN DESCRIPTION

The SNAP 8 nuclear system consists of a shielded compact nuclear reactor moderated by zirconium hydride and cooled by liquid NaK-78. The reactor is controlled by movable beryllium reflectors which control the amount of leakage flux by their position. The current weight objective of the nuclear system is 500 lb not including the shield. The shield weight is strongly dependent upon the payload size and location. (The current weight of the shield is 1045 lb.) Each major assembly is described in the following paragraphs. A weight summary of the SNAP 8 nuclear system is given in Table II.

TABLE II
SNAP 8 NUCLEAR SYSTEM WEIGHT SUMMARY

Core Assembly		296 9
Head Assembly, outlet plenum	6 0	
Upper grid plate	6 7	
Large internal reflectors	7 6	
Small internal reflectors	1 5	
Fuel elements	201 0	
NaK in core vessel	13 6	
Basic vessel	33 8	
Lower grid plate	6 2	
Lower grid plate supports	0 5	
Baffle plates	1 1	
Reflector brackets	1 1	
Lower ring	1 8	
NaK pipe and NaK	14 0	
Temperature sensor switch	2 0	
Reflector Assembly		273 0
Control motors	15 0	
Startup motors	9 0	
Reflector adjust brackets	1 2	
G C shutdown actuator	1 0	
Upper drum support assembly	15 0	
Reflector external, fixed	42 9	
Reflector drums	110 8	
End-of-life shutdown actuator	1 8	
Diagnostic instrumentation	50 0	
Safety circuits	5 0	
Drum release actuators	3 6	
Lower drum support assembly	12 0	
Reflector support structure	1 2	
Reflector ejection springs	4 0	
Drum position servos	0 5	
Shield Assembly		1,045 3
Marmon ring	2 0	
Reactor support structure	7 4	
Gamma shield	33 6	
Destruct charge	11 0	
Upper head	51 1	
Thermal insulation	8 9	
Upper ring	27 5	
Upper and middle diaphragm	57 3	
Shell	72 5	
Cables	10 0	
Multiple connectors	5 0	
Neutron shield	681 0	
Fill pipe	2 1	
Lower ring	13 0	
Lower diaphragm	34 9	
Mounting ring	28 0	
Instrument Compartment	17 0	17 0
TOTAL		1,632 2lb

TABLE III
SNAP 8 NUCLEAR SYSTEM CHARACTERISTICS*

Operating Conditions		Uranium Loading, Kg U ²³⁵	6.6
Reactor Thermal Power, kwt	300-600	Mean Prompt Neutron Lifetime, μ sec	6.7
Net Electrical Power, kwe	30-60	Burnable Poison Material	Sm ₂ O ₃
Operating Life, hr	10,000	Reactivity worth, % $\Delta k/k$	-2.7
NaK Outlet Temperature, °F	1300	Initial Cold Excess Reactivity, % $\Delta k/k$	6.1
NaK Inlet Temperature, °F	1100	Total Reactivity Losses, % $\Delta k/k$	
Operating Pressure, psia	40	Xenon	-0.9
Maximum Fuel Tem- perature, °F	1580	Samarium	+1.8
Maximum Cladding Tem- perature, °F	1480	Temperature and power	-1.7
Reactor Design		Hydrogen Loss	-2.5
Fuel Elements		Hydrogen redistri- bution	-0.5
Number	211	Burnup and fission product poisons	-1.3
Fuel diameter, in.	0.529	Contingency	-1.0
Cladding diameter, OD, in.	0.560	Total Control Drum Worth, % $\Delta k/k$	14.0
Cladding thickness, in.	0.010	Shield Parameters	
Fuel alloy, wt % U in Zr	10.5	Material	
Degree of hydriding, NH	6.0	Neutron shield	LiH
Cladding material	Hast-N	Thermal insulation	Min-K
Active fuel length, in.	16.5	Gamma shield	W
Internal Reflectors		Neutron Shield Cladding	
Composition	BeO	Material	316 SS
Cladding material	347 SS	Thickness, in.	0.109
Cladding thickness, in.	0.018	Emissivity	0.9
Inlet Plenum Grid Plate		Dimensions, Neutron Shield	
Material	Hast-C	Maximum diameter, in.	40
Thickness, in.	7/16	Minimum diameter, in.	31.4
Outlet Plenum Grid Plate		Height, in.	28.6
Material	316 SS	Dimensions, Gamma Shield	
Thickness, in.	7/16	Diameter, in.	9.5
Core Vessel		Height, in.	1.0
Internal diameter, in.	9-3/16	Dimensions, Thermal Insulation	
Length, in.	22.4	Thickness, maximum, in.	3-3/4
Thickness, in.	0.105	Maximum Dose at Pay- load	
Material	316 SS	Fast neutrons	
Reflector Control Ele- ments		Scattered, nvt	5×10^{12}
Number	6	Direct, nvt	5×10^{12}
Material	Be	Gammas	
Nominal thickness, in.	3	Scattered, rad (C)	1×10^6
Nuclear Parameters		Direct, rad (C)	9×10^6
Average Thermal Flux, n/cm ² -sec	5×10^{12}		
Mean Fission Energy, ev	0.20		
Average Isothermal Tem- perature Coefficient, ¢/°F	-0.2		

*Data listed are for operation at 600 kw. Shield data, however, are for 300 kw-operation.

A. FUEL ELEMENTS

The SNAP 8 reactor core contains 211 fuel elements made of Zr - 10.5 wt % U alloy hydrided to a hydrogen content of 6×10^{22} hydrogen atoms/cc of $(U - Zr)H_x$. The uranium is 93.15% enriched. The elements are clad in 10-mil-thick Hastelloy-N tubing coated internally with a hydrogen permeation barrier. The active length of the fuel element is 16.5 in. Other dimensions and temperatures are listed in Table III.

During this reporting period, the reference design of the fuel element has changed slightly. The core internal pressure has been raised from 25 to 40 psia to decrease fuel rod cladding long-term creep. The quantity of uranium in the U-Zr fuel alloy has been increased from 10.0 to 10.5 wt % and the length of the active fuel rod has increased from 16 to 16.5 in. The main reason for these changes is due to the S8ER critical experiments which showed the calculated effective reactivity and drum worth to be lower than the experimentally measured values. The active fuel rod diameter was also decreased one mil to allow for slightly more irradiation swelling.

The fuel element upper end-pin configuration was changed from the piston-type to a taper pin-type. This change was made when analysis of the piston-type and pin arrangement at both ends of the fuel rod showed little damping advantage over using a piston-type at one end and a tapered pin at the other end. The advantage of the taper pin is two-fold. First, manufacturing problems are eased as tolerances are less severe; and secondly, assembly problems are lessened due to the increased tolerances and the taper on the end cap.

An optional reference design using a hold-down device to hold the active fuel rod tight during launch environments was formulated during this quarter. This design will be followed until it is conclusively proved by testing that a hold-down device is not needed.

B. CORE VESSEL AND INTERNALS (less fuel elements)

The core vessel is a cylinder of 316 stainless steel having a reverse dished inlet plenum head and a torispherical outlet plenum head. The coolant inlet nozzle is a flued elliptical opening located in the center of the torispherical head. The core vessel contains an upper grid plate of 316 stainless steel; a lower grid plate of Hastelloy-C; a coolant baffle plate of 316 stainless steel; internal BeO, 347-

stainless-steel-clad reflectors; and, of course, the 211 fuel elements previously described.

In order to accommodate the increased active fuel element length, the core vessel has been lengthened from 21.6 to 22.4 in. The upper grid plate design was modified to accommodate the change in end-pin configuration.

Due to the problems encountered during manufacturing of the S8ER internal reflectors, a critical review of the reference design was initiated. As a result of this review, the cladding thickness was changed from 10 to 18 mils in order to hold the cladding tolerances more readily.

The beryllium reflector mounting arrangement was changed during this quarter. The reflector now is positioned with respect to the cylindrical portion of the core vessel instead of the core vessel outlet head. This design change was incorporated mainly to facilitate remote removal of the reflector and to ease assembly procedures.

C. CONTROL AND REFLECTOR SYSTEM

The beryllium control reflector system is made in two halves so that it forms a hollow right circular cylinder completely surrounding the active core length when the six control drums are inserted. The nominal beryllium reflector thickness is 3 in. Each control drum is driven by its own drive motor. The three startup drums are driven by externally commutated d-c motors and the three control drums, are driven by stepper motors. A startup programmer and long-term controller are provided to achieve proper operational sequence of the drums. A temperature sensor-switch is included to measure the outlet coolant temperature.

As a result of the increased active fuel rod length, the reflector system was lengthened during this quarter. As discussed in Section III-B, the beryllium reflector assembly is positioned with respect to the reactor vessel cylinder. Numerous minor changes have occurred in the reference design of the control and reflector system as a result of the developmental program discussed in Sections V-B, C, D, and E.

D. SHIELD

The shield is designed to limit the neutron dose at the payload to 5×10^{12} nvt direct and the gamma dose to 9×10^6 rads (carbon) direct. (The payload configuration is an 8-ft diameter circle, 15 ft from the reactor.) The total dose, direct

and scattered, at the payload should not exceed 10^{13} nvt or 10^7 rads. The shield is composed of two parts, one a gamma shield and the other a neutron shield. The gamma shield is a plate of tungsten located directly beneath the core vessel lower plenum. The neutron shield is a two-region shield. The region nearest the reactor contains 99.99% Li^7H while the region located below contains natural LiH .

Although the current reference design is still based upon a 300-kwt reactor, a new reference design is being formulated. This new reference design will be based upon a reactor power level sufficient to transfer 600 kwt to the NaK coolant.

E. SAFETY SYSTEM

The safety system includes a reentry band, reflector ejection springs, a destruct charge, an end-of-life scram actuator, a telemetry scram actuator, and associated circuitry. In addition, void filler blocks and a drum lockout system are furnished as part of the ground handling safety equipment.

The reentry band is a thin band of stainless steel which holds the reflector assembly in place at the top of the core vessel. The band contains a spring to provide proper tension, the telemetry scram actuator, and the end-of-life scram actuator. The band is designed and located so that it will be melted by reentry heat. Breaking of this band allows the reflector ejection springs to remove the reflector assembly from the vicinity of the core vessel and this, in turn, opens the marmon clamp holding the core assembly to the shield assembly.

The destruct charge is a shaped explosive surrounding the core vessel, the force of which when fired is sufficient to preclude any possible reactor criticality.

The void filler blocks are neutron poison blocks designed to be slipped into the cavities in the stationary beryllium reflector normally occupied by the control drums when in their full-in position. The drum lockout system is a key operated locking system designed to prevent control drum motion even if the drive motors are energized.

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IV. SYSTEM ANALYSIS

A. NUCLEAR ANALYSIS

1. SNAP 8 Experimental Reactor (S8ER)

Data from the S8ER critical experiments has been correlated with calculations and the results of these studies are reported in Section VI-A-8 of this report.

a. Lifetime Expectation with Present Shims

The excess reactivity of S8ER was calculated as a function of time. The initial excess was assumed to be the \$5.96 excess with A and B shims that was measured in the SNAP critical facility. The reactivity losses were based on experimental results when possible. These calculations indicate that S8ER will operate at a power level of 300 kwt for 10,000 hr, but will not operate for 10,000 hr at the 600-kwt level.

b. Selective Fuel Loading

Calculations were performed to determine the effect on reactivity of various selective loading schemes for S8ER. There appears to be little advantage in selectively loading the S8ER in an attempt to increase system reactivity.

c. Change of Power Distribution with Thicker Reflectors

Some increase in the power peaking at the outer core edge has been calculated due to the addition of the B shims (5-in. nominal reflector thickness). With only the A shims (3-in. nominal reflector thickness) in place, the outer peak power was calculated to be 1.19 times the minimum while the outer peak power rose to 1.26 times the minimum with the thicker reflector.

d. S8ER Vault Worth

The reactivity worth of the S8ER vault wall worth was reviewed during the quarter. Previous calculations of S8ER vault worth had estimated the worth at about 0.011 Δ k based on a similar value for the SER vault wall worth. It was recently discovered that the fuel elements had been switched from the SER in the critical assembly to the SER in the vault and part of the reactivity change must therefore be assigned to the fuel element change, thus leaving no basis for estimating the S8ER vault wall worth at this time.

e. S8ER Neutron Lifetimes

Neutron lifetimes as calculated by ULCER were $7.28\mu\text{sec}$ and $5.1\mu\text{sec}$ for drums in and out, respectively. A FAIM calculation gave a lifetime of $6.7\mu\text{sec}$ for these drums.

2. SNAP 8 Ground Prototype Reactor (S8DS)

a. Design Studies

Efforts were undertaken to define new one- and two-dimensional mockups of the current S8DS reactor design. This design utilized a 16.5-in. fuel length, 6.0 N_H , with 10.5 wt % uranium in the uranium-zirconium hydride. Preliminary calculations indicated the effective multiplication (k_{eff}) to be 1.050 for this reactor with only the A shim on. As compared to the S8ER reflector, the thinner reflector on the S8DS was estimated to reduce system multiplication factor by 1%. Critical mass for the system was estimated at 4.63 kgm U^{235} . Several other calculations were completed to determine the effect of planned or proposed design changes from the S8ER.

b. Internal Reflector Design

Nuclear calculations were performed to determine the reactivity loss in the S8DS due to the design changes in the internal reflectors. Calculations were also made of a proposed D-type internal reflector design which was discarded due to reactivity loss. The calculations also showed that there was little reactivity change due to increasing the steel cladding thickness from 0.010 in. to 0.018 in. The results are listed below.

1) Loss due to changes from S8ER insert	0.05% $\Delta k/k$
2) Loss due to D insert vs new insert	0.68% $\Delta k/k$
3) Loss due to D insert vs S8ER insert	0.73% $\Delta k/k$

The possibility of using solid nickel internal reflectors was also explored but was not used when it was observed that the nickel decreased both the multiplication factor and control drum worth by an amount greater than $2\%\Delta k/k$.

c. Effect of Low N_H Fuel Rod

Some effort was made to investigate the effect of the use of thermocouples on some S8DS fuel elements. However, some hydrogen loss could be expected due to losses around the thermocouple leads. Thus, the fission density

and, therefore, the power would be lowered. The power density at the center of a central rod of $N_H = 4.0$ was found to be 9% low. Power density in the surrounding row of fuel rods was lowered by an average of 3%. Reactivity loss was about 1/10 of 1%.

d. Vault Worth

Previous conclusions that the S8DS vault wall worth was about 0.6 to 0.8% $\Delta k/k$ have been reviewed as errors were found in the experimental vault worth of the SER. Since the estimates of S8DS vault worth were made, in part, from the SER vault wall worth, there is currently no established figure for the S8DS vault wall worth.

e. Post Shutdown Reactor Power Due to Photoneutrons

The reactor power caused by fissions from photoneutrons after shutdown was calculated as a function of reactor operating power prior to shutdown, operating lifetime, time after shutdown, and control drum position.

3. Computer Code and Input Parameter Development

Development continued on new computer codes and cross sections. Two new two-dimensional transport theory codes have been incorporated into the nuclear analysis repertoire; the 2DXY code in cartesian coordinates and the TDC code in cylindrical geometry. Four-group cross sections have also been obtained for use with these computer codes.

B. THERMAL ANALYSIS

1. Fuel Developmental Vibration and Shock Testing

An analytical model of the vibration response of a SNAP 8 fuel element has been developed. The analysis depicts a fuel element which has piston damping pins at both ends and is immersed in an incompressible real fluid. The grid plates are rigid members and the distance between them is 1/16-in. greater than the length of the fuel elements. For a sinusoidal input to the grid plates, the analysis showed that up to ~ 4 cps there was no relative motion between the fuel element and the grid plates; between 4 and 7 cps there was $\sim 1/64$ -in. relative motion between the fuel element and grid plates and the response acceleration of the fuel element was almost constant. Between ~ 7 and 15 cps, the fuel element was moving the full 1/16 in. between the grid plates with the fuel

element response acceleration amplified $\sim 10\%$ compared to the input acceleration. Between ~ 15 and 50 cps the relative motion of the fuel elements and the grid plates decreased until at ~ 50 cps it was essentially zero.

Future work will include the modification of the analytical model to the S8DRM-1 configuration of one damping endpin, and the determination of its response for various clearances between the piston endpin and the grid-plate hole. The results of the analytical model will be compared to an equivalent vibration test.

2. Hydraulic Tests

Since the seven thermocouples located in S8ER are positioned approximately $1/4$ in. above coolant outlets in the upper grid plate, it is believed that the thermocouples do not determine the temperature of the NaK at the coolant outlet holes but report a mixed mean temperature of the NaK in the vicinity of the thermocouples. If thermocouples are to be inserted into the S8DS, a method should be found to ensure that the temperature determined by each thermocouple is the NaK temperature at the outlet. One method to provide this assurance is to insert the thermocouple into an enlarged grid plate coolant hole such that it would not affect the coolant flow shaping. A hydraulic experiment was run to determine the diameter and depth for enlarging the coolant hole in a grid plate. The basis of the test was that the pressure drop across the grid plate would be the same for a $1/8$ -in. diameter thermocouple inserted $1/8$ in. into the grid plate as for an unmodified coolant hole. The test was run for equivalent flow rates corresponding to 300- and 600-kwt core power and showed identical results. For the given configuration, the axial location of the thermocouple could vary by $\pm 1/64$ -in. without affecting the results.

Proposed hydraulic work will be a test to obtain a quantitative effect of cross flow in the core. By replacement of a fuel rod with a heater and thermocoupling various outlet coolant holes, the effect of a thermocouple at a fuel rod, reporting the heat generated by that rod, can be determined.

3. Effect of Fabricating Fuel Elements in Vacuum

Due to the possibility of electron-beam welding the S8DRM-1 fuel element cold endcap before blending, the presence of an initial gas gap pressure of 10^{-6} mm Hg was analyzed. The analysis showed that there were no adverse effects on fuel element operation due to an initial vacuum in the gas gap.

4. Recalculations of Fuel Swelling Due to Radiation Damage

The volumetric expansion of unmodified SNAP fuel is being recalculated using the latest mass spectrometry values for burnup of irradiation test capsules 53-1 and 53-2 and a more realistic capsule temperature history. To determine the capsule temperature the simplifying assumption is used that lower temperature irradiation followed by higher temperature irradiation is equivalent to the same amount of fuel burnup at the higher temperature. Also, it is assumed that higher temperature irradiation followed by lower temperature irradiation is equivalent to the summation of the burnup at each temperature.

5. Fuel Element Cladding Survey

A survey is being performed to determine the effects of various factors on fuel element performance during the reactor lifetime. The factors include hydrogen redistribution due to axial temperature variation, fission gas product buildup, burnup swelling of fuel, NaK corrosion of the cladding, hydrogen dissociation pressure for various (H/Zr) effective and average fuel temperatures, core power level on fuel temperatures, and cladding stress and creep rates for various cladding temperatures.

6. S8ER System Heat Losses

It has been estimated that the heat losses in the primary loop of the S8ER will be 53 kw. The system analyzed included the reactor and coolant lines to and from the primary heat exchanger located in the equipment vault.

Average temperatures assumed were 1200°F for the core, 1100°F for the reactor coolant inlet line, and 1300°F for the outlet pipe. The emissivities used in the calculations were 0.6 for the core, 0.7 for the beryllium reflector, and 0.9 for the reactor pit wall.

The coolant lines in the pit were assumed to be covered with 5 layers of reflective stainless steel insulation. An overall heat transfer coefficient of 2.91 Btu/hr-ft²-°F was used in the calculations. The heat transfer coefficient from the insulation to the helium atmosphere was taken as 8.4 Btu/hr-ft²-°F.

A 3-in.-thick combination of Kaylo and Kaylo-20 was used as insulation for the coolant pipes in the equipment vault, which has a nitrogen atmosphere.

Here a thermal conductivity of between 0.04 and 0.07 Btu/hr-ft²-°F was taken for the insulation and the heat transfer coefficient to the nitrogen was assumed as 6.75 Btu/hr-ft²-°F.

Of the 53 kw, only 11 are lost directly from the core. The nuclear radiation from the core which gives rise to heating in the reflector and the pit wall has not been considered. Such losses were not included in the calculated amount of sensible heat generated in the reactor.

7. S8DS System Heat Losses

It has been estimated that the heat losses in the S8DS primary loop will be in the neighborhood of 20 kw. The system includes the reactor and coolant lines to and from the primary heat exchanger located in the equipment vault.

Average temperatures were also assumed to be 1200°F for the core, 1100°F for the reactor coolant inlet line, and 1300°F for the outlet pipe. The emissivities used in the calculations were 0.6 for both the core vessel and the reactor pit wall and 0.7 for the beryllium reflector.

The coolant lines in the evacuated pit were assumed to be covered with 5 layers of reflective stainless steel insulation. The overall heat transfer coefficient of 1.45 Btu/hr-ft²-°F and a surface emissivity of 0.2 were used in the calculations.

A 3-in.-thick combination of Kaylo and Kaylo-20 insulation was again assumed for the coolant pipes in the equipment vault, which has a nitrogen atmosphere. A thermal conductivity of between 0.04 and 0.07 Btu/hr-ft²-°F was taken for the insulation. The heat transfer coefficient to the nitrogen was assumed to be 6.75 Btu/hr-ft²-°F.

Of the 20 kw only 4-1/4 kw are lost directly from the core. The nuclear radiation from the core which gives rise to heating in the reflector and the pit wall was not considered in this analysis.

8. Modified 30-Node Transient Heat Generation Code (THGC)

The original 30-Node THGC has been progressively modified to best suit the needs of SNAP reactors in general and SNAP 8 in particular. The latest version considers bursting of the fuel elements and provides for more efficient inclusion of decay heat generation.

The old code considered only conductive and radiative heat transfer between nodes. Although it did allow for heat generation within the nodes, the old code was clumsy in handling time-varying heat generation and an iterative process with resubmission of case after case was necessary to handle decay heat generation.

The present modified version easily considers decay heat as it permits inclusion of a table of decay power vs time as input data. This option may be used in accident studies following a scram where heating is by fission product decay. Should a scram not occur, another option allows the reactor to continue operating at full power.

Another feature of the new version is the provision for the burst of the fuel elements. An input temperature tells the computer when to burst the group of rods within a node. Furthermore, upon bursting, the rods are permitted to be cooled by the release of hydrogen due to dissociation of the fuel mixture. The rate of cooling is included as an input table of heat removed by the hydrogen vs temperature of the burst node. Although the rods initially rupture at the burst temperature, they rapidly decrease in temperature with a consequent decrease in hydrogen loss.

An additional option to the code provides for the problem to be cut off when a specified percentage of hydrogen has been released through the burst of the rods. This option is useful in an accident in which the reactor is initially at full power. The hydrogen loss at cutoff can be that amount required to shut down the reactor. A followup case can then include a decay-heat table as with a scram case.

C. SHIELDING ANALYSIS

1. Radiation Heating in Structural Materials

Volumetric nuclear heat generation rates in 316 stainless steel and in Haynes 25 were calculated and transmitted to Aerojet General Corporation (AGC). The heating rates were presented as a function of position outside the reactor-shield assembly.

2. Radiation Damage to Organic Coolant in Radiator

Estimates were provided to AGC of radiation damage to the organic coolant in the proposed 3-fluid SNAP 8 electrical generating system (EGS). The analysis was

based on two radiator designs which were deemed to represent radiation damage extremes. The first design consisted of a conical radiator about 49 ft in length. This radiator was located entirely behind the shadow shield, which was designed to protect the 260-in.-diameter payload. The second design was a 260-in.-diameter cylindrical radiator about 29 ft in length and essentially unshielded from the reactor. The resulting decompositions of diphenyl ether in these radiators following a 10,000-hr reactor operation at 600 kw were 1.4 and 9.7%, respectively.

3. Shield Configuration Studies

An evaluation was made of the shield requirements for six AGC SNAP 8 radiator designs. Fast neutron doses at the payload of 10^{13} , 10^{12} , and 10^{11} nvt were considered. Corresponding lithium hydride (LiH) shield weights of about 500, 700, and 900 lb were obtained for the conical radiator design. Shield weights for the other five designs were grouped around 1800, 2800, and 3900 lb, respectively, for design doses of 10^{13} , 10^{12} , and 10^{11} nvt.

4. Radiation Levels Due to Neutron Source

A determination was made of radiation levels during the prelaunch operation of positioning a polonium-beryllium (Po-Be) neutron source near the subcritical reactor. If a man remains 3 ft from both the source and the reactor, he will receive 6 to 14 mrem/hr/c corresponding to a k_{eff} of 0.94 to 0.99.

5. Relocation of Control Drum Actuators

To reduce radiation damage to the control drum actuators, an alternate concept for relocating the actuators behind the shadow shield was evaluated. Since this relocation entails penetration of the shadow shield by the control drum drive shafts, a consequent increase occurs in the radiation dose at the payload. In order to limit this increase to 10%, specific recommendations for the drive shaft design were made.

6. Scattering from Reflectors

A reevaluation was made of the fast neutron dose outside the SNAP 8 reactor due to scattering in the reflector. The primary purpose of this investigation was to obtain a more accurate representation of the pseudoreflector-source formulation employed in the ray-tracing SNAP shielding codes (FARSE, SCARF, and SCAR). Independent estimates of the fast neutron dose from the unshielded reactor were made on the following bases: (1) leakage data from

neutron economy studies, (2) leakage data from Monte Carlo analysis, and (3) direct evaluation using the ray-tracing codes. The resulting estimates agreed within 28%. The final estimate of the unshielded dose at the 15-ft distant payload following a 10,000 hr reactor operation was 1.46×10^{14} nvt/kw to 0.63×10^{14} nvt/kw from fission neutrons and 0.83×10^{14} nvt/kw from fast neutron scatter in the reflector.

7. Effective Energy Deposition per Fission

A study was initiated to define the actual thermal power level in the SNAP 8 reactor, as distinguished from the thermal power (normally 600 kw) delivered to the power conversion system (PCS) by the NaK. The former power level is greater than the latter by the thermal radiation losses from the surfaces of the reactor and the NaK pipes into the PCS. The corresponding quantity of interest in shielding and reactor physics studies is the actual fission rate. This is related to the actual thermal power by the amount of energy released per fission which is deposited in the reactor. Preliminary analysis indicates a total energy absorption rate in the SNAP 8 reactor (minus shield) of 191 Mev per fission. This value corresponds to a conversion factor of 3.27×10^{10} fissions required to produce 1 wsec of heat in the reactor.

8. Radiation Levels in S8FS Vault

Estimates were made of the postoperation radiation levels in the SNAP 8 Flight System (S8FS) vault due to activation of the concrete enclosure. Results indicate that the dose rate at the surface of the concrete may be as high as 60 mr/hr 30 days after shutdown. The principal radioisotope offenders, iron-59 and manganese-54, result from activation of iron (2 wt %) in the concrete. The AEC contractor for the construction of this facility has been advised to use a concrete mix which conforms to the original AI specification for iron content.

Estimates were made of the fast neutron dose in the vault due to scattering from the aluminum liner and concrete enclosure. A maximum value of 4×10^{16} nvt at the payload was obtained for the worst location of the SNAP 8 testcraft within the vault.

D. CONTROL ANALYSIS

Using the AIREK-II code and an analog model of the primary system,* a study was completed of the startup behavior of SNAP 8. The results show that startup time may be reduced by increasing the neutron source strength and/or increasing the allowable maximum fuel temperature rate during the initial transient. Other methods, such as increasing startup flow rates or additional drum rate changes, may be applied to reduce startup time. These methods will be investigated in future work.

1. Maximum Fuel Temperature Rates

At present, the maximum fuel temperature rate is limited to $150^{\circ}\text{F}/\text{min}$. This maximum rate will dictate the value of the startup rate for any given source. The analog output gives the average fuel temperature rate and these values multiplied by peak-to-average radial and axial power factors give a good approximation to the maximum fuel temperature rates. Figure 1 shows the results of a survey of drum rate vs maximum fuel temperature rates for sources of 10^{-8} to 10^{-2} watts. A click is defined as the total four increments the actuator moves the drum when signalled by the controller. For the analysis, each click consists of one control drum moving 0.5 degrees. The source power is the power of the core when it is 50% subcritical.

From Figure 1 it is seen that the source affects the allowable startup rates. For a fixed source, the startup rate could be increased considerably if the $150^{\circ}\text{F}/\text{min}$ fuel temperature rate were relaxed. For example, for a 10^{-8} watt source level, the present maximum rate is 140 sec/click. If the allowable maximum fuel temperature rate were increased to $300^{\circ}\text{F}/\text{min}$ it would be possible to start up at a rate of about 75 sec/click.

← Figure 1.
Maximum Fuel
Temperature Rates

4-26-63

7568-51176

*S. Birken, "SNAP 8 Reactor Startup," NAA SR MEMO 8084 (CRD), January 4, 1962

2. Startup Rate and Source

Figure 2 is a cross-plot of the previous figure and some additional runs made for a 1- and 100-watt source level. It shows the permissible click rate within which the 150°F/min maximum fuel temperature will not be exceeded.

4-26-63

7568-55135

Figure 2. Startup Rates

3. Peak Power First Excursion

The first transient in power during startup is the largest for a constant drum rate. Figure 3 shows the peak power of the first excursion as a function of drum rate and source level. The peak power increases with increasing drum rate and/or decreasing source level.

4. Time to Startup

Figures 4, 5, 6, and 7 show the time required to reach and stay above the designated outlet coolant temperatures. The last figure shows the time increment to reach the deadband temperature. It should be noted that the time taken to reach a specified temperature is almost independent of source. However, since the criteria of limiting the excursion to a maximum fuel temperature rate limit the drum rate, the startup time is dependent on the source. Figure 8 is a cross-plot of Figures 2 and 7, and shows the startup time as a function of source level.

Figure 3.
Peak Power of First Excursion

4-26-63

7568-55136

Figure 4.
Total Time for Outlet Coolant
to Stay Above 900°F

4-26-63

7568-5498

4-26-63

7568-5499

Figure 5.
Total Time for Outlet Coolant
to Stay Above 1050°F

Figure 6
Total Time for Outlet Coolant
to Stay Above 1150°F

4-26-63

7568-54100

Figure 7.
Total Time to Startup
($T_{\infty} = 1280^{\circ}\text{F}$)

4-26-63

7568-55138

4-26-63

7568-55137

Figure 8.
Effect of Shutdown Power
on Startup Time

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V. COMPONENT DEVELOPMENT

A. FUEL ELEMENT DEVELOPMENT AND FABRICATION

1. Introduction and Summary

The objective of this program is to develop and fabricate fuel elements for the SNAP 8 reactor concept. The concept requires a reactor core that will provide 600 kw to the PCS interface. Thus, the actual power required from fuel elements is approximately 645 kw to provide for system thermal losses and radiation losses. Attainment of this latter requirement is the aim of the fuel element research, development, and fabrication programs.

The fabrication and delivery of the fuel elements for the SNAP 8 Experimental Reactor (S8ER) represented the first platform of the program toward meeting the overall goal. The effort since then has been toward the development and proof of advancements in the technology used for the S8ER core. These advancements are summarized briefly in this section and detailed results are presented under the appropriate subtitles in the following sections.

Additional testing of the AI-8763D glass enamel used for the permeation barrier for the S8ER has been in progress. The endurance data on a cluster of five membranes tested for 6900 hr at temperatures ranging from 1300 to 1500°F show a slight improvement in the barrier during the testing period. These data are in agreement with endurance data on several normal uranium fuel elements assembled for the exploratory testing program, although these elements exhibited sharp noncatastrophic increases due to thermal cycling prior to endurance testing. It has been suspected, but not definitely proven, that the AI-8763D coating will not meet the requirements of the SNAP 8 reactor. Therefore, there has been greater concentration on development of a more thermally stable barrier in which there would be a higher degree of confidence.

Early data illustrated that the four-component SCB glass provided a sufficient permeation barrier, while the data for the other four-component system, SCM, indicated that the SCM type was not satisfactory. Therefore, the effort on the nucleated glasses has been concentrated primarily on the SCB variety. The endurance tests of membranes containing the SCB coating have exceeded 2000 hr at 1500°F with little change in the permeability of the coating. One membrane was also tested for 400 hr at 1900°F to demonstrate the margin of thermal

safety of the new coating. The coating remained an acceptable permeation barrier throughout the testing period, which was discontinued due to a furnace failure. The data to date on the SCB coating have been so successful that the development effort to form final closure seals has been expedited and the coating is tentatively scheduled for the first inpile permeation irradiation experiment to provide complete data for consideration for the S8DS-1 fuel elements.

The data from permeation testing of the coating developed under subcontract at Battelle Memorial Institute (BMI) have also been in the same acceptable range as the SCB coating. However, because of (a) the large number of components in this glass, (b) the success of the SCB coating, (c) the relatively advanced stage of process development of the SCB coating, and (d) the successes in the single refractory oxide coating being developed under subcontract at Stanford Research Institute (SRI), the BMI work will be discontinued. The alumina coating which has been successfully applied in very thin layers at SRI appears to be a better permeation barrier by an order of magnitude. The large amount of process development work still required on this coating has placed it in a backup category for the SCB coating in event of an unexpected failure to meet performance requirements.

Another major development effort this quarter has been directed toward improvement of the final end closure design. The design of the seal cup, cap, and two-piece closure used on the S8ER fuel elements was sufficient as far as initial hydrogen retention was concerned although the requirement for positive mechanical retention of the cup was marginal for shock and vibration requirements of future cores. This effort resulted in the development of a single-piece closure which, on the basis of membrane tests, was statistically as good as a modified S8ER closure. Eight out of ten experimental fuel elements containing the new closure were also successfully assembled for preliminary evaluation. These membranes and fuel elements contained the AI-8763D coating. The initial stages of a program to repeat a portion of the experiments with the SCB coating have been started very recently.

The decision to allow both radial and axial clearance between the fuel rod and the coated cladding in the SNAP 8 fuel element, to accommodate fuel swelling due to irradiation, has caused some concern for the ability of the fuel element to withstand shock and vibration. Though, due to equipment difficulties, vibration and shock data have been obtained on only one SNAP 8 fuel element to

date, design was started on axial holddown devices that will retain the fuel rod during shock and vibration but will allow fuel growth during reactor operation. Two devices, a fixed end column and a hollow ball, appear promising though the fixed end column causes some difficulty in final assembly due to dimensional tolerances and it has been difficult to obtain the hollow ball in the desired wall thickness. To aid the evaluation of the need for holddown as well as the effectiveness of a holddown device, five of ten elements with the new reference closure were assembled with the fixed end column holddown while five were assembled with reference clearances. These elements are being released for testing.

The irradiation program effort was expanded during the quarter as planned. Several new capsules were released for final design: (1) the high temperature, design burnup rate NAA-115 experiment; (2) the low temperature, accelerated burnup rate NAA-116 experiment; and (3) the inpile permeation coating material NAA-77 experiment. The AI-8763D coating irradiation NAA 68-1 experiment and the carbon-modified fuel material NAA-53-3 experiment were 80 to 90% assembled by the end of the quarter. In addition, the research tracer-level post-irradiation annealing work was initiated and several samples were processed. Useful data is not expected from most of these long-term experiments for several reporting periods.

Results on other phases of the research and development in areas of non-destructive testing, performance testing, fuel material studies, and hydride development have been limited. The nondestructive test development for the determination of hydrogen content in the fuel rods and samarium content in the cladding has been proceeding well with procurement of the basic equipment and successful completion of the feasibility experiments. The performance testing program was limited to the exploratory testing phase due to a delay in the normal fuel element deliveries. The equipment procurement and installation for the performance testing program has been proceeding according to schedule. The fuel material and hydriding studies have been limited to the low carbon material (~ 0.15 wt %) except for some dissociation pressure studies made for material of up to ~ 0.50 wt % carbon. Additional mechanical property data on the fuel material were obtained.

The fuel element fabrication during this quarter has been limited because of the difficulty in locating a vendor capable of meeting specifications. The vendor who used a proprietary process of stretch-straightening the cladding tubes for

Figure 9. Chemically Pure, Vacuum Outgassed Al-8763D Coating Aged in H_2 for 1200 Hr at $1500^\circ F$ (500X)

Figure 10. Standard Al-8763D Coating Aged in H_2 for 1100 Hr at $1500^\circ F$ (500X)

the S8ER fuel element has closed his business. Another vendor used a swage-straightening process, but after 23 elements had been processed through assembly, it was found that the tubes grew in length due to stress relieving during permeation testing. A third vendor who uses the original process has now been established and delivery of the first group of 25 tubes has been made.

2. Fuel Element Development

a. Hydrogen Barrier

Development effort on AI-8763D coating was reduced to minor evaluation of vacuum-outgassed material. A program was initiated to measure the thermal conductivity of samples prepared by casting molten glass into heated graphite dies. The cylindrical samples, 0.60 in. diameter by 1.52 in. in length, were submitted to an outside source for measurement.

Experimental AI-8763D coating was prepared by using chemically pure components and by vacuum outgassing the resulting frit. Several lengths of SNAP 8 tubing (both chromized and unchromized) were coated with this material and subjected to hydrogen aging for 1200 hr at 1500°F. The adherence on both substrates improved during aging. The adherence was particularly noticeable on the chromized tubing where there was no adherence on the control specimen but excellent adherence on the aged specimen. Microstructural examination revealed that the coating contained negligible porosity even after hydrogen aging. A comparison of this coating with the standard AI-8763D coating is shown in Figures 9 and 10. Hydrogen permeation tests on vacuum outgassed coating has shown the same barrier quality as standard AI-8763D.

Thermal endurance-permeation testing of the five-membrane cluster containing the standard AI-8763D coating continued. To date, the membranes have been at 1300°F for 1700 hr, at 1400°F for 2000 hr, and at 1500°F for 3800 hr. The latest permeation measurements of the cluster at 1500°F and with 2 atm hydrogen pressure appeared to be stabilized in the range 0.7 to 0.8×10^{-3} cc(STP)/hr-cm². These membranes have also been subjected to 33 moderate temperature cycles from operating temperature to room temperature and 23 pressure cycles from 0 to 4 atm.

A special cladding configuration, in which the AI-8763D hydrogen barrier was sandwiched between two metal cladding tubes, was designed and permeation tested. This experiment was conducted to determine whether the

presence of two metal-ceramic interfaces would markedly reduce the permeation rate. The results, indicated in Table IV, are in the same range as for the normal configuration.

TABLE IV
PERMEATION TEST RESULTS,
SPECIAL CLADDING CONFIGURATION

Time (hr)	Permeation Rate, ϕ , 1500°F, 1 atm H ₂ (cc/hr-cm ²)
4	5.1×10^{-3}
6	2.7×10^{-3}
100	3.3×10^{-3}
178	2.2×10^{-3}

The development effort on the SCM coating (Mg, Al, Si, and Ti oxides) which had been discussed in previous progress reports has been discontinued because of poor hydrogen permeation retention and the success of the SCB coating.

The SCB coating is a four-component glass of Si, Ti, Al, and Ba oxides, which can be crystallized at a high temperature and which will remain stable at reactor operating temperatures. A detailed study of the crystallization of this coating has been completed. The results show:

- 1) Crystallization begins at 1650°F;
- 2) Maximum crystal growth rate is at 1900°F;
- 3) The crystals have a melting point which is greater than 2200°F;
- 4) The glass phase begins softening at 1500°F; and
- 5) The apparent softening temperature of the coating is increased about 100°F by crystallization.

SNAP 8 cladding tubes were successfully coated with SCB without requiring special firing schedules or equipment. A number of SNAP 8 blend cups were coated with SCB. The cups coated well on the top and radius, but the coating tended to spall at the grooves and lip of the cups on cooling. A number of grooveless cups have been successfully coated.

A qualitative study of the adherence of SCB coatings to the metal substrate has been made on short lengths of chromized Hastelloy-N tubing, using a crude flexure test for evaluation. Excellent adherence was obtained when the coated tubes were preoxidized at either 1150°F or 1800°F prior to firing the first layer of the coating at 2250°F. Subsequent layers to form the 0.002-in. coating thickness were fired at 2050°F for 15 min. When the tubes were preoxidized at an intermediate temperature of 1500°F, the coating (bisque) spalled before the samples could be fired at the higher temperature. Preoxidation times of 1/2 to 1 hr gave satisfactory results. Addition of 1 wt% Cr₂O₃ to the coating was explored, but did not improve the adherence. The results of these studies are summarized in Table V.

TABLE V
ADHERENCE OF SCB COATINGS TO METAL SUBSTRATE,
CHROMIZED HASTELLOY-N TUBING

Preoxidation Temperature (°F)	None	1000	1150	1150	1150	1150	1500	1800	1800
Preoxidation Time (hr)		1/2	1/2	1/2	1	1	1/2	1/2	1/2
Firing Temperature	2250	2250	2050	2250	2050	2250	(1)	2050	2250
Adherence after First Layer of Coating	Poor	Poor	Poor to Good	Excellent	(2)	(2)	(1)	-	Excellent
Adherence after Third Layer of Coating	(3)	(3)	(3)	Excellent	Poor	Excellent	(1)	Fair to Good	Excellent

(1) Coating spalled.

(2) No adherence tests made on first layer of coating

(3) Discontinued — no additional coatings applied

The effect of exposure to hydrogen upon adherence of SCB coating was explored by flexure tests on aged and virgin specimens of Hastelloy-N tubing coated with approximately 0.002 in. of SCB. Aged specimens, which had been exposed for 1200 hr to hydrogen at 1500°F, exhibited improved adherence as a result of the treatment. Figure 19 illustrates the microstructure of an aged specimen. The large bubbles near the surface were the result of underfiring of the third layer of the coating.

Figure 11. Three Layers of SCB Coating on Chromized Hastelloy-N After 1200 Hr at 1500°F in H₂ (500X)

The hydrogen retention properties of SCB as a function of temperature are compared with those of other materials in Figure 12. It will be noted that permeability of SCB is comparable to that of Al-8763D coating, with a somewhat higher slope as a function of temperature. Change in permeation rate with time at 1500°F is portrayed in the envelope of Figure 13. The decrease in rate with time is a reflection of increased crystallinity of the coating during aging.

The activation energy for hydrogen permeation can be derived from Figure 14 in which the permeation rate is plotted against $1/T$. Two values for activation energy are involved. Membranes containing a significant fraction of glassy phase (Tests 13, 15, and 18) exhibit activation energies of about 28 kcal/mole, whereas crystallized coatings such as in Specimen No. 28 show a lower activation energy of approximately 16 kcal/mole.

Table VI lists the permeation rates of SCB coating as a function of pressure and temperature. Only one test had been conducted above 1500°F; in this case, a membrane was held at 1900°F in an attempt to observe changes in permeability during the devitrification process. The membrane exhibited a gradual increase in permeation rate with exposure time up to 300 hr, with rates

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Figure 12. Hydrogen Permeation of Solaramic, AI-8763D and SCB Coats as Compared to Bare Metals

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Figure 13. Hydrogen Permeation of SCB Coatings vs Time at 1500°F and 1 Atm Pressure

ranging from 5×10^{-3} cc/hr-cm² at zero time to 9×10^{-3} cc/hr-cm² at 300 hr. After 400 hr, the rate had again dropped to 5×10^{-3} cc/hr-cm².

TABLE VI
HYDROGEN PERMEATION RATES AS A FUNCTION
OF PRESSURE AND TEMPERATURE

Pressure (atm)	Test	H ₂ Permeation Rate, ϕ (cc/hr-cm ² x 10 ⁻³)			
		1200 °F	1300 °F	1400 °F	1500 °F
1/2	13	0.36	0.27	1.13	1.14
1/2	15	-	0.36	0.46	1.02
1	13	0.39	0.73	1.54	3.01
1	15	-	0.44	0.95	1.92
1-1/2	13	0.44	0.73	2.41	3.3
1-1/2	15	-	0.50	1.09	1.95
2	13	0.55	0.91	2.61	4.7
2	15	-	0.61	1.14	2.5
3	13	0.78	1.36	4.05	7.4
3	15	-	0.68	1.89	3.8

Uniform crystallization of SCB coating can be promoted by the addition of a small amount of colloidal platinum (0.01 wt %). The platinum particles in the glass act as nucleation sites for crystals to form. Several SNAP 8 membranes were coated for hydrogen-permeation tests. Test 28 of Figure 14 is one such membrane. Two process problems have been encountered with this coating and they have not been solved completely. First, it has been very difficult to control the amount of platinum that is taken into the glass. Secondly, the platinum induces premature crystallization of the coating during the firing cycle with the result that the coating does not flow and cover well.

The addition of oxides containing elements of small atomic diameter has been investigated as a means of increasing the atomic-packing factor, thus improving the permeation rate. Initial attempts were made by adding lithia (Li₂O) to the SCB system in quantities of from 2 to 10 wt %. A major shift in the SCB composition was required because Li₂O is a very powerful fluxing agent. A eutectic mixture was prepared of BaO · Al₂O₃ · 2SiO₂ (melting point 3270°F) to which increasing amounts of Li₂O were added. This was an attempt to produce

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Figure 14. Hydrogen Permeation vs 1/T for SCB Type Coatings

a lithia-containing glass of suitable refractoriness, while still maintaining a simple four-component system. Table VII lists the compositions and results.

TABLE VII
COMPOSITION OF SCB SYSTEM EUTECTIC MIXTURE*

Composition	Smelting Temperature (°F)	Remarks
BAS ₂ [†] + 2 wt % Li ₂ O	2800	Slight sintering
BAS ₂ + 4 wt % Li ₂ O	2800	Some melting noticed
BAS ₂ + 6 wt % Li ₂ O	2800	Melted to a clear glass
BAS ₂ + 8 wt % Li ₂ O	2800	Eroded Al ₂ O ₃ crucible completely
BAS ₂ + 10 wt % Li ₂ O	2800	Eroded Al ₂ O ₃ crucible completely

*With the addition of increasing amounts of Li₂O

†BAS₂ = BaAlSi Oxides

All of the lithia glasses were also modified by addition of 0.01 wt % platinum smelted-in to serve as a nucleating agent. Results of crystallization studies on the 6% Li₂O glass showed evidence of platinum nucleation, evident by uniform crystallization, but there was little effect on grain refinement since all the crystals were very large (0.005 in.). A long section of this glass, which was crystallized in a gradient furnace, showed crystallization beginning at 1200°F and no apparent softening up to 2200°F. A smelt of 7% Li₂O, produced to gain a lower softening temperature, yielded no promising results. Further effort along these lines was suspended in favor of work on the present SCB composition.

SCB frits are presently prepared by single-smelting of chemically pure oxides followed by a water quench. In an attempt to determine if this procedure is adequate, some SCB glass was prepared by double smelting and quenching on water-cooled copper plates. This new frit appears more dense after smelting, but no comparison of the fired coatings has been made.

Initial studies of the effect of Sm₂O₃ additions to SCB coating indicate that there will be no problem introduced by the poison addition. It appears that the only effect from this addition is to raise the firing temperature of the coating about 50°F.

Thermal expansion studies on standard SCB have been made in the uncrystallized and crystallized conditions. The results, as given in Figure 15, indicate an acceptable match with Hastelloy-N.

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Figure 15. Thermal Expansion of the SCB Coating

At Battelle Memorial Institute (BMI), under subcontract, a glassy ground coat which will not devitrify rapidly at 1500°F has been formulated. To improve adherence, a cover coat to apply to the ground coat has also been developed. This cover coat will be crystallized to provide a good hydrogen barrier. BMI has submitted several coated tubes containing only the ground coat to AI for permeation tests. Permeation test results indicated that the coating was approximately as efficient as the SCB coating. Examination of the coating after 500 hr of hydrogen permeation testing at 1500°F showed that the adherence of coating to metal substrate was excellent in the cold zone but deteriorated slightly in the hot zone. Polished cross-sections showed the coating to be extremely thin (0.6 mil) and the porosity of the coating was essentially zero (Figure 16).

Figure 16. Glassy Ground Coating After 500 hr at 1500°F in H₂ Permeation Test Furnace (500X)

Stanford Research Institute (SRI) is continuing a subcontract study on coating Hastelloy-N with an inorganic material that will remain impermeable to hydrogen at temperatures up to 1700°F. To achieve this goal, the mechanism of hydrogen permeation through metals and inorganic materials and developing techniques of coating Hastelloy-N with inorganic materials such as refractory oxides is undergoing observation.

During this report period, primary research emphasis was placed upon evaluation of thin films of aluminum oxide which were formed on Hastelloy-N by aluminum-chloride vapor deposition and by aluminum-alkoxide pyrolytic vapor deposition processes. When aluminum chloride vapor is reacted with a mixture of hydrogen gas and carbon dioxide, alumina, carbon monoxide, and hydrochloric acid are formed according to the reaction,

It has been determined that a 9:1 ratio of H_2/CO_2 is fairly critical in producing a good dense alumina film. Several good deposits of Al_2O_3 have been produced at SRI by using this hydrolysis method (Figure 17). X-ray diffraction of the type of film shows a crystal size of approximately 100 Å with a highly oriented pattern of alpha alumina. It appears quite difficult to obtain a very thick film by this method as the coating begins spalling at a thickness of 20 microns.

Figure 17.
Photomicrograph of a
 3μ Al_2O_3 Film Deposited
on Hastelloy-N by the
Hydrolysis Method
(700X)

Several different aluminum organic compounds can be used to deposit Al_2O_3 successfully by the pyrolysis method. At present, SRI is studying aluminum alkoxides such as aluminum isopropoxide $\{\text{Al}[\text{OCH}(\text{CH}_3)_2]_3\}$. When this compound is heated, it forms Al_2O_3 , alcohol and water, and the Al_2O_3 is deposited on the hot substrate. The pyrolysis method appears less sensitive to thickness

or geometry than the hydrolysis method but no permeation data are available to date. The as-deposited Al_2O_3 by the pyrolysis reaction has such a small crystal size as to appear amorphous. Heat treatment of this film at 1800°F for 4 hr in argon produced electron diffraction patterns for this recrystallized coating that were identical with patterns obtained from kappa-alumina.

Because of the difficulty of making direct measurements of integrity of thin films, the dielectric properties of vapor deposited films are being used as an index of integrity. Films approximately three microns thick produced by both processes have been evaluated mechanically and electrically with encouraging results. Dielectric breakdown strengths, measured over the surface of 1/2-in. diameter specimens, have been as high as 900 volts/mil. Adherence was demonstrated by bending a coated specimen three times over a 1/8-in. mandrel to an angle of approximately 15° . No noticeable loss of dielectric strength was noted. Thermal cycling resistance was demonstrated by cycling a specimen from 1500 to 200°F five times. The dielectric breakdown strength of the shocked specimen was then measured with no noticeable difference being shown.

A hydrogen permeation test has been run on a Hastelloy-N specimen coated by the aluminum-chloride deposition process. Scatter in the results, due to sample size, made accurate determination of the activation energy for permeation impossible, but results of actual diffusion of hydrogen through Al_2O_3 appeared very encouraging. Tests are in process to determine differences, if any, in permeation when the coating is changed from the high pressure side of the system to the low pressure side.

A vapor-coating system designed for coating the inside diameter of a SNAP 8 tube has been designed and construction is nearing completion. This coating device may be used for both the aluminum-chloride and the pyrolytic vapor-deposition processes and consists of an extension to the existing apparatus and the use of a traveling furnace to control deposition.

Thin films of silicon nitride have been produced on Hastelloy-N and exhibit about triple the dielectric breakdown strengths shown by the Al_2O_3 deposits. No other information on this coating is available to date. Other materials to be tested include zirconium oxide, titanium oxide, and polycrystalline silicon oxide.

TABLE VIII
HYDROGEN LEAK RATE BLEND TESTS
1300°F - 2 atm H₂ (~20 cm² surface at temperature)

Reference Design			Alternate No. 1 Cup			Alternate No. 3 Cup		
Sample No.	Rate (cm ³ /hr)	Grooves on Cup Surface	Sample No.	Rate (cm ³ /hr)	Bad Weld and/or Blend	Sample No.	Rate (cm ³ /hr)	Bad Weld and/or Blend
1	0.97	Yes	1	Off scale	BW	1	0.00	BB
2	0.00	Yes	2	0.04	BW	2	0.00	
3	0.01	Yes	3	0.41		3	No test	
4	0.70	Yes	4	1.09		4	3.80	
7	1.40	Yes	5	No test	BW & BB	5		
8	0.69	Yes	6	No test	BW & BB	6	0.00	
14	0.59	Yes	7	0.82		7	0.03	
15	0.00	Yes	8	No test	BW & BB	8	0.00	
17	0.94	Yes	9	0.04	BW	9	0.00	
20	0.00	Yes	10	0.07		10	0.00	
21	0.00	Yes*	11	Off scale		11	0.00	
22	0.60	Yes*	12	0.07	BW	12	0.00	
5	0.03	No	13	No test	BW & BB	13	0.00	
6	0.00	No	14	No test	BW & BB	14	0.00	
9	0.02	No	15	No test	BW & BB	15	0.00	
10	0.01	No	16	0.76	BW	16	0.00	
11	0.01	No	17	0.01		17	0.00	
12	0.04	No	18	0.82	BW	18	0.02	
13	0.07	No	19	0.00		19	0.07	
16	1.02	No	20	1.59		20	0.09	
18	0.03	No	21	0.66	*	21	0.31	*
19	0.06	No	22	0.32	*	22	0.05	*
23	0.01	No*	23	0.72	*	23	0.04	*
24	0.00	No*	24	1.00	*	24	0.01	*
25	0.00	No*	25	4.00	*	25	0.02	*

*Samples were electron-beam welded

b. End Closure

The development of an improved method of end closure of fuel tubes was expanded during this quarter. Three designs utilizing pressurized empty cladding tubes were fabricated and tested for hydrogen retention. These designs are:

- 1) Reference Design — a blend-cup and endcap combined into a one-piece configuration with the leading edges of the cup entering the cladding tube first to fit over the end of the fuel rod. Both grooved and smooth cups were evaluated.
- 2) Alternate No. 1 Design — a cup similar to the present S8ER blend-cup but slightly longer. The cup is welded to the tube and the end plug is welded to the cup.
- 3) Alternate No. 3 Design — identical to the S8ER blend cup except that grooves were eliminated on the outside surface.

Two other designs, a grooved solid end plug (alternate No. 2) and a plug designed for Magneforming, were not considered because of fabrication problems and delays in component and equipment deliveries. In all cases, closures were assembled using standard SNAP 8 Hastelloy-N tubes coated with AI-8763D ceramic. Five specimens out of 25 for each design were welded via the electron-beam process, and the remainder were welded by the tungsten inert gas (TIG) process.

Evaluation of the relative merit of the three designs was made by hydrogen-leak tests at 1300°F and 2 atm pressure. For this evaluation, the test specimens were equipped with pressure probes to provide an internal source of hydrogen, and the opposite, or test-closure end, was inserted in the test furnace. Hydrogen leak rates, as measured by mass spectrometers, are recorded in Table VIII.

From Table VIII it can be seen that the reference design without grooves and the alternate No. 3 design were superior to the alternate No. 1. In view of the unequal statistical population of 13 samples vs 25, a statistical analysis was made in order to compare the two best designs. Results of this analysis are given in Table IX. It can be seen that little difference exists in the permeation worth of the two designs.

TABLE IX
STATISTICAL ANALYSIS OF BLEND TEST DATA

	Reference Design (No Grooves)	Alternate No. 3 Design
Average leak rate (cm^3/hr @ 1300°F)	0.023	0.016
Standard deviation (cm^3/hr)	0.024	0.026
Lower control limit @ 95% confidence level (cm^3/hr)	0.000	0.000
Upper control limit @ 95% confidence level (cm^3/hr)	0.075	0.069

Additional tests were run to explore the effect of higher temperatures on the blends. Four samples of each design were retested at 1300°F , followed by 1400° and 1500°F . Results reported in Table X indicated that the two designs were of nearly equal value, with the reference design being possibly superior at 1500°F .

TABLE X
PERMEATION RATE AT VARIOUS TEMPERATURES
(2 atm H_2)

Design	Sample No.	Rate (cm^3/hr) 1st 1300°F Test	Rate (cm^3/hr) 2nd 1300°F Test	Rate (cm^3/hr) 1400°F	Rate (cm^3/hr) 1500°F
Alternate No. 3	2	0.00	0.00	0.06	0.21
	22	0.05	0.00	0.00	0.21
	24	0.01	0.00	0.01	0.14
	25	0.02	0.02	0.03	0.07
Reference (No grooves)	11	0.01	0.01	0.06	0.07
	13	0.07	0.00	0.02	0.07
	19	0.06	0.02	0.02	0.04
	12	0.04	0.01	0.05 @ 1455°F	

In view of the favorable permeation resistance, ease of fabrication and lower cost of the reference design, this design was selected for further developmental work.

At present, 10 prototype fuel elements have been assembled to explore the adaptability of the reference blend closure to a production process. The initial results of these tests (Table XI) show that 8 out of 10 would meet the present specification on the basis of permeation rate.

TABLE XI
REFERENCE DESIGN CLOSURE ASSEMBLY DATA

Assembled Element No.	Fuel Rod No.	Average N_H	Average Carbon Content (wt %)	Axial Gap (in.)	Element Permeation Rate (cc/hr @ 1400°F)
599	216-D2	6.01	0.37	0.050	0.72
100-4	231-B1	6.15	0.19	0.050	0.60
86-5	216-C3	5.97	0.38	0.050	0.55
211-4	217-A2	6.01	0.33	0.050	0.62
604	214-E4	5.95	0.46	0.050	4.49
559	232-D4	6.19	0.16	0.0	0.68
603	216-D3	5.94	0.37	0.0	3.60
648	216-D4	5.96	0.36	0.0	0.42
655	217-C2	5.99	0.37	0.0	0.57
645	231-C1	6.17	0.18	0.0	0.16

Both TIG and electron-beam welding procedures have been developed for the final closure of S8DRM-1 elements. The electron beam process has shown several advantages: (1) the ability to make the final seal in a vacuum; (2) the elimination of a weld-cleanup step, (3) the possibility of combining welding and blending operations in the same piece of equipment, and (4) the elimination of argon gas from the inner annulus of the element. Experiments are in progress to develop the electron-beam process for making the endcup-to-cladding tube ceramic seal. Two techniques are being used. The first is to impinge the beam on the cladding tube wall while the tube is revolving, thus heating it to blending temperature. Although this work is in the very early stages, some success has been achieved, as determined by a room temperature helium leak test. The second approach involves the use of a hot tungsten plug or collar as a source of heat. The tungsten plug is placed in contact with the seal zone and is heated by the electron beam. Limited success has been achieved, but again, this work is in its early stages. All the present work is being conducted using the S8ER type end cup for relative comparison.

c. Fuel Rod Restraining Device

Effort was initiated to develop a fuel rod holddown device which would allow radial and axial clearance for irradiation growth and swelling during launch shock, vibration, and acceleration. The criteria for the device is that it holds down the fuel rod by supporting a load up to 30 lb during launch, and that it fail during reactor startup and during long-term fuel growth or swelling by supporting no greater than 50 lb at 1100°F.

Four designs were investigated during the quarter: (1) a fixed end column known as the "collar button" because of its geometry, (2) a hollow collapsible sphere, (3) a slotted thin-wall tube, and (4) a bimetallic hemispherical dome. The latter design was found to be extremely difficult and expensive to fabricate and was eliminated. The preliminary test results shown in Table XII also eliminated the thin wall cylinder design. The hollow ball was not eliminated even though the failure loads were high because the wall thickness of the balls tested was greater than desired.

TABLE XII
RESULTS OF PRELIMINARY TESTS ON VARIOUS
FUEL ROD RESTRAINING DEVICES

Type	Room Temperature Load (lb)	Deflection (in.)	Room Temperature Failure (lb)	1100°F Failure (lb)
Sphere	35	0.003	420	-
	35	0.003	455	-
Thin walled cylinder (slotted)	35	0.002	168	-
	35	-	80	-
Fixed end column (collar button)	35	0.0025	63	-
	35	0.0015	63	-
	-	-	-	24
	-	-	-	32
	-	-	-	31
	-	-	-	24

Further mechanical testing was conducted on a slightly modified collar button device with the results shown in Table XIII. The data are shown for the original dimensions, the loaded and unloaded conditions at room temperature, and 1100°F failure load and stress at failure.

TABLE XIII
COMPRESSIVE TEST DATA FOR COLLAR BUTTON
RESTRAINING DEVICE

Sample No.	Original Diameter (in.)	Length Before Load (in.)	Load at Room Temp (lb)	Total Deflection (in.)	Total Permanent Set (in.)	Rupture Load at 1100°F (lb)	Bucking Stress (psi)
1	0.0323	0.3761	31.5	0.002	0.0002	28	34,200
2	0.0328	0.3756	31.5	0.0019	0.0002	26	30,800
3	0.0330	0.3753	31.25	0.0019	0.0002	29.25	34,000
4	0.0325	0.3752	31.75	0.0018	0.0002	23.75	28,600
5	0.0325	0.3749	33.25	0.0021	0.0005	24.5	29,500
6	0.0330	0.3745	31.25	0.0020	0.0002	24	28,000
7	0.0322	0.3745	30.5	0.0022	0.0005	23.75	28,000
8	0.0324	0.3744	31.5	0.0021	0.0004	24.25	28,000
9	0.0323	0.3752	31	0.0019	0.0002	20.5	24,000
10	0.0322	0.3745	36	0.0019	0.0002	26	30,500
11	0.0326	0.3748	31	0.002	0.0002	27	32,400
12	0.0324	0.3750	32	0.0019	0.0002	24	29,000
13	0.0327	0.3756	31.5	0.002	0.0002	27.75	33,000
14	0.0322	0.3758	32	0.002	0.0002	18.5	22,600
15	0.0324	0.3756	31.5	0.002	0.0002	26	31,500
16	0.0324	0.3745	32	0.002	0.0002	28	34,000
17	0.0325	0.3744	31.5	0.0022	0.0004	18.5	22,400
18	0.0329	0.3752	32	0.0019	0.0002	27.5	32,000
19	0.0331	0.3756	31	0.002	0.0002	33.5	38,500
20	0.0326	0.3746	31	0.002	0.0002	24.7	29,600
21	0.0334	0.3756	32	0.002	0.0002	32	36,800
22	0.0325	0.3758	31	0.002	0.0002	21	26,400
23	0.0334	0.3752	32	0.0018	0.0002	28.5	33,900
24	0.0325	0.3757	32	0.002	0.0002	26	31,400

In addition to the mechanical tests, five of the elements assembled with the reference closure also contained the collar button fuel restraining device. However, due to differential thermal expansion during permeation testing, the device moved and was damaged during the cooling cycle. A minor design revision is being made to correct this problem.

d. Irradiation Testing

The effort on the irradiation testing program was expanded during this reporting period to cover all the new experiments in addition to completing the work on those presently in progress. The progress on the various experiments is discussed under the various capsule numbers.

(1) NAA 53-1 and NAA 53-2 Irradiation Capsules

The uranium isotopic distributions for the unmodified-fuel, accelerated-burnup capsules, NAA 53-1 and -2, were measured before and after irradiation using mass spectrographic analysis to determine the burnup experienced by the NAA 53-1 and NAA 53-2 fuel specimens. The results are shown in Table XIV.

TABLE XIV

URANIUM ISOTOPIC DISTRIBUTION
IN NAA 53-1 AND NAA 53-2 FUEL

	U ²³⁴ (wt %)	U ²³⁵ (wt %)	U ²³⁶ (wt %)	U ²³⁸ (wt %)
Before irradiation NAA 53-1 and -2	1.09	93.25	0.65	5.01
After irradiation NAA 53-1 (upper specimen)	1.19	89.59	3.20	6.02
NAA 53-2 (upper specimen)	1.16	74.75	5.98	18.11

To calculate burnup from these data, several simplifying assumptions were made, in series and parallel: (a) the amount of U²³⁸ was unchanged during irradiation, (b) the increase in the amount of U²³⁶ was due only to non-fission capture of neutrons by U²³⁵, and (c) the decrease in the total amount of uranium was due only to fission of U²³⁵. Using these assumptions singly and in pairs, six equations relating burnup to the isotopic concentrations were derived. Of these six relationships, five depended on α , the ratio of U²³⁵ non-fission captures to U²³⁵ fissions. Since this ratio was dependent upon both time and neutron energy integrated flux actually obtained in the sample, the typical α reported for the MTR was not a satisfactory value. The required time and energy integrated average α for the samples was obtained by solving all possible combinations of the six equations taken two at a time.

Using this method, the average value of α for NAA 53-1 was calculated to be 0.122 ± 0.006 . Using this value of α , the average burnup for the upper fuel sample of NAA 53-1 was calculated to be 0.684 ± 0.008 metal at.%. This value compares favorably to the 0.591 metal at.% value obtained from reduction of the inpile thermal data.

Attempts to calculate α and burnup for the NAA 53-2 capsule have led to the conclusion that the isotopic analysis of NAA 53-2 after irradiation was incorrect. A repeat analysis of this fuel is in process to verify the conclusion. In the interim, a best estimate of the burnup (BU) for the NAA 53-2 was calculated from the relationship:

$$BU_{53-2} = \left(\frac{\text{mass spec BU}}{\text{data reduction BU}} \right)_{53-1} (\text{data reduction BU})_{53-2}$$

The burnup for the lower specimen of NAA 53-1 was also calculated in this manner. The current best estimates for the burnup of the NAA 53-1 and 53-2 fuel are shown in Table XV.

TABLE XV
NAA 53-1 and 53-2 FUEL BURNUP

	Burnup (metal at. %)	
	Lower Specimen	Upper Specimen
NAA 53-1	0.737	0.684
NAA 53-2	1.31	1.45

Metallographic examinations of the fuel from NAA 53-1 and 53-2 have shown the fuel from both capsules to be primarily epsilon-phase zirconium hydride with a small amount of delta phase present. Inasmuch as the original fuel was all epsilon phase, this indicates that the fuel lost some hydrogen during irradiation. Since the lower limit of the delta-plus-epsilon region is 1.65 wt %, this must be the lower limit of postirradiation hydrogen content. Samples from the irradiated fuel were analyzed for hydrogen. These analyses indicated the hydrogen content of NAA 53-1 fuel was 1.62 ± 0.11 wt % and NAA 53-2 fuel was 1.80 ± 0.11 wt %. Both fuels contained 1.74 ± 0.002 wt % H_2 before irradiation.

This discrepancy was not unexpected because of the indicated limits of error for the various analyses. However, it appears from both metallographic examination and hydrogen analysis that the NAA 53-1 and 53-2 fuels lost very little hydrogen.

To obtain information regarding the maximum useful operating temperature of the SNAP 8 fuel, a series of postirradiation annealing experiments using the fuel from the NAA 53-1 and 53-2 capsules is planned. The test apparatus for postirradiation annealing has been designed and built but requires modification since these experiments will be performed in a different hot cell.

(2) NAA 53-3 Irradiation Capsule

Fabrication of NAA 53-3, the first capsule designed for irradiation of modified fuel, is currently about 85% complete. This capsule is scheduled to achieve 0.4 metal at. % burnup at a rate of 3 metal at. % burnup for 10,000 hr, at central core temperatures of 1300°F for the upper fuel assembly and 1500°F for the lower fuel assembly.

The upper fuel assembly has four 1/2-in.- and two 1/4-in.-long specimens of 0.5 wt % nominal carbon, 10 wt % enriched uranium and 90 wt % zirconium hydrided to an N_H of 6.0. The lower fuel assembly contains four 1/2-in. specimens of the same fuel and one 1/4-in. section of original fuel prepared for NAA 53-3 with preferred grain orientation.

Both fuel subassemblies were completed during the week of October 8, 1962. The upper assembly had a measured hydrogen permeation rate of 0.234 cc/hr after 20 hr (1300°F) and the lower assembly 0.663 cc/hr after 49 hr (1300°F). During microbrazing of the inner cap assembly, which contains the NaK, the internal thermocouple of the upper assembly lost continuity; however, two surface thermocouples remain operable.

(3) NAA 68-1 Irradiation Capsule

The fabrication of the hydrogen barrier coating material irradiation capsule, NAA 68-1, progressed during the quarter. Due to the unavailability of straightened membranes, in addition to capsule design changes, several months' delay occurred in fabrication of this capsule. Two production-type membranes with acceptable straightness and permeation rates were recently leased for this test. Four dosimeters have been incorporated in the capsule which will enable

postirradiation determination of both the thermal and fast neutron exposure. The 10^{21} nvt fast exposure specified for this capsule has changed the plans for insertion into a core position instead of the reflector. The new core position effects a gamma heat rate of ~ 5 watts/gm, which will account for $\sim 90\%$ of the heat required to attain membrane temperatures of 1500°F .

In the initial assembly of this capsule, the heaters were damaged and three of the thermocouples were found to be out of place. The heater assembly has since been rewound with new heaters and the thermocouples placed into their proper positions.

(4) NAA-77 Inpile Permeation Test

After review, design was initiated on equipment for inpile permeation testing of coated cladding tubes. This experiment replaces other NAA-68 capsule experiments and has been designed to eliminate possible misinterpretation of the data due to handling damage. Two-dimensional heat transfer studies are in progress, but problems have been encountered with the heat-transfer computer code. Preliminary instrumentation and layout drawings for the experiment were completed and equipment and instrumentation required were ordered.

(5) MTR Multipin Cosine Flux Irradiation Capsule

The preliminary design layout and the one-dimensional heat transfer analysis have been completed for the MTR Fuel Element Engineering Multipin Cosine Flux Irradiation Capsule. The capsule is designed to determine the effect of central temperature, N_H , and carbon content on fuel swelling. The capsule is designed with two intermeshing heaters for temperature control and pyrolytic graphite for a radial heat transfer barrier and axial heater distributor. The capsule will contain eight fuel specimens (2-1/2 in. x 0.560 in. diameter) with two surface thermocouples on each specimen. The uppermost fuel specimen will also contain a central thermocouple.

A product specification has been written for the fuel specimens and assembly drawings have been completed. The fuel alloy (90Zr-10U) has been melted, extruded, and is currently being hydrided.

(6) NAA-116 Materials Development Low Temperature Irradiations

The Materials Development Low Temperature Fuels Irradiation Capsule, NAA-116, was designed during this quarter for the investigation of the effect of structural and compositional parameters on fission gas bubble size and distribution. The fuel specimens will be irradiated at temperatures below 700°F to minimize fission-gas migration during irradiation and avoid hydrogen containment problems. The irradiated specimens will be subjected to postirradiation annealing at temperatures beginning at approximately 1200°F and increasing to temperatures at which the fuel exhibits swelling. The irradiation will be accomplished using two 10-specimen subcapsules contained within NAA-116. The fuel for this irradiation program is currently being prepared.

(7) NAA-115 Fuel Element Engineering Hanford Irradiation

This experiment was designed for irradiating short SNAP 8 fuel elements with various fuel compositions. The experiment was completed through the preliminary design stage during this quarter. The test elements will be of nominal 4-in. length and will have various effective H/Zr ratios and wt % carbon additive. The irradiations will be performed at design SNAP 8 burnup rate at various fuel operating temperatures.

(8) Tracer Level Material Studies

The fuel irradiation program has been initiated with a study of fission gas mobility in the fuel. Low-temperature, low-irradiation tracer studies on thin sections of fuel (5 mil) have begun. Irradiation of five samples of fuel has been completed and the specimens are being annealed and the fission-gas monitored.

e. Performance Tests

The performance testing of SNAP 8 fuel elements has been initiated with exploratory tests on 19 normal S8ER fuel elements. These tests were planned to establish the nature of problem areas which may exist in the fuel-element design.

Three fuel elements were placed on endurance test at 1400°F to replace the four lost during the previous quarter. These elements have completed over 2100 hr of operation. The effect of time at temperature on permeation rate for the SNAP 8 exploratory test elements is given in Table XVI.

TABLE XVI
EFFECT OF TIME AT TEMPERATURE
ON PERMEATION RATE

Cumulative Endurance Time (hr)	Permeation Rate [cc(STP)/hr]		
	E-142-N	E-204-N	E-222-N
0	0.72	1.35	0.44
72	*	1.45	0.67
240	*	1.75	1.56
408	6.04	2.09	2.14
552	4.90	2.27	2.38
662	2.69	1.96	2.17
864	1.34	1.37	1.91
1004	1.18	1.53	1.69
1656	0.36	1.00	1.20
1872	0.51	1.12	1.18
2016	0.58	1.11	1.14
2184	0.59	1.09	1.14

*Permeation rate too high to be measured

The steeply increasing slope plus the high initial leak rates, described by the data for fuel element E-142-N, indicate some defect. The initial increase and subsequent decrease in the permeation rates of E-204-N and E-222-N correspond with data collected during permeation control tests.

Two normal fuel elements were subjected to control tests to establish the effects of the handling and thermal cycling sustained by elements during permeation rate measurements. Fuel elements E-261-N and E-263-N were subjected to a series of permeation tests. At the completion of each test the elements were unloaded and reloaded into the permeation test furnaces. Permeation rates were found to increase in succeeding tests. The elements were then maintained at temperature. This subsequently resulted in decreasing permeation rates. These fuel elements are now undergoing additional permeation test cycles. Data collected to date are shown in Table XVII.

TABLE XVII

EFFECT OF ADDITIONAL TESTS ON PERMEATION RATE, ϕ

E-261-N			E-263-N		
Test Cycle	Cum. Hr @1400°F	ϕ @ 1400°F cc(STP)/hr	Test Cycle	Cum. Hr @1400°F	ϕ @ 1400°F cc(STP) hr
1	17	0.79	1	14	0.51
2	61	0.77	2	54	0.32
3	65	0.90	3	58	0.46
4	99	1.14	4	78	0.54
5	119	1.75	5	82	0.57
6	139	2.03	6	93	0.81
7	143	1.89	7	220	1.00
*	-	2.48	*	224	1.33
*	250	2.61	*	583	1.76
*	308	2.51	*	631	1.62
*	322	2.49	*	677	1.61
*	678	2.48	*	697	1.57
*	726	1.72	*	797	1.63
*	772	1.73	*	822	1.69
*	792	1.78	8	885	1.72
*	892	1.61			
*	918	1.75			
8	980	2.00			

*Indicates measurements made while fuel elements were maintained at 1400°F without cycling.

Three fuel elements were started through a thermal test series: ramp heating, thermal cycling, and startup heating. One of these elements, E-259-N, was a high-permeation-rate element which had been removed from an endurance test. The data on tests completed are presented in Table XVIII.

TABLE XVIII
EFFECT OF THERMAL TESTS ON PERMEATION RATE

Fuel Element	Permeation Rate, ϕ , @ 1400 °F [cc(STP)/hr]
<u>E-259-N</u>	
Acceptance Test	0.74
After 500 hr @ 1400 °F	1.39
After three ramp heats (ambient to 1400 °F @ 50 °F/min.)	10.
<u>E-211-N</u>	
Acceptance Test	1.40
After three ramp heats	1.10
After 46 thermal cycles from 1200 °F to 1300 °F @ 50 °F/min.	1.28
After 54 thermal cycles from 1250 °F to 1350 °F @ 50 °F/min.	1.92
<u>E-166-N</u>	
Acceptance Test	0.49
After three ramp heats	0.49
After 100 thermal cycles from 1250 °F to 1350 °F @ 50 °F/min.	0.47

Tests to determine the effects of vibration, shock, and acceleration on fuel element permeation rate have been started. A small (19 element) fixture was fabricated. One normal fuel element, E-194-N, has been subjected to design level vibration, shock, and acceleration using this fixture. The test was made using 18 stainless-steel dummy fuel elements and E-194-N. The fixture was filled with water and pressurized to 25 psi for the tests. Permeation measurements made on E-194-N after the test indicate no damage was incurred: initial ϕ @1400°F = 0.49 cc(STP)/hr and postexposure ϕ @1400°F = 0.42 cc(STP)/hr. Additional fuel elements will undergo testing to obtain a further indication of the effects of vibration, shock, and acceleration.

12-13-62

7568-5425

8-31-62

7568-1805

Figure 18.
Nine-Furnace
Permeation Test
System

Figure 19.
Thermal Endurance
Furnaces

NAA-SR-8092

A specification which defines the procedures and conditions for the qualification testing of fuel elements for the S8ER, S8DS, S8FS, and S8FR, was issued.

Several systems have been designed and built for testing SNAP 8 elements for the exploratory, qualification, and environmental (developmental) test programs. Since these test programs will run concurrently, use of a test system for any particular program will be determined on a priority basis.

A nine-furnace test system is operational at present (Figure 18). This system has been used for fuel element prototype, blend, and membrane permeation testing.

An additional eight-furnace system was designed and almost all of the equipment was ordered. Construction of the support frame was completed and work on the vacuum system is in progress.

The SNAP 8 fuel element qualification and environmental testing specification requires thermal cycling of fuel elements at transient rates which cannot be obtained with conventional resistance furnaces. Induction heating units, of 15 kw power output, have been ordered for three test systems. Each system will be comprised of an induction unit, inductor coil and retort, vacuum system, temperature control console, and a mass spectrometer for measuring fuel element permeation rate. The equipment, with the exception of two induction units, has been received.

A second endurance furnace system is under construction. Work is progressing on the vacuum system piping and manifolds. This system will essentially duplicate the system which was completed during the previous quarter. Photographs of the first system are presented in Figures 19 and 20. The three endurance furnaces will each accommodate 24 fuel elements, a total capacity of 72. The system under construction has the same capacity.

A new three-furnace permeation test system, shown in Figure 21, became operational during the quarter. This system was designed primarily for permeation testing blend or membrane specimens. Each furnace contains two separate ceramic retorts. Each retort is capable of holding seven specimens, which gives the system a total concurrent test capability of 42 specimens. Each specimen under test can be individually evacuated or pressurized with hydrogen. This ability ensures that, although the specimens may be grouped

Figure 20.
Thermal Endurance
System Controls

8-31-62

7568-1804

Figure 21.
Membrane and Blend
Permeation Test
System

8-31-62

7573-5459

NAA-SR-8092

in a cluster, the permeation of any one specimen can be obtained by pressurizing that specimen with hydrogen while leaving the remaining specimens in the cluster with an internal vacuum. All the permeation vs temperature and pressure data reported in Section V-A-2-b was obtained from this system.

f. Nondestructive Tests

During the reporting period, a study was conducted to check the feasibility of nondestructively measuring the hydrogen content of hydrided fuel rods. The neutron-scatter method was evaluated using an isotopic source of high-energy neutrons rather than the L-77 reactor which was employed previously. The isotopic neutron source was used to permit construction of a hydrogen measuring gage which would be compact, safe, and accurate enough to be used as a process control device. The feasibility of the neutron-scatter method was investigated using a 5/8-curie Pu-Be neutron source, BF_3 neutron detector, decade scaler, moderator of paraffin, and a neutron absorber of cadmium foil. Fuel samples consisting of hydrided and unhydrided normal uranium fuel rods were measured. From the accumulated data, it appears feasible to determine the hydrogen content of hydrided fuel rods by the neutron-scatter method employing a Pu-Be neutron source.

The optimum amount of moderator and shielding around the detector was determined as 1-1/2-in. of paraffin plus a layer of Cd foil. No moderator was required around the source or sample. The greatest difference in total count between the hydrided and unhydrided fuel was obtained with the source sample and neutron detector aligned on the same axis. The source-to-sample distance was not critical over the range of 1.5 to 10 in., although less external reflector scatter occurred at a distance of 3 in. Supporting calculations show that an elongated 10-curie Pu-Be neutron source will produce the desired 0.5% fractional standard deviation with a counting time of 10 to 15 min. Based on the above information, a neutron source and neutron detection instrumentation were ordered to permit assembly of a prototype hydrogen gage.

The development of a neutron activation method of measuring the poison distribution in coated cladding tubes was continued. Special scintillation-type radiation detectors were procured for use inside the previously irradiated coated tubes. Initial tests with these probe-like detectors have qualitatively shown a difference in linear distribution of poison along the length of irradiated tubes.

Experiments have been conducted to measure the gross gamma spectra produced by irradiated coated tubes at varying times after exposure. A preliminary gross gamma radiation decay curve for the tube and coating was established. Gamma radiation spectra from a reference standard tube were determined after each of a series of irradiations to determine whether the gamma spectra changed with time. Additional data in both areas are required to assure the accuracy of the technique.

Preliminary evaluation of an eddy-current thickness gage for measuring the wall thickness of cladding tubes indicates that wall thickness can be repeatedly measured to the nearest 0.0002 in. over the range of 0.008 to 0.017 in. The instrument drift is only about 0.0001 in./hr, which was corrected by periodic checks with reference standards. Two reference standard tubes were fabricated from Hastellon-N by progressive chemical pickling. These will be used for calibrating the low and high ends of the measuring range. The eddy current thickness gage is advantageous over the Vidigage in taking measurements of thin wall tubes because no water or oil couplant is required and the meter on the eddy current gage is easier to adjust and read than the cathode ray tube in the Vidigage.

A program was initiated near the end of the quarter to develop non-destructive test methods and equipment for measuring the depth of weld penetration in the seal endcap weld of fuel elements. This effort is being conducted concurrently with a quality assurance process control technique on the new closure design. To date, the endcap welds of 10 reject S8ER-type fuel elements have been sectioned, polished, and etched. Examples were obtained from these elements of lack of penetration or nonfusion in up to 40% of the tube wall thickness, and of undercut or local tube wall thinning. Deliberately defected samples of endcap welds will be required to determine the feasibility of the nondestructive test method.

g. Fuel Development

Thermal cycling studies on both carbon-modified and nonmodified SNAP 8 fuel compositions have been completed. The radial temperature profile of the fuel specimens was nearly constant during heating and cooling, with a calculated maximum temperature differential between the center and surface of 27°F during cooling. The results show no growth or dimensional instability

after as many as 400 cycles between 390 and 1300°F. No differences in microstructure were noted as a result of the cycling tests. One sample of nonmodified material developed a small crack which terminated within the sample. The evidence indicated that the crack probably occurred during thermal cycling and not as a result of subsequent handling. An x-ray diffraction study to determine degree of residual strains in the material is underway.

Short-time tensile properties and creep behavior of SNAP 8 fuel material have been explored at temperatures near the reactor operating range, and a few tests have been conducted at room temperature. Results of tensile data are shown in Table XIX. It should be noted that, in all cases, the specimens were tested in an atmosphere of hydrogen at equilibrium pressure for the temperature and hydrogen compositions.

TABLE XIX
TENSILE TEST DATA FOR CARBON-MODIFIED FUEL

Temperature (°F)	N _H	Carbon (wt %)	Ultimate Tensile Strength (psi)	Proportion Limit (psi)	0.2% Offset Yield Strength (psi)	Modulus of Elasticity (psi x 10 ⁶)
76	6.5	0.40	8850	3900	-	6.43
76	6.5	0.05	5000	3200	-	7.06
1400	6.0	0.30	6050	1740	3840	4.0
1450	5.9	0.25	7200	4320	3870	-
1500	5.7	0.40	9030	4600	6700	-
1400	6.1	0.15	10000	3000	-	4.0
1400	6.1	0.15	4000	3060	-	4.0

On the basis of creep tests conducted on three specimens, it is apparent that SNAP fuels behave abnormally in creep, exhibiting a cyclic change in creep rate with time. That is, after the normal primary creep period, the samples will creep at a relatively low constant rate for several days and then an increase in the rate is observed which is followed, in a day or so, by a return to the previous constant low rate. During the last quarter, a sample of SNAP fuel containing 0.4 wt% carbon, which was being tested at 1400°F in hydrogen maintained at 19.5 psia pressure and under a tensile stress of 4000 psi, was

observed to be creeping at a rate of about 2×10^{-7} in./in./hr. This rate continued for about 700 hr, after which a transient condition was observed. After the sample had strained about 1.5 mil/in., it returned to the previous creep rate.

As a result of dissociation pressure measurements in SNAP 8 carbon-modified fuel, an equation has been derived expressing the relationship between pressure, temperature, and composition for alloy of 0.15 wt % carbon:

$$\log P \text{ (atm)} = 3.34 \text{ H/Zr} + 2.85 - \frac{8555}{T(^{\circ}\text{K})}$$

This equation is valid for compositions between 1.65 H/Zr and 1.70 H/Zr.

h. Fuel Element Engineering

Design of the S8DRM-1 fuel element was completed. Several changes were made from the S8ER design. End closure configuration was changed from a two-piece grooved, coated, endcup-plus-endplug combination to a single, grooveless, coated cup-plug. This change was based on data presented in Section V-A-2-b. Detail and assembly drawings also provide for the incorporation of either the hollow sphere or collar-button spacers or of no spacer. A selection of one or more of these alternates will be made when further shock, vibration, and acceleration data become available.

Although the present design of SNAP 8 fuel elements allows both radial and axial clearance to accommodate fuel swelling, a crude experiment has been performed in an attempt to determine the effect of fuel rod swelling on hydrogen permeation if there is insufficient initial clearance. Because the fuel grows in diameter during hydriding, in this test the hydrogen permeation of a coated cladding tube was measured while hydriding a fuel slug inside the cladding. Dimensional measurements were made of the specimens before and after the hydriding cycle. A preliminary analysis of the results indicates that a severe breakdown of the ceramic hydrogen barrier will not occur until a diametrical interference of ~ 10 mils exists between the fuel and the ceramic coating. However, this data was obtained over a very short time period and includes no irradiation effects. A more definite test series would have to be conducted before a fuel element design could be established with a criteria for cladding yield to accommodate fuel swelling.

A study was initiated to determine the adequacy of 0.010 ± 0.001 in. wall Hastelloy-N tubing as a cladding for the SNAP 8 reactors. During startup, the stresses acting on the cladding are due primarily to the hydrogen dissociation pressure corresponding to the maximum fuel surface temperature. Considering the calculated temperature profile for the SNAP 8 fuel element, the 0.010-in. cladding is adequate with a safety factor of at least 5. The study for the stresses in cladding as a function of lifetime was not completed. This study includes hydrogen redistribution, fission gas pressure, fuel swelling, cladding creep as a result of hydrogen pressure, fission gas pressure, and fuel swelling, to determine the adequacy of the cladding wall thickness.

A process has been developed and evaluated to chemically remove the chromium oxide scale which forms on the surface of cladding tubes during ceramic coating firing. The process utilizes a bath of KMnO_4 , NaOH , and Na_2CO_3 , followed by a bath of dilute H_2SO_4 and CuSO_4 . For comparison, two processes were used: (1) eleven elements were chemically descaled, and (2) twelve were vapor honed. Following the cleaning processes, the elements were permeation tested. An increase in the permeation rate of the elements was shown following cleaning by each process. Further evaluation and testing is required before accurate statements can be made.

3. Fuel Element Fabrication

a. Fuel Element Deliveries

The fuel element fabrication effort was concentrated on completing the delivery of natural uranium performance test fuel elements of the S8ER design. The problem of obtaining straightened cladding tubes and top-cap subassemblies was not successfully solved until the end of the reporting period. Because of this problem, the scheduled delivery of 55 fuel elements planned for this quarter was delayed by three months.

Delivery of straight subassemblies was interrupted when the vendor who used a proprietary die-straightening process closed his shop. Another vendor used a swage-straightening process which appeared to be satisfactory on the basis of as-straightened dimensions. However, after 23 fuel elements had been assembled with the swage-straightened subassemblies, the elements

were found to increase in length from 0.0006 to 0.033 in. during the 1400°F permeation test. This increase had not been expected as the subassemblies had been held at 1800 to 1900°F for 45 min. during the coating-firing cycle after the straightening operation. Because of the excessive axial fuel rod-to-cladding gap and the unknown effect of the residual stress in the ceramic coating caused by the cladding growth, these elements were rejected for performance testing. They were subsequently used for evaluation of surface cleaning. A third vendor, using the original die-straightening process, started to deliver acceptable straight subassemblies. A detail study of possible dimensional changes in this type of straightened tubing has been initiated; however, no serious problems are anticipated.

Fifteen special experimental fuel elements were fabricated and delivered during the reporting period for physics experiments in conjunction with the S8ER critical experiment. The delivery consisted of two elements, with no ceramic coating on the cladding, each with fuel sections of $N_H = 0, 3, 5,$ and 6 ; two elements with no coating containing short sections of fuel of $N_H = 0$ and 6 which could be interchanged; and five elements with no coating and containing fuel with low wt % uranium.

b. Fuel Element Components

The fabrication of components for final assembly proceeded per schedule during the quarter. The fuel material for 68 normal uranium fuel rods was melted and extruded. The composition of the zirconium-base material was modified slightly to 10.5 wt % U and 0.15 wt % C. The melting and extruding of special fuel compositions for irradiation capsules was initiated during the quarter. The melting and extruding of 10 and 5 wt % U material was completed during the quarter.

The overall melting and extruding process for SNAP 8 fuel was reviewed to determine the feasibility of incorporating a complete third melt to the process. This effort, if successful, will improve the homogeneity of the finished fuel rod.

A program was initiated for recovering scrap zirconium by melting and extruding. The recovered zirconium would be used as getter plugs in the hydriding operation. Both induction and arc melting were successfully tried with an estimated overall savings of 50% from the cost of procuring zirconium bar stock.

Sixteen of the 0.15 wt % C fuel rods were processed through hydriding using a modified cycle. The change in the cycle was required with the change in the nominal carbon content from 0.40 to 0.15 wt %.

The study of determining hydrogen content by weight gain was extended. A correlation and regression analysis of the hydrided enriched and normal fuel rods was made to establish a correlation coefficient between the weight gain and chemical analysis method. At the 95% confidence level, correlation coefficients of 0.278 and 0.895 were determined for the normal and enriched uranium fuel rods, respectively. No explanation for the large difference in the correlation coefficient has been determined at this time.

In addition to the coating of production components with the AI-8763D enamel, approximately 150 developmental components were coated for use in the development program. The AI-8763D coating process has become very routine and no major problems have occurred with either the production or developmental components.

The analytical chemistry of the Sm_2O_3 content of the tubing coated for the S8ER was completed. The average amount of $\text{Sm}_2\text{O}_3/\text{in.}$ in the 215 elements was 3.22×10^{-3} gm compared to a specified value of $3.42 \times 10^{-3} \pm 1.0 \times 10^{-3}$ gm.

The processing of cladding components for the production and irradiation programs continued. A yield of 83% was achieved on a total of 227 cladding and top-cap subassemblies that were welded and inspected. Fifty short subassemblies were also processed for the irradiation program. A savings in time and money was achieved with the modification of the tube-and-top-cap subassemblies from S8ER to S8DRM-1 configuration. This was accomplished by electron-beam welding a longer endpin of larger diameter on the subassembly.

c. Hydrogen Permeation Evaluation

A review of permeation data for SNAP 8 fuel elements was conducted to see if the difference in dissociation pressure of the fuel material due to variation in carbon level could be observed on the finished fuel element hydrogen permeation rate. Table XX shows that this effect is very evident on an average basis:

TABLE XX
AVERAGE PERMEATION RATE FOR VARIOUS CARBON CONTENTS
IN THE FUEL ALLOY

No. of Elements	Fuel	Carbon Content wt %	Average Permeation Rate (cc/hr at 1400°F)
19	enriched uranium	~0.02	0.37
12	enriched uranium	0.15 to 0.20	0.51
50	enriched uranium	0.40	0.48
134	enriched uranium	0.50	0.62
10	normal uranium	0.15 to 0.20	0.09
13	normal uranium	0.30 to 0.40	0.73

Twelve fuel elements which had been subjected to considerable handling during the S8ER critical experiment were returned for re-permeation tests. The average permeation rate of these elements was 0.76 cc/hr at 1400°F before the critical experiments. The average rate after the experiments was 0.67 cc/hr at 1400°F. The average reduction was probably due to the time-at-temperature aging of the AI8763D coating, which has been discussed in previous reports. One of the twelve elements had measurably increased in leak rate although the rate was still within the specified value.

The effect of gross machining on finished fuel elements was reverified on the 23 elements with swage-straightened cladding. Because of the cladding growth during permeation testing, the ends of these elements were machined to return the elements to the specified overall length. The increase in permeation rate as a function of machining is shown in Table XXI. Of 24 permeation rates taken after a machining operation, only three were less than the reading taken before machining.

TABLE XXI
HYDROGEN PERMEATION TEST HISTORY OF NORMAL
SNAP 8 FUEL ELEMENTS

Fuel Element No.	Hydrogen Permeation Rate at 1400 °F (cc/hr)			
	1st Test	2nd Test	3rd Test	4th Test
E-561-N	1.38	-	-	-
E-567-N	0.03	0.21*	0.47*	-
E-579-N	0.24	-	-	-
E-582-N	0.12	0.16*	0.25*	-
E-583-N	0.06	0.09*	0.08*	-
E-584-N	0.05	0.09*	0.26*	-
E-596-N	0.24	-	-	-
E-600-N	0.04	0.26*	0.49*	-
E-601-N	0.20	-	-	-
N-630-N	0.44	0.06*	0.08†	0.06†
N-631-N	1.15	-	-	-
E-575-N	0.23	0.25*	0.57*	-
E-564-N	0.24	-	-	-
E-589-N	1.41	-	-	-
E-585-N	0.45	-	-	-
E-602-N	0.08	0.06*	0.06*	-
E-569-N	0.15	0.10*	0.17*	-
E-644-N	0.94	-	-	-
E-613-N	0.86	0.69*	1.14*	-
E-591-N	0.03	0.14*	0.27*	-
E-611-N	1.74	1.42*	-	-
E-576-N	0.10	0.05*	0.08*	-
E-623-N	0.21	-	-	-

*Retested after machining

†Verification tests of previous tests

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Figure 22. Post Torque and Vacuum Test No. 8,
Control Drum Bearing Set No. 13

B. REFLECTOR AND DRIVE DEVELOPMENT

1. Reflector Bearing

A relatively long-time friction test at temperature and vacuum was completed on a bearing design applicable to the SNAP 8 reflector bearing. This test was conducted to evaluate the performance of the Molykote X-15 dry lubricant (graphite-molydisulfide with sodium silicate binder) on mating cermet surfaces. The mating surfaces of aluminum oxide (Linde LA-2) and titanium carbide-nickel cermet (Kentanium 162B) were coated with the dry lubricant and subjected to torque testing for 1487 hr at 750 and 930°F (two bearing assemblies per test fixture) at 8×10^{-7} Torr pressure during which the bearings were rotated a total of 570 cycles $\pm$ 85° rotation. The frictional characteristic initially was close to that previously exhibited by the carbon graphite-aluminum oxide combination which was very favorable. The friction increased gradually during the test but was still less than that exhibited with this material combination without the lubricant and at much shorter duration. Inspection after the test as shown on Figure 22 indicated that the dry film was still present although slightly worn and burnished in places. Further testing is required to fully evaluate the performance of this and other dry lubricants under the higher space vacuum and also the lower ground test vacuum for 10,000 hr operation.

A test program has been initiated to evaluate the performance of various bearing material combinations and lubricants under the low vacuum (1×10^{-3} Torr pressure) environment of the S8DS-1 system. The combinations being tested or to be tested include aluminum oxide vs carbon graphite, chrome carbide, aluminum oxide, and titanium carbide, and the dry lubricants molydisulfide and calcium fluoride.

Test fixtures are being fabricated to conduct the bearing tests in the new ultra-high clean vacuum facilities now being installed. One of the fixtures is for a basic development investigation of the various parameters such as temperature, bearing loads, contact configuration, sliding velocities as well as material combinations and lubricants.

The other fixture will be used to evaluate the SNAP 8 flight bearing design under simulated load, temperature, and vacuum conditions. Both of these fixtures, as well as the test parts, will be ready for initiation of the test program

when the new vacuum facility installation is completed. In addition, a shock-vibration and thermal shock fixture is being fabricated. This fixture will be completed early in the next reporting period.

2. Control Drive Gear

A test fixture capable of testing the SNAP 8 drive gear configuration and material combination in the ultra high vacuum facility is being fabricated. This fixture is designed so that the gear teeth are subjected only to inertia loads to simulate spatial operation.

3. Springs

Additional René 41 reflector ejection springs were received and are being set up to continue the long-time high temperature performance evaluation. Previously, performance at 1000 and 1100°F under 41,700 to 44,200 psi stress were evaluated up to 1000 hr. The new tests are being set up for higher stresses at 1000°F and lower stresses at 1100°F.

4. Beryllium Structure

A test assembly, which includes one-half of the beryllium reflector structure, one beryllium control drum and shims, has been assembled and is awaiting shock and vibration test. The remainder of the components on the reflector half are in a mass simulation configuration. This assembly will be subjected to flight launch shock and vibration schedule. If tests are completed without failure of this assembly, the vibration spectrum will be increased until the beryllium structures fail or the limit of the test facility is reached. The preparation of the test has been delayed due to excessive beryllium oxide present on the metal surfaces.

5. Beryllium Oxidation

It has been shown that under the temperature and atmosphere conditions to which beryllium metal reflectors may be subjected during ground testing of SNAP 8 reactors, the beryllium metal undergoes a type of accelerating or catastrophic oxidation. Objectives of the basic research study on accelerating oxidation of beryllium, are to: a) measure oxidation rates and duration of induction periods prior to acceleration, and study oxidized surface structures of samples with a range of purity and microstructure from 500 to 900°C; b) formulate mechanisms to describe the oxidation process and causes of surface disintegration

resulting in acceleration; and c) explore possibilities of developing oxidation-resistant beryllium by control of microstructure or purity, or by use of surface treatments.

Further microscopic and electron microscopic evidence has confirmed that pitting of the metal beneath the oxide film during the initial protective period is the mechanism causing metal fragmentation and catastrophic oxidation. Substantial progress has been made in assessing the effects of impurities in the beryllium, in the finding that none of the common impurities appear to be responsible for the catastrophic behavior. Moisture in the corrosive atmosphere has been shown to markedly shorten the protective period prior to acceleration.

a. Microscopic Evidence of Pitting

Oxidized specimens have been lightly polished to remove the oxidized surface in steps, and microscopic examination after each step has confirmed the earlier observation that there are pits in the metal of the order of 5 to 20 microns deep. Also electron micrographs of replicas of the oxide surface have revealed the blisters in the oxide film and have shown their height to be about 2 or 3 microns.

b. Effect of Water Vapor as an Oxidizing Gas

A series of experiments at different temperatures in oxygen saturated with water vapor has yielded the protective periods shown in Table XXII. The

TABLE XXII
EFFECT OF WATER VAPOR
ON BERYLLIUM

Temperature (°C)	Time (hr)	
	Moist Oxygen	Dry Oxygen
600	30	>500
650	20	-
700	10	>150
750	3	25 to 150
800	>1	45

test runs were made in 0.10 atm oxygen and ~3% moisture at 750°C. In a similar run at 1 mm of oxygen +0.03 mm H₂O, or about 1/100 of the former pressure, a linear rate was found after acceleration, and the linear rate persisted for 85 hr, a much more stable behavior than was predicted. Further study will be needed to find an adequate explanation for the more damaging effect of moisture.

c. Effects of Beryllium Oxide Content

It has been found that the inclusion of oxides tends to shorten the time to acceleration as well as increase the rate constant, as shown in Table XXIII.

TABLE XXIII
EFFECT OF OXIDE CONTENT ON
BERYLLIUM OXIDATION*

Source	Oxide Content (%)	Time at Rate Transition (hr)	Initial Parabolic Rate Constant (mg/cm ² /min)
<u>Brush</u>			
S-200,	2.0	25 to 150	6.4×10^{-7}
vacuum-cast	0.33	150	3×10^{-7}
S-200 + oxide	1.25	65	3×10^{-6}
S-200 + oxide	1.50	50	2.3×10^{-6}
S-200 + oxide	2.0	20	8×10^{-7}
Instrument Grade	5.0	2	1×10^{-6}
<u>Pechiney</u>			
CR	0.4	20	7.7×10^{-7}
SR [†]	0.08	100	1.1×10^{-7}

*Oxidation at 750°C in 0.1 atm dry oxygen

†Temperature 775°C

d. Character of Anodized Surfaces

The anodized surface that is being used to protect beryllium from oxidation in the S8ER has been studied after exposure to oxygen for a prolonged period. Although the coating is still providing protection, various microscopic

tests showed it to be very fragile and to consist of a microporous layer of sub-microscopic crystals.

e. Oxidation Behavior of Commercial S-200 Beryllium

A test of several typical lots of S-200 beryllium has shown a variation of time to acceleration from 25 to 150 hr. This indicates the need for a preliminary test of a particular shipment of beryllium before using it in an application where oxidation resistance is needed for a prolonged period. Also there is the possibility that some distinguishing characteristic of the more oxidation-resistant samples can be detected and specified.

6. Flight System Design

A preliminary study was initiated to evaluate a drive system which locates drive motor and the position sensor below the payload shield. Figure 23 shows one conceptual design of this approach. The main advantages are the lower environmental temperature and irradiation requirements for the actuator and the sensor.

The additional complication due to differential thermal expansion and disconnection of the shaft for reflector ejection will require further evaluation. Complete design review and comparison with the current S8DRM-1 design will be conducted.

C. ELECTROMAGNETIC DEVICES

1. Startup Actuators

The purpose of the startup actuator is to provide torque for rotating the reflectors during the reactor startup period. The startup actuators bring the reactor rapidly up to near criticality. This rapid rise is accomplished by simultaneously energizing three startup actuators connected directly through a coupling to the reflector control drums.

Design and fabrication has been initiated (Figure 24). Test fixtures are being designed to perform component development testing on the actuators.

2. Long-Term Actuators

a. Indirect Drive

The purpose of the control drum actuators is to maintain the reactor at operating conditions. This purpose is accomplished by energizing the

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Figure 24. Startup Actuator

actuators which position the control drums and maintain the desired temperature. These actuators must provide control during the life of the reactor. The design chosen for the S8DRM-1 is a stepping type actuator using a slotted, magnetless rotor to produce discrete steps. The rotary motion is reduced by step-down gears to produce the required incremental movement of the reflectors.

During the reporting period, a detail design was completed. Due to the extreme environment under which the actuators are expected to perform, the proper selection and application of materials is a prime consideration.

The present design requires a spring loaded brake to prevent the control drums from rotating during the no-power-on condition. A tractive magnetic brake solenoid is energized to release the spring and allow the slotted rotor to turn. The rotational step is produced by energizing a stator field.

During the continuing development of high temperature wire insulation systems a technique has been developed whereby the insulation encapsulant and the wire are wound onto a solenoid bobbin simultaneously, producing an insulation system which appears far superior to any thus far developed on this project. Further testing on this technique is scheduled. Figure 25 is an enlargement of a sectioned coil. Note the excellent spacing and insulation (fill). A present limitation with the winding techniques currently being used on the stator field is the penetration of the insulation encapsulant into the coils, which are wound previous to encapsulating.

Figure 25. Sectional Coil of SNAP 8 DRM-1 Control Drum Actuator

b. Direct Drive

(1) Linear Actuator

An advanced concept of the control drum actuator is the linear actuator. The linear actuator is connected directly to the reflector control drums, thereby eliminating the gearing required with the present design and is designed to be mounted below the reflector. Designed as a rotary reversible stepping actuator, it will produce the desired angular step of 0.521° upon command signal, and also hold the control drum fixed under no-power conditions and in the absence of a signal. During the reporting period, detail drawings and design calculations have been completed.

(2) One-Half Degree Stepper

The one-half-degree stepper is an inhouse invented and developed control drum actuator. In its present flat configuration, it will be mounted under the reflector and will drive the control drums directly without gears. As its name implies, it rotates in steps of 0.5° . It will incorporate a self-contained conventional electromagnetic brake.

In contrast with other reluctance-operated steppers being developed, all the stator teeth in each pole have the same relationship to their adjacent rotor

teeth and a more efficient design is realized. Other advantages are large pole-face area and shorter magnetic path.

A low-temperature model has been built, and preliminary testing indicates that the principle of operation is sound and that a motor of this type is capable of producing the required torque. However, the test made did not produce the anticipated torque for two reasons: a) the endplay in the ball bearings necessitated an excessive air gap, and b) the parts do not conform to specifications and there is excessive misalignment of teeth.

Experience gained from this proof-of-principle model is being used to redesign the high temperature model. The components of the design are shown in Figure 26, the details of stator assembly in Figure 27, details of stator pole and winding in Figure 28, and Figure 29 shows the detail of rotor (note poor repeatability of teeth which detracted from output torque).

D. CONTROLS AND INSTRUMENTATION DEVELOPMENT

1. Controller

A prototype controller for use on S8DRM-1 was designed, breadboarded, and tested. Detail drawings were prepared and parts procured for the assembly of this controller into a rack-mounted functional prototype. This controller utilizes standard digital modules with transistor output switches. Its basic logic design is identical to that which must be developed in a radiation-hardened version for flight systems.

Development of radiation-hardened components and circuits continued with emphasis in four basic areas:

a. Transistor

The development of the SNAP 10A controller is directly applicable in this area. The basic transistor logic design has been developed (as noted above) in the S8DRM-1 controller.

b. Electromechanical

The possibility of using an electromechanical controller was evaluated and will remain under consideration. However, due to the increased complexity of the SNAP 8 controller, the longer lifetime, and the higher reliability required as compared to the SNAP 10A controller, it is felt that an electromechanical approach will probably not be feasible.

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Figure 26. Components of 1/2° Stepper Control Drum Actuator

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Figure 27. Stator Assembly of $1/2^\circ$ Stepper
Control Drum Actuator

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Figure 28. Stator Pole and Winding of $1/2^\circ$ Stepper Control Drum Actuator

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Figure 29. Rotor Detail of $1/2^\circ$ Stepper Control
Drum Actuator

c. Thermionic Integrated Micromodule

Fabrication and testing techniques were developed for use with thermionic integrated micromodule (TIMM) components. Several circuits were constructed using welding techniques and were tested at their design operating temperature of 580°C. Figure 30 shows a typical circuit. Testing of all components in stock was started so that complete design information will be available. Figure 31 shows the apparatus used in testing triodes. Quads of resistors, capacitors, diodes, and triodes were readied for inclusion in the high flux irradiation program at Battelle Memorial Institute (BMI). These components were assembled on ceramic circuit boards and will be monitored in pile for changes in their operating parameters at 580°C and during exposure to high neutron and gamma fluxes.

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Figure 30. Typical Prototype Controller Circuit

d. Magnetic Core

The logic for implementation of the controller requirements using magnetic cores was developed. A design approach was selected and key problem areas were defined. Basically, magnetic cores will be utilized for the logic with latching relays for power output switching.

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Figure 31. Apparatus for Triode Testing

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Figure 32. Prototype Relay

Transistors will be eliminated as coupling devices in favor of diodes. It is hoped that the diodes can eventually be eliminated by using multi-aperture core techniques. As mentioned above, a latching relay is required for the output switching. This device must switch as a result of the relatively short pulses available from the logic system. A prototype relay (Figure 32) which meets this requirement was designed, built, and tested. The relay will switch on pulses of less than $100\mu\text{sec}$. The two windings, as shown in Figure 32, change the magnetic state of the core, and the armature then follows to the appropriate pole. Transit time of the armature is, of course, much greater than $100\mu\text{sec}$, but is not significant in the controller application.

Further work in this area was devoted towards the development of a two-phase clock, a reversible ring counter, and a test circuit for readout of core states.

2. Programmer

A number of design approaches were investigated during the period. Both electronic and electromechanical devices have certain advantages. Three basic building blocks are required in the programmer regardless of the particular approach taken. These are: a) a stable time-base oscillator, b) a frequency divider network, and c) an output circuit.

a. Electronic

To achieve good oscillator stability, a relatively high frequency, ~ 10 to 100 pulses/sec, is required. To achieve a six-hour program period, the frequency divider circuitry must divide by approximately $100,000$. This would require 18 stages using conventional binary techniques and, for this reason, magnetic core incremental counting circuits were carefully investigated.

The circuit shown in Figure 33 was built and bench tested and is being evaluated for radiation tolerance. The circuit was designed to divide by six. Tests over an ambient temperature range of -65 to $+165^\circ\text{F}$ indicated no change in pulse division ratio. Utilizing a silicon-controlled rectifier to reset the core, the magnetic core circuit is being considered as a primary building block for the SNAP 8 startup programmer. Any number of stages may be cascaded to achieve the desired division ratio. No pulse degradation is experienced as each stage forms a pulse for the following stage. Utilizing this approach, the S8 programmer would require eight divider stages as opposed to 18 , if binary dividers were

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Figure 33. Programmer Circuit

used. The SCR-magnetic core circuit can also be used to provide an output stage as shown in Figure 34.

Effort was directed toward the development of a stable, reliable low frequency oscillator. A number of circuits have been investigated including a magnetic-core-transistor oscillator, a double base diode-RC oscillator, and a vibrating reed oscillator. Of the above oscillators, the double base diode oscillator appears to be the most practical for the SNAP 8 application. Although its accuracy is not as good as that of the other oscillators mentioned above, it is still well within the $\pm 5\%$ timing accuracy requirement and this circuit does have the advantage of relative simplicity over the others. The circuit of Figure 35 was designed and tested over a temperature range of -30 to $+150^\circ\text{F}$. The circuit was found to have a frequency stability of $\pm 4.5\%$ over this temperature range. This stability value could be reduced by proper temperature compensation techniques.

Figure 36 illustrates a possible block diagram for the SNAP 8 startup programmer. A breadboard model is planned.

Figure 34. SCR Magnetic Core Output Circuit

Figure 35. Double Base Diode Oscillator

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Figure 36. Startup Programmer Block Diagram

b. Electromechanical

The primary emphasis in programmer development to date has been on the electronic approach. Initially, the accuracy requirement was $\pm 0.5\%$ which is somewhat beyond the capability of conventional electromechanical techniques. However, with the reduction in the accuracy requirements to $\pm 5.0\%$, the electromechanical approach becomes quite feasible. The environmental stresses and operational life requirements are also within the capability of an electromechanical programmer.

An extensive evaluation program is planned which will include evaluation of complete programmers as well as component parts.

3. Temperature Sensor-Switch

In light of the more stringent temperature control requirements of SNAP 8, an electronic system was chosen as the reference design. The system consists of two-stage magnetic amplifier (see Figure 37) and temperature-sensitive resistance elements. This circuit has the advantage over the previously developed switch of an additional margin of gain to allow for degradation due to long-term operation and radiation effects. The system was breadboarded and component-optimization tests were started.

A backup thermomechanical switch, which utilizes a Bourdon tube to produce switch action, is being designed. Its advantage over previous mechanical switches is that it displays greater sensitivity in contact motion vs temperature changes. The preliminary design is shown in Figure 38.

4. Diagnostic Instrumentation

A position sensor-demodulator system has been developed for SNAP 10A and is being modified for the higher temperature requirement of SNAP 8. These modified sensors, along with prototype demodulators, will be used for the S8DRM-1 system. The sensor is a variable-reluctance, axial air-gap device and the demodulator is a magnetically regulated, transistor-amplifier device. Another type of sensor which is being evaluated for possible higher accuracy capabilities is a radial air-gap wound rotor device. Devices of this nature should eliminate repeatability difficulties experienced due to thermal expansion caused by the wide temperature operating range.

The limit switch to be used on S8DRM-1 is a standard high-temperature switch. However, for the higher vacuum flight environment, a redesigned switch is being obtained. The engineering work on this modification has been accomplished.

Figure 37. Temperature Switch Circuitry

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Figure 38. Temperature Sensor Switch

E. MATERIALS DEVELOPMENT

1. Advanced Spring Materials

Investigations have shown that several alloys appear to be superior to René 41 for spring applications above 1100°F. These are molybdenum, tantalum, or niobium base alloys recently developed for high temperature applications and are commercially available. Of these, T-111, a tantalum base alloy with tungsten and hafnium, shows the most promise on the basis of comparative high-temperature yield strength.

A program has been developed to obtain elevated temperature torsional stress relaxation data on experimental alloys and compare them with René 41 data taken between 1100 and 1500°F. If the experimental alloys are significantly better, springs will then be fabricated for testing.

2. Oxidation-Resistant Coatings for Beryllium

Based upon results from the basic beryllium oxidation studies which showed large variations in oxidation characteristics of S200 beryllium, a new lot of beryllium was obtained for this program. This was cut into coupons 0.100-in. thick and polished in sulfuric-phosphoric-chromic acid. Samples which were anodized and sealed under high pressure in 210°F, 350°F, and 650°F water, respectively, are being tested for oxidation resistance and compared with similar coatings. The oxidation resistance tests were conducted in air having a dew point of -115°F. The results will be compared to evaluation in -40°F dew point air. The test temperature is 1450°F.

It was found that anodizing of beryllium could be accomplished in chromic acid but not in sulfuric acid and that direct-current anodizing was more reliable than alternating-current anodizing.

Metallographic evaluation on anodized coatings with various degrees of oxidation resistance is continuing.

The development of porcelain enamels has been very encouraging.

3. Electrical Insulation

Development of electrical insulation for the stepper motors and electrical brakes is proceeding. The insulation must be capable of encapsulating the motor windings and provide electrical insulation at 1150°F for 10,000 hr. The SAP II

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Figure 39. Effect of Temperature on Insulation Resistance of SAP II

insulation system is a phosphate-bonded silica and has been reported in SNAP 2/10A Quarterly Reports. This insulation system has possibilities for use at SNAP 8 temperatures. Data has been obtained on the effect of temperature on the insulation resistance of the SAP II system and is shown in Figure 39.

One brake each was made of two insulation systems, designated System A and System B. Both brakes were potted by wet winding, and tested at temperature in air. The insulation system components are shown in Table XXIV.

Figure 40 shows a tendency toward greater stability in electrical insulation resistance at 1250°F in air for brake coils made with System B than with System A.

System A contains a glass-coated wire made by using a porcelain enameling technique. The choice of glass that can be fused to the copper magnet wire by porcelain enameling is limited by the melting point of copper (1980°F). More desirable physical, chemical, and electrical properties can be obtained from higher softening point glasses. A machine has been designed and built to apply coatings which soften at higher temperatures than the melting point of copper. The machine is designed to rapidly draw the copper magnet wire through molten electrical insulating material (at temperatures higher than 2000°F). The rate of drawing the copper wire through the molten glass will be rapid enough to keep the temperature of the copper below its melting point.

TABLE XXIV
INSULATION SYSTEM COMPONENTS

System	Insulator to Ground	Encapsulant	Magnet Wire
A	Baked out Macalam	SAP II	Ceramic-Eze
B	Reconstituted mica with aluminum phosphate	Vitreous	Bare Oxalloy

Figure 40.
Electrical Resistance
of Brake Coils
for SNAP 8

5-13-63

7568-55146

Figure 41.
Effects of Temperature
on Carbon Graphite
in Ultrahigh Vacuum

4-12-63

7573-54105

4. Friction Tests

Special tests have been conducted with carbon-graphites as a bearing surface in combination with metals and ceramics. These tests have been conducted in high vacuums at temperatures up to 1000°F. Typical results are shown in Figure 41. Note that the coefficient of friction drops off rapidly as a function of temperature. As reported last quarter, the break point from high to low friction is approximately 600°F; this may be expected with combinations of either ceramic or metal vs impregnated carbon-graphite. For unimpregnated or plain carbon-graphites, the break point still occurs but the coefficient of friction is two to three times higher.

Special tests have been conducted with various dry film lubricants and limited testing has indicated that ceramic bonded MoS_2 or WS_2 will provide low (less than 0.3) coefficients of friction in high vacuum at any temperature from room ambient to 1000°F. The operating life of these dry films is limited by evaporation; however, several tests have been conducted for over 1000 hr with no increase in friction, and very slight losses in weight; i. e., 5% in 1056 hr at 700°F. Typical results are shown in Figure 42 for several tests of dry films with metallic surfaces.

The limitation of the carbon-graphites appears to be the high friction at temperatures below 600°F. The dry films, however, exhibit low friction independent of temperature, though they evaporate excessively at temperatures above 800°F, thereby shortening the low friction life of this system.

It is expected that combinations of these two materials may provide the least amount of friction during startup operations and for 10,000 hr at the service temperatures. Figure 43 is a plot of one test illustrating the results of this system. A carbon-graphite was impregnated with pure MoS_2 powder and then tested to determine the coefficient of friction, μ , as a function of temperature and time. It is interesting to note that the MoS_2 served to provide low friction through the low temperature range but the resultant friction at temperatures above 600°F was above the level established for the same combination without the MoS_2 . Further work is being scheduled in all three of these areas to establish the reason for the increase in friction. Included as Table XXV is a tabulation of the combinations of sliding pairs evaluated during this quarter in screening tests for bearing compatibility at high temperature and in ultra-high vacuums. Some of the materials in Table XXV are shown in Figures 44 through 49 (material friction in high vacuum conditions) and in Figures 50 through 54 (material friction at high temperature conditions).

4-12-63

7573-54106

Figure 42. Effects of Temperature on Several Ceramic Bonded Dry Film Lubricants in Ultrahigh Vacuum Between Metal Flats

4-12-63

7573-54107

Figure 43. Friction Tests of Carbon Graphite Impregnated with Pure MoS_2 Against Metal Flats

TABLE XXV

SLIDING PAIRS EVALUATED IN SCREENING TESTS FOR BEARING
COMPATIBILITY AT HIGH TEMPERATURES AND
ULTRAHIGH VACUUM

Ultrahigh Vacuum Sample Combinations	Surface Finish		Hardness	Maximum μ	Minimum μ	Run Time Cycles	Remarks (All bearing loads are 40 psi)
	Pre	Post					
Inconel-X Inconel-X with LUBECO 6001A	14 3	- -	R15N 74 R30N 36	0.14	0.1	924 contin- uing	Still in test (12-20-62). After three months, the maximum friction level has increased from 0.14 to 0.23 for μ
Al ₂ O ₃ , LA-2 National Carbon, CDJ-83	2 8	10 15	- -	0.32	0.23	648	The graphite surface galled and material transferred to the Al ₂ O ₃ plating. There was no effect from the impregnate indicating complete evaporation had occurred. The at room temperature was 0.58
Tungsten Al ₂ O ₃ , LA-2	1.5 2	- -	- -	0.7	0.54	168	Severe galling occurred and seizure took place after a short dwell.
Ti(6Al-4V) Al ₂ O ₃ , LA-2	2 1.5	- -	R30N 54 -	1.0	0.5		Seizure did not occur after a 92 hr dwell but sliding was erratic - stopped for sample evaluation. Severe galling of the titanium had occurred
Ti(6Al-4V) WC, LW1N	2 2	- -	R30N 54 R30N 26	>1.0	0.48	108	Seizure occurred after 48 hr dwell. Metal transfer and galling had occurred
Purebond, P-5-N with pure MoS ₂ Inconel-X	12-14 2-3	- -	- -	0.34	0.14	309 contin- uing	The MoS ₂ provides for low friction (μ between 0.2 and 0.14) at temperatures below 600°F. Above 600°F μ rose to 0.34, however, after continued operation (200 cycles) has dropped to 0.14, or equal to the minimum.
Purebond, P-5-N Inconel-X	12-14 2-3	- -	- -	0.39	0.21	840 contin- uing	The μ at room temperature was 0.74. Has been running at $\mu = 0.3$ for approximately 500 cycles.
Stellite 6-B Copper with Molykote X-15							Test failed due to sample misalignment. Will be restarted later
Purebond, P-5-N Ti(6Al-4V)	14 7-8	20 2.5	R30T 70 R15N 77	0.25	0.2	264	The μ at room temperature was 0.65 (typical for two tests)
Tests in Air Sample Combinations							Remarks (All dry film coatings are 0.0006 ± 0.0002 in. thick)
Sapphire (single crystal) Ti(6Al-4V)	3-4 3	- -	R30N 54	0.78	0.17	144	Metal transfer and galling occurred.
H-11 Tool Stl. Inconel-X with Molykote X-15	4 4	- -	R _C 47 -	0.57	0.26	2082	Galling and metal transfer occurred as the dry film broke down. Temperature was 910°F, total run time was 24 hr. Seizure occurred after 17 hr dwell. Complete oxidation of the dry film occurred.
Stellite 6-B Ti(6Al-4V) with Molykote X-15	3 3-5	- -	R30N 65 -	0.9	0.19	7	Stopped after 2 hr at 650°F due to high friction. Poor bond of X-15 to the titanium resulted in metal to metal contact. All values of "running" friction were less than 0.28.
Stellite 6-B Inconel-X with Molykote X-15	3 -	- -	R30N 66 -				In test
Stellite 6-B Inconel-X with LUBECO 6002	3 -	- -	R30N 66 -	0.66	0.19	15	Galling and metal transfer occurred as the dry film broke down. Temperature was 550, total run time was 26 hr at 550°F. Reduction in temperature to room ambient reduced the starting friction to $\mu = 0.38$
Stellite 6-B 316 with Molykote X-15	3 -	- -	R30N 65 -	0.86	0.26		High starting friction occurred for each cycle but oxidation of the dry film was minimal with an indicated weight loss of 0.0013 g.

5-14-63

7573-4756A

Figure 44. Al₂O₃, LA-2-CDJ-83 Friction in Ultrahigh Vacuum

5-14-63

7573-4729A

Figure 45. Tungsten - Al₂O₃, LA-2 Friction in Ultrahigh Vacuum

Figure 46. Ti(6Al-4V)–Al₂O₃, LA-2 Friction
in Ultrahigh Vacuum

Figure 47. Ti(6Al-4V)–WC, LW1N Friction in
Ultrahigh Vacuum

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7550-4637

Figure 48.

Purebond, P-5-N-
Ti(6Al-4V) After Friction
Test in Ultrahigh Vacuum,
View A

Figure 49.

Purebond, P-5-N-
Ti(6Al-4V) After Friction
Test in Ultrahigh Vacuum,
View B

4-9-62

7550-4638

SAPPHIRE

5-14-63

TITANIUM

7573-4753A

Figure 50. Sapphire (single crystal) - Ti (6Al-4V) Friction at High Temperature

STELLITE 6-B

5-14-63

MOLYKOTE X-15

7573-4757A

Figure 51. H-11 Tool Stl - Inconel-X with Molykote X-15 Friction at High Temperature

5-14-63

7573-4761A

Figure 52. Stellite 6-B—Ti (6Al-4V) with Molykote X-15
Friction at High Temperature

5-14-63

7573-4760A

Figure 53. Stellite 6-B—Inconel-X with LUBECO 6002
Friction at High Temperature

Figure 54. Stellite 6-B-316 SS with Molykote X-15 Friction at High Temperature

5. Self-Welding

Seventy-one self-weld tests have been conducted to date. Results for this quarter are tabulated in Table XXVI. In general adhesion has occurred with all

TABLE XXVI
COMBINATIONS OF SLIDING PAIRS
EVALUATED IN SELF-WELD TESTS

Self-Weld Tests Sample Combinations	Time @ ~1000°F (hr)	Load (psi)	Time @ ~1000°F (hr)	Load (psi)	Adhesion (psi)	Remarks: All samples have between 4 and 12 RMS finish
316 SS Inconel-X with Molykote X-15	2020	53				
Al ₂ O ₃ , LA-2 WC, LW1N	1954 96	62 62	1540	100	0.12	One test was conducted for 96 hr with 8 temperature cycles from ambient to 1000°F
Inconel-X A-286			720	64	6.7	Metal transfer occurred.
Al ₂ O ₃ , LA-2 CDJ-83	1780	58				
TiC, K162B Al ₂ O ₃ , LA-2	1536	59	1780	68	2.0	Binder material (Ni) fused with the Al ₂ O ₃ surface.
Inconel-X Relay 5 stl.	1258	67				
TiC, K162B Cr ₃ C ₂ , LC-1			1818	58	1.6	
Ti(6Al-4V) Beryllium	172	60	1263	59	0.25	
W W	1166*	67				
Be Micraceram						In test; data to be obtained.

*This test run at 1300°F

Figure 55. Completed Test Can
Being Lowered Into
Safety Container

10-30-62

7568-1827

Figure 56. Test Can Casting Assembly

10-30-62

7568-1832

material combinations tested except those with carbon-graphites. Special ceramic bonded micas are being evaluated and short-term test results indicate no adhesion. Dry film lubricants are also effective in preventing adhesion until they evaporate enough to allow direct contact between asperities. Future testing will involve graphite cloth and pressed graphite foil between contacting surfaces where the prevention of self-welding is desired.

F. SHIELD DEVELOPMENT

1. SNAP 8 Test Can (Two-Region Casting)

a. Fabrication

The two-region SNAP 8 test can* is a simple, right circular cylinder about 23 in. long and 40 in. in diameter. The bottom plate is flat and the top plate is a flattened cone tapering to the filler tube.

The lower region, simulating the Li-7-H portion of the shield, is separated from the upper or natural lithium hydride (LiH) portion by a conical membrane (Note: The casting operation is performed with the shield in the inverted position). Concentric filler tubes connect concentric reservoirs to the upper and lower regions of the test can to supply molten lithium hydride during the freezing cycle.

The test can is fabricated of heavy gage 304 stainless-steel sheet and pipe (0.093 to 0.125-in. thick) for simplicity and strength.

Figure 55 shows the completed test can being lowered into the safety container. Incorporated into the safety container is the structure which supports the test can, reservoir, and associated hardware during the casting operation. The final casting assembly is shown in Figure 56. Figure 57 is a closeup of the supporting framework, the reservoir and the coverplate on the loading access hole, and the thermocouples used for following the temperature cycle. The welded thermocouple feed-throughs are shown in Figure 58. All fittings are welded and leakchecked during assembly.

b. Powder Loading

Upon receipt of the test can at the casting facility, the assembly was visually inspected before initiation of the final assembly. All bolts were coated

*O. P. Steele, III, "SNAP 8 Quarterly Progress Report, July-September 1962, Vol. III, Component Development," NAA-SR-7792 (CRD), May 10, 1963

Figure 57. Supporting Framework,
Reservoir, and Loading
Access

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10-30-62

7568-1831

Figure 58. Closeup of Thermocouple
Welding Fittings

with silver flake to prevent galling, male thermocouple plugs were attached to all thermocouples, and all shipping hardware, clamps, tape, wire, etc., were removed from the assembly prior to raising and mating it with the underside of the powder loading dry box (Figure 59). Figure 60 shows the complete assembly attached to the loading dry box with air operated vibrators in place.

The upper portion of the test can, simulating the Li-7-H region of the SNAP 8 shield, was loaded with 132.6 lb of high purity (97.5 wt %), granulated LiH and the lower portion, simulating the natural LiH region, was loaded with 371.3 lb. These amounts were sufficient to fill the regions with solidified LiH and to provide 10 to 13% excess material to cover such errors as the calculated volume and volume of the filler tubes.

Some difficulty was encountered during the loading operation when the LiH granules failed to feed properly under vibration through the outer feed tube into the test can. However, after some effort, the loading was completed. It is believed that insufficient clearance between the inner and outer feed tubes and insufficient slope in the bottom of the reservoir caused most of the loading difficulty. These deficiencies will be corrected for future casting work.

c. Casting

The lithium-hydride-filled test can assembly was installed on the retort base pedestal and the thermocouples and heater leads were connected. One stainless-steel-sheathed, magnesia-insulated, nichrome heater is used to apply additional local heat to the bottom of the reservoir. The two other heaters supply heat to the filler tube.

The retort was installed on the base, evacuated to about 60 microns pressure and backfilled with purified hydrogen gas to 850 mm Hg pressure. The bell furnace was placed over the retort and the contents heated to 1350 to 1400°F; the melting point of LiH is 1270°F. The furnace was then partially elevated and the test can allowed to cool. Directional cooling is thus accomplished by keeping the upper part of the assembly hot while the lower portion cools.

About 21 hr were required to liquefy the charge of LiH, and 47 hr of cooling were required to produce solidification. Figure 61 shows the heating and cooling cycle indicated by the thermocouple located uppermost in the test can, in the portion which remained liquid the longest time. As may be seen in

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Figure 59. Underside of Powder
Loading Dry Box and
Casting Assembly

Figure 60. Test Can Assembly Ready
for LiH Powder Loading

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NAA-SR-8092

the figure, the heating cycle is much more rapid than the cooling cycle. Once cooling starts, the temperature of the LiH drops to a plateau temperature of about 1270°F which indicates the change of phase from liquid to solid. When solidification is complete, the temperature again decreases.

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Figure 61. Heating and Cooling Cycle on SNAP 8 Test Can Casting

At the end of the cooling cycle (68-hr total), a reheat cycle was made by bringing the entire casting assembly to a uniform temperature of 1200 to 1220°F over a period of 21 hr. This temperature was maintained for 10 hr. At this time, the furnace was turned off and the entire assembly allowed to cool slowly in an attempt to produce a more uniform casting. After 60 hr of cooling, the furnace was removed and the casting allowed to cool for about 46 hr (total 106 hr cooling). The retort then was evacuated, backfilled with argon, unsealed and removed.

d. X-Ray Examination

A total of about 130 hr of cooling were required before the test can was sufficiently cool for x-rays to be made. Nineteen exposures were taken of the specimen to determine the soundness of the casting. No shrink-pipes or

casting voids were observed although a definite void was seen at the metal diaphragm separating the two regions. This void appeared to be 1/8- to 1/4-in.-thick and parallel to the diaphragm. It is possible that this void may have been caused by a shift or "oil-canning" of the diaphragm while the lithium hydride was in a softened condition. A similar void was later found below the top cover plate of the test can which had definitely bulged upwards.

e. Void Volume Measurements

The concentric filler tubes were severed and the void volume determined for the two regions. Table XXVII summarizes the test results.

TABLE XXVII
VOID VOLUME MEASUREMENT

Region	Volume (ft ³)	Void Volume (ft ³)	Void (%)	Full Can Density (%)	Bulk Density (gm/cc)
Li-7 H	2.52	0.1746	6.9	93.1	0.725
Li-n H	6.92	0.4929	7.1	92.9	0.724
Total	9.44	0.6675	7.1	92.9	0.724

f. Thermal Cycle Test

One thermal cycle was applied to the test can by heating it to 1000°F during a 9-hr period. A 5-hr isothermal hold was made, then the furnace was removed and the test can allowed to cool at room temperature.

g. Vibration and Shock Tests

Vibration and shock tests on the SNAP 8 test can followed the post-casting thermal cycle. The test specification was as shown in Table XXVIII. Major resonances occurred at 120 cps (10 g) in the longitudinal direction and at 90 cps (18 g) in the lateral direction.

The large lateral response was measured at top outside diameter of the test can, a distance of about 3 ft from the position where the test can vibration assembly was bolted to the vibration table.

Post-test visual examination did not reveal external structural damage. Results of an x-ray inspection will be reported in the next quarter.

TABLE XXVIII
VIBRATION AND SHOCK TEST SPECIFICATIONS

	Test Duration
<u>Vibration</u>	
Longitudinal*	
1/2 in. double amplitude	5 to 12 cps
3.5 g	12 to 400 cps
7.5 g	400 to 3000 cps
Lateral†	
1/2 in. double amplitude	5 to 10 cps
2.5 g	10 to 250 cps
5.0 g	250 to 400 cps
7.5 g	400 to 3000 cps
<u>Shock</u>	
Longitudinal	
10 g	6 msec
Lateral	
6 g	7 msec

*The vibration sweeps were conducted at a constant octave sweep rate requiring 24 min for the longitudinal direction and 32 min for each lateral direction.

†Two vibration cycles at 90° to each other.

2. Supporting Development

a. Pressure Casting

At present, cast shields have a density of about 93.9% of theoretical density. Shield void volume due to cracks in the LiH and shrinkage from the container walls is an important factor in overall quality of the shield.

The goal of the pressure casting experiment is to obtain a casting with increased density (about 0.74 gm/cc or 95% theoretical density) by decreasing the void volume. This decrease is to be accomplished by the application of a pressure of hydrogen gas over the LiH as the latter changes from the liquid to the solid phase and is in a semiplastic state.

Since the LiH casting facility is designed for a 10-psi gage pressure, the laboratory experiments to date have been limited to this pressure. To increase the pressure differential, the LiH casting is evacuated and backfilled with H₂ gas to a pressure of 1.3 psia. A pressure regulator consisting of a vacuum pump and a 3.55-in. column of Hg maintains this pressure head. The 1.3 psia pressure is slightly larger than the dissociation pressure of LiH at 1400°F. The casting can then be carried out at a maximum operating temperature of 1350°F without loss of hydrogen. The total pressure differential obtained is 23.5 psia.

The test assembly (Figure 62) consists of the test capsule, a reservoir, and a filler neck for the feed system. The capsule measures 5.70 in. long and 3.75 in. OD with a volume of 57.4 cu. in. The reservoir is 3.75 in. OD and 4.5 in. long. The total assembly is slightly longer than 12 in. Thermocouples are placed in the upper, middle, and lower portions of the capsule as well as in the reservoir and neck of the can. Auxiliary heaters, in the form of 0.079-in. diameter thermo-coaxial wire, are wound around the neck and reservoir. The entire assembly is sealed in a gastight retort throughout the casting cycle. The retort is evacuated and backfilled with H₂ under the 1.3 psia pressure until all the LiH is molten. The 23.5-psia pressure is applied as the LiH in the capsule is in the liquid-solid phase change. The neck and reservoir heaters keep a supply of molten LiH to fill the voids left by the freezing LiH in the capsule and the pressure head is maintained until the entire capsule is cooled below 900°F.

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Figure 62.
Pressure
Casting
Test
Assembly

The first test run was unsuccessful due to improper thermal control of the furnace. Additional tests are being prepared.

b. Corrosion Studies

The results of several material compatibility tests of materials of interest to the SNAP 8 shield development program were obtained during the quarter. The tungsten alloys

are of interest for use as gamma shielding and the silicon as a possible hydrogen diffusion barrier. The microbrazed and stainless steel are of interest for structural purposes.

(1) Tungsten Alloys

(a) Kennertium W-10

A 3.5131-gm sample of Kennertium W-10 (0.90 W, 0.075 Cu, 0.025 Ni) showed a weight gain of 0.0603 gm when heated for 48 hr at 1150°F in air. The surface turned light brown during exposure. A specimen of Kennertium W-10 weighing 9.0164 gm showed a weight loss of 0.0014 gm when heated for 120 hr at 1150°F in vacuum. After heating, a red coating was observed on the surface of the metal. Additional samples heated in vacuum at 1160°F for 145 hr showed no visible change.

A 4.0931-gm specimen of Kennertium W-10 exposed to 200°F salt water for 48 hr showed a weight loss of 0.0013 gm. Three specimens have been exposed to LiH at 900, 1200, and 1400°F for 137 hr and have been submitted for metallurgical analysis.

(b) Fansteel

A 5.4548-gm sample of Fansteel (0.99 W, 0.01 Ni) showed a weight gain of 0.0376 gm after 48 hr exposure to air at 1150°F. After exposure, a yellowish film was noted on the surface of the specimen. Two samples heated to 1160°F in vacuum for 145 hr showed streaks of red on their surfaces. A 2.8533-gm sample lost 0.008 gm in weight when heated in 200°F salt water for 48 hr. Samples of Fansteel heated in LiH at 900 and 1200°F for 264 hr showed no surface discoloration. However, a sample heated to 1400°F in LiH for 264 hr showed scaling and pitting.

(2) Silicon with Lithium Hydride

A sample of elemental silicon heated to 900°F for 6 hr in LiH showed no visible change. A sample of silicon heated in lithium for 6 hr at 1200°F reacted with the LiH to form a black layer on top of the hydride. The reaction product reacted vigorously with water to produce flame and gas.

A sample of silicon maintained in LiH at 1400°F for 6 hr reacted to form a uniform reaction product throughout the capsule. This product also reacts vigorously with water. Samples of both reaction products have been submitted for chemical analysis.

TABLE XXIX
MECHANICAL PROPERTIES OF 316 STAINLESS STEEL EXPOSED TO
LITHIUM HYDRIDE FOR 4000 HOURS

Specimen* Number	Exposure Temperature (° F)	Test Temperature (° F)	Ultimate Tensile Strength (psi)	0.2 Yield Tensile Strength (psi)	R/A† (%)	Elongation (%)
C-9-4	900	Room	86,800	46,000	56.2	47.0
E-9-4	900	Room	89,000	39,600	55.8	47.7
C-12-4	1200	Room	96,600	43,400	53.0	36.5
E-12-4	1200	Room	108,900	49,200	43.0	27.0
C-14-4	1400	Room	88,900	35,700	21.6	21.7
E-14-4	1400	Room	95,900	39,000	21.4	22.7
C-9-4	900	400	71,000	31,500	54.9	36.5
E-9-4	900	400	73,200	33,750	53.4	36.0
C-12-4	1200	400	79,900	35,400	46.6	24
E-12-4	1200	400	90,700	43,800	41.3	26
C-14-4	1400	400	74,200	32,500	38.5	25.5
E-14-4	1400	400	80,300	29,900	39.9	24.7
C-9-4	900	800	66,100	25,900	51.0	36
E-9-4	900	800	68,500	27,200	51.9	34.5
C-12-4	1200	800	68,400	29,700	47.7	23.5
E-12-4	1200	800	77,550	35,700	38.1	19.7
C-14-4	1400	800	66,100	24,500	38.5	22.2
E-14-4	1400	800	71,200	27,300	36.8	20.5
C-9-4	900	1200	54,800	22,400	48.9	35.5
E-9-4	900	1200	57,100	25,400	48.9	34.2
C-12-4	1200	1200	57,700	27,000	49.3	26.5
E-12-4	1200	1200	62,850	32,300	42.3	21.7
C-14-4	1400	1200	57,300	22,400	39.1	23.2
E-14-4	1400	1200	60,300	24,250	39.0	23.0

*All specimens 316 stainless steel

†Reduction in area

(3) Nicrobraz 130

A Nicrobrazed joint was maintained in LiH at 1350°F for 114 hr, then removed and cleaned. The joint failed under hand pressure.

(4) Stainless Steel

(a) Mechanical Property Tests

The results of tensile tests on AISI 316 stainless steel exposed to LiH for 4000 hr at 900, 1200, and 1400°F are shown in Table XXIX. Comparison of the exposed and control specimens shows that 1200°F exposure caused the greatest change in mechanical properties. However, the rate of change was much greater between 1000 and 2000 hr exposure (as noted in earlier quarterly reports) than it was between 2000 and 4000 hr exposure.

Figures 63, 64, and 65 show the effects of test temperature and exposure temperature on the ultimate tensile strength, the 0.2% yield strength and the percent elongation of control and LiH exposed specimens.

(b) Weight Loss Tabs

Weight loss tabs of 0.020-in.-thick 316, 347, and 19-9DL stainless steels were exposed to LiH for 1000 hr at 900, 1200, and 1400°F. These data will be reported in the progress report for the following quarter.

Figure 63. Ultimate Tensile Strength of 316 Stainless Steel

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Figure 64. 0.2% Yield Tensile Strength of 316 Stainless Steel

5-10-63

7568-56170

Figure 65. Elongation of 316 Stainless Steel

c. Emissivity Studies

Emissivity coatings for the external surface of the SNAP 8 radiation shield are presently being investigated. The coatings are required to have a total normal emissivity of 0.8 or greater at 800°F and to be stable in a space vacuum for a period of 10,000 hr. Two methods of application are being considered: (1) plasmaspraying, and (2) porcelain enameling.

The coating will be applied to the shield casing after the shield has been cast. However, if a coating is found that meets the specification and does not deteriorate in the hydrogen atmosphere present during casting, the coating would be applied prior to fabrication thus eliminating unnecessary handling and shipping of the shield to outside vendors.

Coatings have been applied to 1-in.-diameter stainless-steel discs. The total normal emissivities of all the coatings are being determined by a radiometer method before and after exposure.

Table XXX shows the results of tests conducted to date and tests that are in progress. In addition to the coatings listed, approximately five other plasmasprayed coatings will be tested.

d. Radiation Damage to Lithium Hydride

A reevaluation of the literature and data on radiation damage to LiH was performed. It was postulated that the swelling which had occurred during the experiments of Pretzel, et al,^{*} had been caused by a depletion of hydrogen (or actually tritium) with the attendant buildup of free lithium stone. When the free lithium atoms agglomerate, the crystal structure changes from face-centered to body-centered cubic with a 28% increase in volume. Thus, there was reason to place a fairly high degree of confidence in the experiments of Welch,[†] particularly when considering that high purity Li⁷H is to be used in the high-dose region of the shield. Investigations are continuing.

e. Dosimetry

The problem of instrumentation for the shield in the S8DS system was studied, and the following plans were formulated: (1) four radial tubes will be placed in different axial locations in the shield (two in each lithium hydride region).

^{*}F. E. Pretzel, et al, "Radiation Effects on Lithium Hydride," LA-2463, August 24, 1961

[†]F. H. Welch, "Summary Report on the Lithium Hydride Development Program for Project 103," TID-6543 (CRD), August 6, 1959

TABLE XXX
STATUS OF EMISSIVITY TESTS

Coating	E_{th}^*	Test [†] Environment	Result	Status
1. Porcelain enamel cerium oxide opacifier	0.85	A	Deterioration of coating	Complete
		B	--	In progress
2. Porcelain enamel titanium dioxide opacifier	0.87	A	Darkening of coating	Post-test emissivity required
		B	Coating intact	In progress
3. Porcelain enamel NBS-418	0.82	A	No change	Post-test emissivity required
		B	--	In progress
4. Titanium dioxide- alumina plasma- sprayed	0.86	A	Deterioration of coating	Complete
		B	--	In progress
5. Chrome oxide plasma-sprayed		A	--	In progress
		B	--	In progress
6. Nichrome plasma-sprayed		A	--	In progress
		B	--	In progress
7. Inconel plasma-sprayed		A	--	In progress
		B	--	In progress

* E_{th} = Total normal emissivity at 800°F - pretest

†Environment A = H₂ at 1300°F (5 days) then vacuum at 800°F (14 days)

Environment B = vacuum at 800°F (14 days)

Each of these tubes will contain a thermal neutron sensitive material (gold or cobalt), and nickel and sulfur threshold detectors for measurement of the fast neutron dose. (2) The fast neutron flux and gamma dose rate are to be measured at the bottom of the shield with appropriate chamber-type detectors. A modified "long-counter" is being considered for the fast neutron measurement and an ionization chamber will be used for the gamma dose rate determination. (3) The shield will contain internal thermocouples from which important data on heat-generation rates will be measured.

f. Liquid-Solid Lithium Hydride Shield

Due to interest in increased power levels, and other factors, it has been decided to study the feasibility of a LiH shield in which the upper (Li⁷H) region would be liquid and the lower (natural LiH) region would be solid with a

layer of insulation separating the two regions. An existing thermal code has been modified to handle this configuration and analytical work will start shortly on required surface emissivity values, insulation values, etc., to maintain the liquid region (top 10 in. of the shield) at 1400°F.

At 1400°F the dissociation pressure of LiH is sufficiently low to allow good hydrogen containment without exotic hydrogen barriers. The expansion compensation required to contain the LiH through the phase change is being investigated. It appears that a single bellows compensator will be sufficient.

Despite the lower material density in the liquid phase, the liquid-solid configuration may still provide a minimum weight shield, assuming that the operating temperature can be maintained with a high degree of reliability, and expansion compensation can thus be simply accomplished. The formation of agglomerations of gas bubbles in a no-gravity condition is a possible problem.

G. SHIELD EXPERIMENTS

This project supported some of the shielding experiments conducted at the SNAP Shielding Test Facility. Included were some of the measurements of fast neutron attenuation made with a Hurst-type dosimeter for LiH shields up to 30 cm in thickness and the measurement of the thermal neutron distribution in a 30-cm LiH shield produced from the fast neutron slowing-down source. The results of these measurements, unnormalized with respect to each other, are given in Figure 66. Both curves have an apparent attenuation length of 4.75 cm. It must be noted that these data should not be used without taking additional geometric considerations into account.

Figure 66.
Centerline Attenuation in LiH of Fast
and Thermal Neutrons from the Fast
Neutron Slowing-Down Source

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Figure 67. Test Parts for Rupture Link

NAA-SR-8092

H. SAFETY COMPONENT DEVELOPMENT

1. Squib Operated Actuator

Preliminary tests were conducted on the SNAP 8 mechanical actuator which is used for unlocking the control drum and also for ejecting the destruct ring at the base of the reactor. The SNAP 8 actuator incorporates an improved ball-detention locking device to lock the piston (plunger or pin) at either end of its stroke. Not only is the fabrication simplified but the plunger can be ground-fired without damage to the locking device.

Static pressure tests using nitrogen supply were used to evaluate actuator performance under the simulated load conditions. Approximately 20 psi is required to release the locking device and an additional 80 to 100 psi is required for the pin-pulling function. The M-130 squibs to be used with this actuator develop at least 1000 psi in the same cylinder volume. The actuators are now being prepared for firing tests with the M-130 squib after soaking at various temperatures and pressures.

2. Thermomechanical End-of-Life Actuator

The SNAP 8 reflector retaining band release actuator utilizes a flexible rupture link which is an integral part of the retaining band. The thermomechanical actuator provides the bending force for rupturing the link.

Test parts of various configurations and materials for the rupture link have been designed and are being procured. Tests are underway to determine the correlation between the calculated and actual force and deflection generation as a function of temperature change of the thermomechanical actuator. Figure 67 shows the test installation on high temperature tensile test equipment. Preliminary data indicate that further refinement in the design analysis is required.

3. Low Energy Input Squib

The performance test of the M-130 squib for use in the control drum unlocking and destruct charge ejecting actuator was previously conducted. The testing determined the limits of "sure fire" and "no fire" and the pressure generation at various volumes. The complete test data are being evaluated and reports are being prepared.

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Figure 68. Ground Command Shutdown Actuator

4. Electrically Operated Fusible Link Actuator

The development of the electrically operated fusible link actuator initiated last quarter has been intensified. This device is being developed as a command shutdown actuator for releasing the reflector restraining band. The design which represents the current status of development is shown in Figure 68.

The actuator consists of two cylindrical parts, one soldered inside the other with high-temperature solder (1225°F). Upon energizing, an electrical heater placed within the inner cylinder raises the temperature above the solder melting point to cause separation of the two cylinders. The two cylinders are under external tensile loads.

Analytical studies have shown that a minimum of 300 to 500 watt-sec of energy is required (depending upon configuration) to raise the actuator temperature from room to solder melting temperature. In order to minimize the electrical energy input, the heater must be designed for optimum efficiency. The main effort to date has been to develop a heater design and fabrication technique. Approximately 60 developmental actuators have been built and tested. Of these, approximately 50% have operated satisfactorily. The failures were due to premature coil burnout. The energy input for activation ranged from 750 to 1400 watt-sec. These test actuators represent the following parameters:

a) Actuator Cylinder Configuration

- 1) 3/16 and 1/4-in. tube diameter
- 2) 3/8 and 1/2-in soldered overlap

b) Heater Wire Size and Material

- 1) 28 gage and 30 gage Nichrome
- 2) 28 and 30 gage Kenthal A-1
- 3) 30 gage tungsten

c) Insulating Material

- 1) PBX
- 2) PBX - Temporell combination
- 3) Beryllium oxide - PBX combination
- 4) Temporell

d) Heater Configuration

- 1) Number of coils from 8 to 30
- 2) Hollow coil with insulation between coils and outside
- 3) Coil completely insulated
- 4) Coil wound on hollow insulating sleeve with potting on outside only.

The tungsten wire has shown the best resistance to burnout. However, the 30-gage wire requires high energy input, and additional models with 32-gage wire are now being built. No definite conclusion on the insulating material or the heater winding configuration has been reached. Since BeO has high thermal and low electrical conductivities, it appears attractive.

A parallel effort on the heater development is being conducted on a limited basis by a manufacturer of commercial miniature heaters.

As soon as the heater development can be advanced to reasonable operational reliability, environmental tests on the complete actuator will be initiated. In addition, compatibility tests will be conducted on the inorganic insulation and the cylinder-joining solder at operational environmental condition.

5. S8DS-1 Scram Mechanism

Fabrication has been initiated on one of the S8DS-1 reflector and drive command shutdown devices described in the previous section. This unit will be assembled and subjected to development tests in the Mark I vacuum facility under S8DS-1 operating temperature and vacuum conditions. These tests are designed to check the operating performance of the mechanism and to determine design improvements as required.

VI. SYSTEM TESTS

A. SNAP 8 EXPERIMENTAL REACTOR (S8ER)

1. Introduction

The first nuclear test in the SNAP 8 development program is the SNAP 8 Experimental Reactor. This will be a test of the core only, without power conversion equipment. Test objectives are to (a) demonstrate reactor performance at both 300 and 600 kwt, and (b) determine reactor performance characteristics. The purpose of the test is to (a) verify the SNAP 8 core design, and (b) provide experimental data on the dynamic behavior of the reactor required to design the automatic startup and control system.

The critical experiment, conducted in a facility especially equipped for physics experiments, was successfully completed. In Section VI-A-2 of this report, results from this experiment have been compared with analytical predictions for critical loading, worth of reflectors and internal core components, and certain nuclear parameters. Permeation testing of fuel elements after completion of the critical experiment showed that hydrogen retention capability had not been compromised by the shipping and handling techniques employed.

Fabrication of the core vessel to be used in power tests was completed, and the vessel was welded to the shutdown shield, which will support the vessel in the containment vessel. The shield will also serve as the mounting base for the external reflector elements which are the reactor control devices. Due to reflector element positioning requirements, the core vessel and shield had to be aligned radially within 0.015 in. Assembly and welding techniques to meet this requirement were checked on two prototype assemblies prior to making the actual joint. Successful completion of this difficult task constituted a significant milestone.

A second functional test of the NaK heat transfer systems demonstrated that the major problems uncovered during the July test had been successfully solved. Checkout of instrumentation continued during the quarter. Bearing materials for the high temperature reflector guide bushing, located near the bottom of the core, have been selected. All parts will be made of Hastelloy-B. The shaft and race will be coated with aluminum oxide; the ball, with chrome carbide.

2. Nuclear Analysis

The S8ER Critical Experiment was completed during this quarter. The results were correlated with analytical predictions. A comparison of experimental results with the calculated values is described below.

a. Critical Loading

Table XXXI lists the critical loadings experimentally determined during the S8ER critical experiment and the calculated loadings using the FAIM diffusion theory code. The error is the theoretical reactivity difference between theory and experiment.

TABLE XXXI
CRITICAL LOADINGS OF S8ER

Effective Beryllium Reflector Thickness (in.)	Beryllium Shims Added	Critical Loading		Calculational Error (% $\Delta k/k$)
		Experiment (Number of Rods)	Theory	
2.34	None	213.2*	196	1.5
3.08	A	185.4*	169	2.0
3.78	A and B	172.2	152	3.0
4.91	A, B, and C	-	161 [†]	-

*Extrapolated.

[†]Predicted for S8ER in Bldg. 010. This value has been corrected for data obtained during the S8ER critical experiment.

The calculated critical loadings are corrected for low hydrogen content and low uranium content in asbuilt fuel rods. A correction was also applied for the presence of the ceramic hydrogen barrier coating, which acts as a poison. The error appears to result from inadequacies of the analytical model, overestimation of the worth of the beryllium reflector, limitations of the FAIM code, and limitations of diffusion theory.

b. Multiplication Factor

The multiplication factor (k_{eff}) for the S8ER with A and B shims, as calculated by the ULCER code and corrected for ceramic, was found to be 1.0565 using hydrogen cross sections designated as HZR and 1.0635 using HZRB

hydrogen cross sections compared to a measured value of 1.0454. Thus the absolute error involved in using ULCER was found to be $+0.0111\Delta k$ and $0.0181\Delta k$ using HZR and HZRB, respectively. The FAIM code was found to give a k_{eff} $0.0075\Delta k$ higher than ULCER.

c. S8ER Reflector Worth vs Thickness

K_{eff} as a function of Be reflector thickness is plotted in Figure 69. The experimental curve is based on measurements of total excess reactivity, shim worth, and critical loading. The theoretical curve is from calculations using the ULCER diffusion theory code.

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Figure 69. S8ER Radial Reflector Worth

d. Worth of Low- N_H Rods

The worth of fuel rods at the center of the core with different values of N_H is shown in Table XXXII. The FAIM code was used to find the calculated values.

TABLE XXXII
WORTH OF LOW N_H RODS
AT CORE CENTER

N_H	Worth	
	Experiment (¢)	Theory (¢)
0	0	0
3	22.6	20.8
5	34.1	35.1
6	39.6	41.6

e. Power Distributions

Axial and radial power distributions are shown in Figures 70 and 71. The experimental results are indicated by data points, and are normalized to the calculated peak power at the center of the core.

f. Kinetic Parameters

The ratio of the effective delayed neutron fraction to the mean prompt neutron lifetime, β_{eff}/ℓ , was measured to be $950 \text{ sec}^{-1} \pm 10\%$, and it was calculated as 1060 sec^{-1} . β_{eff} is assumed to be 0.0077, so the experimental mean prompt lifetime, ℓ , is $8.1 \mu\text{sec}$; whereas, by using the ULCER code, ℓ was calculated as $7.28 \mu\text{sec}$.

g. Worth of BeO Internal Reflectors

Attempts were made to calculate the worth of the internal reflectors using the ULCER and FAIM programs. The results are shown in Table XXXIII.

TABLE XXXIII
WORTH OF BeO INTERNAL REFLECTORS

	Worth		
	FAIM (¢)	ULCER (¢)	Experimental (¢)
Removal of one flat insert	-21.3	-19.5	-12
Removal of second flat insert	-	-21.1	-15
Removal of D insert	-	- 8.7	- 6.3
Removal of all three inserts	-	-47.7	-34.7

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Figure 70. S8ER Axial Power Distribution

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Figure 71. S8ER Radial Power Distribution

The experimental results were then extrapolated by means of ULCER calculations to obtain a total worth for all 18 internal reflectors of about \$4.50.

h. Worth of S8ER Control Drums

The S8ER control drums were calibrated with varying shim configurations (see Section VI-A-2-a). One control drum was calibrated with the A shim only, with A and B shims, and with A, B, and C shims; the results are given in Table XXXIV.

TABLE XXXIV
CONTROL DRUM WORTH

	Total Worth Per Drum
A shim	2.59% Δk /105° rotation
A + B shim	2.83% Δk /105° rotation
A + B + C shim	2.75% Δk /105° rotation

The range from 105 to 137° from the full-in position was calibrated and found to be worth 0.0308% Δk for the drum with the A shim, and 0.185% Δk for the drum with the A, B, and C shims.

Two dimensional transport calculations of S8ER control drum worths (A shim attached), using the 2DXY with four-group cross sections, resulted in a drum worth of 2.7% Δk with no effort to simulate axial leakage in the void and 2.95% Δk with an artificial end leakage term of the DB^2 form. The experimental value was found to be about 2.6% Δk . Previous three-group CURE diffusion calculations had predicted a Δk of 2.45%.

i. Worth of Poison Splines

Three of the experiments to measure the worth of poison splines (proposed for use during core loading) in the S8ER core were calculated using ULCER and a self-absorption factor of the form

$$\Sigma_{a, \text{effective}} = \Sigma_{a, \text{original}} \left(\frac{1}{2\Sigma_a r} - \frac{3}{32\Sigma_a^3 r^3} \right),$$

where Σ_a is the original macroscopic absorption cross section of the material and r is the tube radius. The results are shown in Table XXXV.

TABLE XXXV
WORTH OF POISON SPLINES

Number of Splines	Position	Worth	
		Measured (¢)	Calculated (¢)
6	Clustered around center rod	- 52.5	- 51.1
19	Distributed over core	-146.5	-130.5
6	Distributed between fuel rings VI and VIII	- 34.5	- 40.2

Considering the complexity of these calculations, the agreement seems good.

3. Core Engineering

The weld that joins the S8ER core vessel to the shutdown shield was executed with great care. Because of the reactor control mounting on the core vessel base, the maximum allowable misalignment is 0.015 in. Figure 72 shows the extensive tooling required to maintain and monitor this alignment. Two of the four dial indicators used are also shown in the figure. Six simulated external reflector control drums and mounting shafts were used to ensure clearance and final alignment of the final assembly. After completion of core welding, the NaK inlet and outlet spools were welded to the reactor, as shown in Figure 73.

Two trial welds were made on a prototype assembly before the actual weld was attempted. The first weld was performed using a backup ring (E. B. type). This weld, although externally appearing satisfactory, was unsuccessful due to uneven melting of the E. B. ring during the initial root weld pass. The uneven melting caused an uneven buildup on the ID of the vessel. The second prototype weld was a V-groove joint design with an established gap of 3/32 in. The results of this weld were satisfactory. This procedure and weld design was used to perform the final weld. Besides establishing the proper weld parameters and procedure to maintain alignment, the weld mockup also was used to determine if stress relieving after welding would be necessary. Heating of the first trial welded assembly to reactor operating temperature did not result in significant misalignment of the vessel relative to the shield. The stress relieving of this weld would have been almost impossible.

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Figure 72. Welding the S8ER Core Vessel to the Shutdown Shield

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Figure 73. S8ER Core Vessel Welded to Shutdown Shield
(Coolant inlet and outlet tubes have been added)

The tolerances established for core vessel No. 1 presented a number of fabrication difficulties. Vessels No. 2 and No. 4 were spun to the nominal ID and wall thickness, but, due to subsequent welding and annealing operations, the shell section was distorted. Despite attempts to correct distortion, considerable difficulty was experienced in maintaining the ID to the specified tolerances. Serious consideration was given to the possibility of hydraulically sizing the ID of the vessel. This operation would have involved stressing the shell section beyond the yield point (2% offset); but, because of time limitations, this approach was not pursued. Instead, a new vessel, No. 6, was spun to the low side of the ID tolerance and the wall thickness was made larger, thus allowing sufficient material for machining the ID and OD of the shell section after all welding and annealing operations were completed. With this approach, it was possible to maintain close tolerances of the core vessel ID and wall thickness.

4. Reflector-Drive Engineering

a. Development Test

The high temperature test to evaluate the operational performance of the S8ER reflector and drive system under the operational temperature conditions was completed. The test setup consisted of a simulated core vessel and one complete set of reflector control drums including the bearing support and the drive and scram mechanism. Due to the hazard of possible beryllium oxidation, the beryllium components were fabricated from stainless steel to simulate the external outlines, weight, and rotational inertia. The entire system was enclosed and electrical heaters were used to heat the various components to the calculated operational temperatures. These operational temperatures were maintained for 1100 hr during which the control drum was driven in and out a total of 3753 cycles including 940 scrams. The operation of the system was satisfactory except for three failures of the drive motor. These motors failed after 312 to 450 hr of test (duty cycle approximately 60%). The failures were due to stator winding burn, probably caused by local hot spots. An attempt is underway to remedy the deficiency. In addition, a high-temperature unit has been proposed for motor replacement. The proposed design requires further testing.

The test on the two types of materials for the reflector bearing ball was concluded after 2933 hr in nitrogen atmosphere at 1100°F under operational

Figure 74. Plated-Ball Bearing
Assembly After Test

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Figure 75. Solid-Ball Bearing
Assembly After Test

load. During the test, the bearing journal was rotated in 2.5° increments for 133,642 cycles at a rate of one rotation per minute. This was followed by various dwell periods, ranging from 2 to 168 hr, and subsequent rotations. The total number of rotations was 140,516. Both bearing assemblies (solid chrome carbide, Carboloy 608 ball; and Hastelloy-B with Linde LC-1A-plated chrome-carbide surface ball) exhibited calculated apparent coefficients of friction varying in a random manner from 0.22 to 0.35. The various dwell periods did not affect the frictional coefficient. The solid material showed somewhat greater variation between the static and dynamic friction ($\Delta\mu \sim 0.2$) than the plated material ($\Delta\mu \sim 0.1$). During the test, the solid chrome carbide ball showed evidence of increasing tightness in the outer race. Inspection after the test confirmed this evidence. The Hastelloy ball was free throughout the test and after the test. Post-test examination indicated that the plated Hastelloy ball had better surface wear characteristics (Figure 74) than the solid chrome carbide ball (Figure 75). This difference probably accounts for the greater difference between the starting and running friction of the latter. Based upon the results of this test, the Hastelloy-B ball with the plated chrome carbide surface has been selected for the S8ER. The mating components, both shaft and outer race, are Linde LA-2 aluminum oxide plating on Hastelloy-B.

The high-temperature structural test on the beryllium control drum was also completed. In this test, the beryllium control drum was subjected to the dynamic operating loads after being soaked at 1300°F for varying periods. The dynamic loads were imposed by scrambling the drum and snubbing its rotation at the end of the scram using the S8ER operational components. A fixture was provided to simulate the minimum drum clearance. The drum rotated satisfactorily before, during, and after the various scrams (25 to 100) conducted after each period of temperature soak (10 to 168 hr). Inspection after these dynamic tests showed excessive oxidation of the beryllium. This oxidation was probably due to air leakage and outgassing of the walls of the argon-purged furnace. The oxidation between drum and the shims had produced an oxide layer approximately $1/8$ in. thick as shown on Figure 76. The force due to oxide buildup was sufficient to rupture the four shim retaining bolts; however the shims were still retained as shown on Figure 77. It is concluded that the beryllium control drum has adequate structural capacity to withstand the expected dynamic loads. The anodized oxidation-inhibiting coating developed for

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Figure 76. Oxide Layer on Control Drum

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Figure 77. Reflector Shim Intact After Structural Test

the S8ER beryllium reflector will protect the elements for the planned life of this test reactor from oxidation and resultant shim bolt failure.

b. Design

The early results of the dry critical experiment indicated the desirability of additional shims for the control drum to increase its total worth. The design of the control drum and shims was revised to add another 1.30 in. thickness of shim on the outer face of the drum as shown on Figure 78. Due to presence of the reflector support shaft, it was necessary to add the shim in two pieces. The effective worth of the added shim thickness was checked by adding an equivalent thickness of beryllium to one of the control drums on the S8ER dry critical assembly. The effect of the additional weight and the rotational inertia was checked by adding equivalent weight and inertia to both the high-temperature beryllium structural test and the reflector and drive system operational checkout test, no problems were encountered.

c. Fabrication

The beryllium reflector components, including the control drum and the original shims, were anodized by Brush Beryllium Co. and delivered. Armour Research Foundation conducted the oxidation test required for acceptance. The additional control drum shims are being fabricated and anodized by Brush.

The final fabrication of the subassemblies to convert the S8ER critical reflector and drive system to the operational system for the power test is being completed. Maintenance and overhaul of the components used for the critical system are being completed prior to installation on the operational system.

The revised drive motor assembly for higher temperature has been ordered.

5. System Assembly and Checkout

a. Installation of S8ER Equipment

The shutdown shield, with the core vessel and NaK inlet and outlet pipe spools welded in place, was moved to the S8ER facility (Building 010). Installation and final checkout of the control drums and control drum drives was initiated. The control drum drives were removed from the critical machine, disassembled, cleaned, and relubricated. The control drum drive shafts were

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Figure 78. Reflector Plan

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machined to permit assembly with optimum positioning of the control drums. The shutdown-shield vent lines, air cooling piping, and NaK heater overflow piping in the containment vessel were prefitted and are ready for installation when the shutdown shield is lowered into the containment vessel. A special platform was fabricated to be used to support personnel while they are performing work in the containment vessel on top of the shutdown shield.

With the exception of the reactor inlet and outlet NaK lines, all S8ER heat transfer and service equipment outside of the containment vessel has been installed and checked out satisfactorily.

b. Instrumentation

(1) NaK Surge Tank Level Indicators

The S8ER NaK surge tank level indicators were found to be inoperative during Phase II of the preliminary heat transfer checkout test (see Section VI-A-7-a). The malfunction was attributed to the formation of a ground loop between the J-tube level indicators and the EM pumps. This ground loop was broken by installing an isolation transformer in the J-tube signal circuit and by raising the internal circuits of the level indicators (secondary elements) above chassis ground. These circuits were then operationally checked using simulated J-tube signals. The effectiveness of these new isolation methods cannot be verified until the EM pumps are operated again.

(2) Testing of the Intermediate Power Range Nuclear Channels

A test was conducted to confirm that the intermediate power range nuclear channels (Channels III and IV) would respond to periods shorter than 3 sec. It was determined that these channels would detect a simulated reactor period of less than 2 sec, and that these channels would then cause the reactor safety circuit to trip within 2 sec after the period signal was initially applied to the channel.

(3) Nuclear Channel Calibrator

A special instrument was designed and constructed to furnish signals for checking neutron level indications and alarm trip points in the intermediate and power-range neutron monitoring instruments. This instrument, the nuclear channel calibrator, also furnishes time-varying signals for checking

reactor period indication and alarm set points in the startup and intermediate-range neutron monitoring instruments. The nuclear channel calibrator is a very versatile instrument in that it can be used for checking out and calibrating nuclear instruments built by several different manufacturers.

(4) S8ER Noise Analyzer

A special reactor noise analyzer system was constructed and is now under test. This equipment will provide a continuous visual indication of the relative dynamic stability of the reactor. A continuous spectrum plot of the reactor noise, which is proportional to the transfer function of the reactor, appears on both an oscilloscope and an X-Y or Y-T chart recorder.

The noise spectrum analyzer samples the output from an ion chamber, thermocouple, etc., and by means of a matrix of bandpass filters and rms detectors, produces a simultaneous display of the entire noise spectrum from 0.005 to 50 cps. The system consists of an input amplifier section, a matrix of frequency selective amplifiers serving as narrow bandpass filters, a rectifier and smoothing circuit for each filter, a mercury jet commutator switch for sampling the output of each filter circuit, and an oscilloscope for displaying the output spectrum. An additional automatic sampling switch is included to provide a simultaneous output to a chart recorder.

The basic part of the analyzer, the matrix of filters is composed of 40 tuned amplifiers, each with a bandwidth of 20% of its center frequency. The center frequencies of the filters are in 20% steps starting at about 0.005 cps. Each filter consists of an operational amplifier with a Twin-T network in the feedback loop.

A full-wave linear rectifier circuit at the output of each filter changes the a-c filter component to pulsating dc which is smoothed with an appropriate rectifier circuit network. The time constants of these networks depend on the frequency of the filter component and range from about 5 min at the high frequencies to over an hour at the lowest frequencies. The smoothed d-c output from each of these rectifier circuits is proportional to the rms value of the noise component at the filter frequency.

6. Operations Planning

a. Training Program

The second session of the S8ER Training Program was conducted during this report period. A total of 81 lecture hours included system description and operation, 41 hr; health physics and safety procedures, 27 hr; and theory, 13 hr.

A reactor operator certification program for S8ER operators was initiated. The certification examination consists of written and oral section. Four members of the S8ER Operations Unit have completed the written section; two members have completed the oral section.

b. Selective Fuel Loading

A selective fuel loading plan has been prepared. The fuel-moderator elements will be selectively loaded according to the following criteria.

- 1) The fuel-moderator elements with the largest radial gaps will be located in the central regions of the core. Thus the higher burnup, higher temperature fuel will have more volume within the cladding for expansion due to irradiation damage swelling. The fuel-moderator elements with the smallest radial gap will be located at the core's periphery. These elements will experience less swelling since the neutron flux is lower and coolant streaming at the periphery will keep the fuel temperature low in this region.
- 2) The fuel-moderator elements with the lowest hydrogen permeation rates will be located in the central regions of the core. Thus hydrogen loss will be minimized in the area of maximum reactivity importance.
- 3) The fuel-moderator elements will be positioned so that the average N_H and total uranium per rod in the central part of the core are close to the fuel specification values. To prevent fuel operation at excessively high temperatures in this region, the actual values should not be above specification. To minimize distortion of radial flux profiles, the values should not be below specification.

- 4) The fuel-moderator elements will be positioned so that there will be a minimum number of elements with large dimensional envelopes clustered together. The elements will be positioned this way to assure that all the elements can be loaded into the core with a minimum amount of element-to-element contact.
- 5) The axial gap was not taken into account since the values of the gap were fairly constant. Also, experimental data on the irradiation swelling of U-Zr-H_x had shown that the volume change was due to the diameter increasing, with the length decreasing or remaining constant.

A final fitup check of this selective fuel loading plan will be performed during the water immersion test in Building 012.

c. Test Program

The SNAP 8 Experimental Reactor test program specification was written. The specification delineates the various experiments that comprise the S8ER test program, outlines the objective of each test, specifies the test procedure review and approval sequence, and gives the test program sequence.

7. Experimental Program

a. Preliminary Heat Transfer Checkout Test

Phase II of the preliminary heat transfer checkout test (8ER-24) was completed. The purpose was to demonstrate that the S8ER heat transfer system would operate according to design specifications. The following major problems which existed during Phase I, conducted in July 1962, were found to have been satisfactorily solved.

(1) Overheating of the Saturable Reactor

The overheating of the saturable reactor was found to be due to undersized lead wiring, which was removed and replaced by wire capable of carrying the maximum current. The saturable reactor was then found to operate within its design specifications.

(2) Overheating of the Primary and Secondary EM Pump Coils

The overheating of the primary and secondary EM pump coils was found to be due to lack of thermal insulation between the pump throats and pump

coils. The coil heating was determined to be caused by radiation heat transfer from the 1300°F pump throats to the pump coils. The overheating problem was solved by placing a thin sheet of dimpled stainless steel between the pump throats and the pump coils. The stainless-steel sheet had a polished side which was inserted facing the pump throats to act as reflective insulation. A thin sheet of mica was then inserted between the stainless-steel sheets and the pump coils to serve as electrical insulation. Since the primary EM pump operates in a higher ambient temperature ($\sim 250^\circ\text{F}$) than the secondary EM pump ($\sim 100^\circ\text{F}$), flexible ducts were arranged to blow cooled nitrogen (from the primary equipment vault nitrogen-to-water heat exchanger and blower unit) over the primary EM pump coils. Secondary pump coils were cooled by forced circulation of reactor room air. As a result of these modifications, the pump coils of both EM pumps operated below their design temperature of 600°F in Phase II of the test.

(3) Excessive Leakage Through Primary Vault Penetrations

The excessive leakage through the primary vault penetrations was found to be due to gas leakage between the electrical insulation and the conducting wire. The problem was solved by removing $\sim 1\text{-}1/2$ in. of insulation from each wire, covering the bare wire with Tygon tubing of sufficient diameter to leave a radial gap of approximately $1/8$ in. between the wire and the inside of the Tygon tubing, and then immersing the entire bundle of wires and Tygon tubing in an epoxy resin. The epoxy resin then flowed into the annulus between the wire and Tygon tubing and around the Tygon tubing to form a gas seal. With this modification, no further leakage problems have been encountered.

(4) System Heater

The system heater wiring was changed from a Y system to a delta system. Two heating elements are now connected in series in each leg of the delta system. Since three of the thirty individual heating elements have failed, the voltage taps on the 150-kva heater transformer were changed to permit a maximum heater power of 118 kw. In Phase II of the test, it was demonstrated that this heater capacity will be adequate for the S8ER test program.

b. Repermeation Testing of S8ER Fuel Elements

(1) Fuel Element Bend Test

A repermeation test was performed on an S8ER fuel element. The element was subjected to a test simulating the bending which may be required

to load an element into the reactor. The objective was to learn whether the previously determined hydrogen leak rate of 0.07 cc/hr was affected. The maximum envelope of this element was 0.574 in. compared with a maximum permissible envelope of 0.570 in. (The fuel elements are located on 0.570-in. centers in the grid plates.)

The bending method used was as follows. The element was horizontally suspended by its endpins between two "V" blocks. The maximum eccentricity position was placed at the top. The fuel element was then depressed by hand at the center, 0.030 to 0.040 in. This bending was then repeated 14 times. At the conclusion of the bending test, the fuel element was re-permeation tested and it was determined that the permeation rate of the fuel element changed from 0.07 to 0.12 cc/hr. The rate is well within the specification of 1.7 cc/hr, even after the extreme bending test. The permeation history of the fuel element, No. E-188, is presented in Figure 79.

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Figure 79. Permeation Test History of S8ER Fuel Element No. E-188

(2) Hydrogen Leakage of Fuel Elements After Handling

Twelve qualified S8ER fuel elements were tested after the S8ER Critical Experiment to determine any adverse effects on the integrity of the ceramic hydrogen barrier due to fuel handling and dry critical experiments.

The criteria for selection of the elements to be permeation tested were:

a) amount of handling at the dry critical facility, b) nominal radial gas gap, c) initial permeation leak rate, and d) manufacturing delivery date (some changes in fabrication techniques were made during the production of S8ER fuel elements).

The results of these tests, presented in Table XXXVI indicate fuel handling and dry critical tests caused no adverse effects on cladding integrity.

TABLE XXXVI

TYPICAL S8ER FUEL ELEMENT LEAK RATES BEFORE AND
AFTER THE S8ER CRITICAL EXPERIMENT

Element Number	Amount of Handling	Radial Gap (in.)	Leak Rate at 1400°F		Delivery Date
			Before (cc/hr)	After (cc/hr)	
E-181	Maximum	0.0023	0.42	0.13	8-6-62
E-182	Normal	0.0009	0.24	0.18	7-26-62
E-193	Normal	0.0010	1.42	1.23	7-26-62
E-271	Normal	0.0023	0.33	0.12	8-6-62
E-292	Above Average	0.0011	0.50	0.57	7-26-62
E-313	Considerable	0.0008	0.96	1.05	7-26-62
E-342	Normal	0.0025	0.27	0.24	8-6-62
E-348	Normal	0.0022	1.57	1.27	8-11-62
E-462	Normal	0.0013	1.65	1.28	8-14-62
E-503	Normal	0.0013	0.21	0.07	8-23-62
E-513	Normal	0.0017	1.09	1.76	8-23-62
E-519	Normal	0.0022	0.47	0.09	8-23-62

B. SNAP 8 GROUND PROTOTYPE (S8DS)

1. Introduction

The ground prototype test will be the first test in which a SNAP 8 reactor (S8DS-1) and power conversion system (PCS) are coupled together to operate as an electrical generating system (EGS) and will be the second nuclear test in the SNAP 8 development program. The EGS for this test will consist of the S8DS-1 reactor and neutron shield and a 30-kwe flight power-conversion system. The reactor and the PCS will be coupled together in a nonflight configuration so that the PCS and the test support equipment may be shielded from the reactor to facilitate changing and maintaining this equipment.

The test objectives are to: (a) demonstrate satisfactory 300-kwt operation of the flight reactor and determine its operating characteristics, (b) demonstrate the radiation stability of the LiH shield at 300 kwt, (c) determine the control requirements for automatic startup of the reactor, (d) determine the

control requirements for reactor and system stability during PCS startup, (e) determine the stability of the complete electrical generating system, and (f) demonstrate 20 automatic startups of the complete 30-kwe electrical generating system.

2. Power Conversion System Modification

Aerojet General Corporation (AGC) has tentatively changed the power conversion system. These changes will affect the Ground Prototype Test as follows:

- a) A decrease in power conversion system efficiency will require an increase in the expected operating power for the S8DS-1 reactor from 300 kwt to about 450 kwt. The reactor design power will remain at 600 kwt.
- b) Prior to AGC design changes, a new AI reactor shield design criteria increasing the power from 300 to 600 kwt had been contemplated. The PCS design revisions will result in changing the test objective power from 300 to 450 kwt for demonstrating satisfactory operation of the flight reactor and the radiation stability of the LiH shield.
- c) The incorporation of extra PCS loops containing organic coolants and other changes in the power conversion system equipment will render obsolete the equipment arrangement proposed for the Ground Prototype Test Facility (GPTF) test vault. These changes are not expected to greatly affect the layout and design of the nuclear system test support equipment.
- d) Unresolved changes in the flight system radiator dimensions and geometry have delayed S8DS-1 shield design criteria. Since the existing shield casting facilities and the GPTF reactor test pit will not accommodate a shield over 4 ft in diameter, it is not certain that a full-size flight shield can be used for the S8DS.

3. Reactor Core Vessel and Support Structure

A preliminary design criteria specification for the reactor core vessel was prepared. Preliminary layout studies of the reactor and shield support structure were made. A structure designed in two sections — an upper and a

lower section — was considered. The lower section would support the shield assembly at the bottom ring flange and the upper section would support the reactor assembly. An adjustable shield support which would permit the position of the shield relative to the reactor to be varied was also studied. The adjustable shield support was considered for use with a 300-kw shield. It would be desirable to be able to move the shield away from the reactor in the event that reactor power levels higher than 300 kwt are required. It is anticipated that the current shield will be designed for 600 kwt and will not require this adjustability.

4. Neutron Shield

Plans to use the S8DRM shield design for the S8DS test were abandoned. The preliminary design studies were initiated for a shield which would fulfill the present design criteria requirement of a 600-kwt reactor operating power level.

The low thermal conductivity of lithium hydride causes peak temperatures in the S8DRM-1 shield to be excessive when operated at 600 kw. Several design modifications were considered from the thermal standpoint to alleviate this problem. The use of a high conductivity material on the top head of the shield to permit better radial heat conduction appears promising.

It is presently planned to place the shadow shield as close as possible to the bottom of the reactor vessel. The reactor system and the shield will be supported independently of one another. The environment of the pit will satisfactorily simulate space operation of the shield as far as vacuum and pit wall temperatures are concerned, but may cause problems with respect to neutrons scattering from the pit wall into the shield. These scattered neutrons will increase the heat generation rates, particularly near the outer regions of the shield. The expected axial and radial heat generation rates in the shield are being analyzed for both space and reactor pit environments (Section IV-B-1). These results will be incorporated into the thermal analyses.

The design criterion for neutron and gamma ray dose at the payload in the SNAP 8 reactor is 10^{13} nvt fast flux and 10^7 rad, respectively, for 10,000 hr of operation. Approximately one-half of the neutron dose in outer space is expected to be due to direct passage of neutrons through the shield. In order to have added confidence in the calculated dose rates at the payload, it is desirable to make neutron measurements during the course of S8DS-1 operation. The

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Figure 80. Ground Prototype Control Mechanism

scattering of neutrons from the reactor pit walls makes any quantitatively accurate dose measurements a formidable task. The use of chamber type devices to measure low level fast neutron currents in the high gamma background of the area beneath the shield is questionable. Investigations of this problem are discussed in Section IV-B-1.

5. Reflectors and Drives

Preliminary design of the reflector and drive system was initiated. The current design concept is a modification of the S8DRM-1 which provides for:

- a) Installation, removal, and replacement of the reflectors and associated mechanisms by means of remote handling equipment.
- b) Scram and restart of the reactor without disturbing the physical adjustment between the beryllium reflector structure and the core vessel. The scram and restart capability are provided with minimum alteration of the flight components and configuration.
- c) Operation in the ground prototype test environment.

Figure 80 shows the control system. Figure 81 shows the S8DS control drum drive and scram mechanism which will be mounted on the top of the reflector and engaged to each of the six control drum shafts. Each unit is a complete package which simulates the flight control drive. The mechanism also permits the fail-safe spring-powered scram of each control drum by rotating the drum outward to a less reactive position. The drive motor, gearing, gear end bell, and gear arrangement is essentially the same as the flight design. The bearing materials may not be the same as those of the flight system. Indications are that the current flight system bearing material will not be suitable in the ground prototype test environment.

The torsion spring which produces the energy for the scram is designed to be locked in a cocked position by an electromagnetic scram clutch. The force of the scram spring would be applied to the control drum shaft only when the clutch is disengaged for scram. This mechanism introduces no additional loads into the drive system and gives close simulation of the flight drive system.

Detailed development drawings of the drive and scram mechanism have been completed. Fabrication of one development test unit has been initiated.

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Figure 81. Ground Prototype Control Drum Scram Assembly

6. Heat Transfer System and Instrumentation

The design of the nuclear shakedown heat transfer system is nearing completion. The design drawings and specifications were prepared with separate specification packages for the primary and secondary NaK heat transfer loops. This will permit the construction of the secondary NaK loop by an outside contractor.

Fabrication of the diffusion cold traps, NaK filters, and "J" tubes was completed.

The long delivery time components for the plugging meter have been received and fabrication of these items is now approaching completion. The two EM pumps and the NaK flowmeters are well along in fabrication and modifications have been started to attach to indicating switches to the NaK valves.

Much of the equipment has been received including the high-temperature bellows-seal NaK valves, inert-gas manifolds, vacuum pumps, diffusion cold-trap refrigeration system, diaphragm-seal gas valves, EM pump transformers and capacitors, parasitic load heaters and control transformer, and various tank and pipe heaters. The intermediate heat exchanger has been completed by the vendor and is in shipment. Five of the six tanks have been fabricated. A leak was detected in the sixth tank and repairs are being made. Fabrication of the airblast heat exchanger has also begun.

The nuclear instrumentation specifications have been completed.

The core inlet, outlet, and differential pressure gages will be attached to the reactor fill and drain lines and located outside the reactor pit rather than inside as previously reported. The compact pressure transducers for the inlet and outlet core pressure measurements and installation inside the reactor pit are to be used in the secondary NaK system. Since there is no longer a size limitation on the transducers, the NaK-filled capillary pressure gages, a type used successfully on previous tests, will be used for measuring the core inlet and exit pressures. The compact differential pressure gage will still be used for measuring the core differential pressure.

7. Facility

The area surrounding the reactor pits in the GPTF was redesigned to increase the flexibility of the facility by providing for the future removal and

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Figure 82. Proposed Addition to Ground Prototype Test Facility

replacement of the reactor pits. This was accomplished by replacing the concrete fill with sand which can be more easily removed if future modifications are required.

The quantity of instrumentation required by the power conversion system has increased considerably over the original design requirement and it has been proposed that the control room wing of the GPTF be lengthened 20 ft as shown in Figure 82.

C. SNAP 8 DEVELOPMENTAL REACTOR MOCKUP (S8DRM-1)

1. Introduction

The SNAP Developmental Reactor Mockup-1 (S8DRM-1) is a flight prototype system without live fuel elements which will be used in attempting to qualify the design for flight application. As this system is the first attempt to qualify the SNAP 8 nuclear system, some failures will undoubtedly occur. These failed components will be redesigned, if necessary, for the final qualification test program, the S8DRM-2 system.

Figures 83, 84, and 85 are assembly drawings of the core, shield, and reflector control system respectively.

The main objectives of the S8DRM-1 program are to: a) assist the development program in the design of flight qualified systems and components; b) demonstrate that systems and components can withstand simulated environmental conditions; c) demonstrate that systems and components can operate satisfactorily before, during, and after simulated environmental conditions; d) demonstrate the capability of assembling all individual systems and to assemble all systems together to form the nuclear system, and e) test the adequacy of fabrication techniques to provide flight qualified systems and components.

In order to prevent the S8DRM-1 from being a one-of-a-kind or hand-finished product, sufficient components are being purchased for development testing and for replacement parts for the system testing.

During the reporting period, the detail design of the S8DRM-1 was completed and fabrication was started. A number of parts were completed. As the S8DRM-1 moves ahead in the fabrication phase, numerous fabrication problems have arisen which will be eliminated by redesign in the S8DS-1 program.

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Figure 83. Reactor Core Vessel

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Figure 84. Shield Assembly

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Figure 85. S8DRM-1 Reflector System Layout

2. Test Program

The formal test program for the S8DRM-1 includes an assembly test, a startup reliability test, a prelaunch environmental test, a launch environmental test, an orbital startup test, and an endurance test. Each test is described as follows.

a. Assembly

This test is designed to prove the ability to assemble the complicated control and safety system about the reactor core vessel. It is also designed to check the installation of the ground handling safety systems. In addition, as the reflector control system is designed to be removable for shipping purposes, this test will determine the variations in tolerance to be expected due to this removal-reassembly procedure.

b. Startup Reliability

After the assembly test, the reactor and control assembly will be shipped to Santa Susana where it will be mated with the LiH shield. Nine consecutive startup reliability tests will then be conducted. This test is designed to simulate reactor startup up to the point of sensible heat generation. The test will include the firing of all drum lockout squib units.

c. Prelaunch Environment

The prelaunch environmental test shall be performed under a vacuum of 10^{-3} mm Hg. It is designed to simulate prelaunch conditions of the SNAP 8 nuclear system. The nuclear system less core internals (grid plates, fuel elements, etc.) will be thermally cycled five times. Each thermal cycle will simulate the reactor startup temperature transient.

d. Launch Environment

The launch environmental test is a vibration and shock test designed to test the mechanical design of the SNAP 8 nuclear system. For this test, the core internals will be assembled in the core vessel and the system filled with water to mockup the NaK. The nuclear system will then undergo a series of vibration and shock tests designed to simulate launch conditions. Following this series of tests, another series of startup reliability tests shall be performed. The two types of tests will provide input to component level testing by determining various attenuation and amplification factors.

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Figure 86. Reactor Heat Simulator

e. Orbital Environment Startup

For the orbital environmental startup test, the core will be drained and the core internals (fuel elements, grid plates, etc.) removed and replaced by core heaters. This test is designed to simulate startup of the reactor. The test will be conducted in a vacuum chamber with a vacuum of 10^{-3} mm Hg. During the test (168 hr), the control system shall continuously operate to simulate hydrogen loss, prepoison burnout, etc., and the switching on and off of the parasitic load.

f. Endurance

The endurance test will be a high vacuum and high (operating) temperature test designed to simulate the life and space environment of the SNAP 8 nuclear system.

During the reporting period the reactor heat simulator was designed and fabrication started. Figure 86 is a layout of the reactor heat simulator.

3. Description

The S8DRM-1 is composed of the following major items: a) flight design reactor vessel and support structure; b) forty natural uranium - zirconium hydride fuel elements and 211 hollow dummy stainless steel fuel elements; c) flight design reactor control system composed of a flight design reflector-and-drive system and a breadboard electronic system excluding diagnostic instrumentation; d) flight design reactor safety system; and e) flight design reactor shield.

a. Reactor Vessel

The reactor vessel and support structure consists of a core vessel, grid plates, baffle plate, internal reflectors, and control system supports. The core vessel is a right circular cylinder of 316 stainless steel having a reversed dished inlet head, a torispherical outlet head, a flued elliptical inlet nozzle located in the cylindrical portion of the inlet plenum head, and an outlet nozzle located in the center of the torispherical head. The core vessel holds the grid plates, baffle plate, internal reflectors, and the natural and dummy fuel elements, and supports the reflector assembly from lugs attached to the bottom head.

During the reporting period, all design drawings were released for fabrication. Changes in the manufacturing techniques for the core vessel were incorporated to improve core vessel yield over the S8ER procedures. The vessels are first rough machined and all fixtures welded on. They are then stress relieved and final machined.

The design of the internal reflectors was also modified to improve yield. This modification essentially increased the cladding wall thickness and made the tolerances less severe.

The upper grid plate was modified to accommodate the change in fuel element endpin configuration as shown in Figure 83.

b. Fuel Elements

The forty natural uranium – zirconium hydride fuel elements will be manufactured in a manner identical to the manufacturing of S8DS-1 fuel elements. The dummy stainless-steel fuel elements will have the same external dimensions as the natural uranium fuel elements and will weigh approximately the same.

The design drawings were completed and fabrication started. The design of the upper endpin was changed to facilitate manufacturing when analysis indicated the double-end piston rod previously reported has little dampening advantage over a single-end piston rod.

c. Control System

The reactor control system consists of a beryllium reflector, a startup programmer, a long-term controller, long-term actuators, startup actuators, cabling, and position sensors.

The beryllium reflector is mounted on an inconel plate supported from the core vessel. It is divided into two halves; each half contains three movable beryllium drums plus a fixed beryllium segment such that when the drums are in their full-in position the core is surrounded by a nominal 3-in. - thick beryllium cylinder.

The startup programmer is a solid-state electronic device designed to control the proper sequence of events during startup.

The long-term controller is a breadboard version designed to achieve proper long-term control drum sequencing. The long-term actuators are

bidirectional stepping motors which receive their input signal from the long-term controller and cause the control drums to take a discrete step when necessary. The startup actuators are unidirectional, externally commutated d-c motors which receive their signal from the startup programmer and drive the startup drums from their full-out to full-in position.

The design of the long-term actuators, using a slotted, magnetless rotor to produce discrete steps, was completed. The rotary motion is reduced by step-down gears to produce the required step size for the control drums. Due to the extreme environment under which the actuators are expected to perform, the proper selection and application of materials is of prime consideration. Parts have been fabricated and the actuators are currently being assembled. The present design requires a spring loaded brake to prevent the control drums from rotating during the no-power-on condition. A tractive magnet brake solenoid is energized to release the spring and allow the slotted rotor to rotate. The rotational step is produced by energizing a stator field.

During the continual development of high temperature wire insulation systems a technique has been developed whereby the insulation encapsulant and the wire are wound onto a solenoid bobbin simultaneously, producing an insulation system which appears far superior to any thus far developed on this project.

A present limitation with the winding techniques currently being used on the stator field is the penetration of the insulation encapsulant into the coils, which are wound previous to encapsulating. An arrangement is being designed which will permit the insulation encapsulant and the wire to be wound onto a stator bobbin simultaneously.

The design of the reflector and drive system including the flight safety system was completed. The stress analysis and functional analysis of this design was completed. The final design of the reflector and drive and flight safety systems is shown on Figure 85.

The fabrication of the component hardware was initiated. The current status of the major components is:

- 1) The beryllium reflector components were fabricated and are being processed with the anodized coating for oxidation resistance and emissivity control.

- 2) The drive components which include the gears, coupling reflector bearings, cams, and the bracketing were completed.
- 3) Fabrication is continuing on the main reflector structure and several miscellaneous items.

During the reporting period, it was decided to eliminate the flight-type long-term controller and substitute a breadboard controller. The main reason was the inability of vendors to provide a radiation-resistant controller without very nearly duplicating our present irradiation program. This, of course, meant that the controller would be an exceeding long-lead, and unduly costly, item which would delay the overall program.

In addition, the temperature sensor-switches were eliminated. The main reason for this was the inability to procure switches to the required specifications. Further development effort is needed on this item before it can be included in the program.

d. Safety System

The flight design safety system is also a part of the S8DRM-1. The system includes a reentry band, reflector ejection springs, a destruct charge (mockup), and associated circuitry. In addition, void filler blocks and a drum lockout system are furnished as part of the ground handling safety equipment.

The reentry band is a thin band of stainless steel which holds the reflector assembly in place at the top of the core vessel. The band contains a spring to provide proper tension, a telemetry shutdown fuse to provide for ground command ejection, and an end-of-life actuator which breaks the band when the outlet NaK temperature drops approximately 400°F. The band is also designed and located so that it will be melted by reentry heat. Breaking of the band allows the reflector ejection springs to remove the reflector assembly from the vicinity of the core vessel and this, in turn, opens the marmon clamp holding the core assembly to the shield assembly. The purpose is to maximize reentry burnup by dispersing the major assemblies.

The destruct charge is designed to physically destroy the reactor assembly and thereby prevent any possible criticality. The S8DRM-1 will have a mockup destruct charge of the proper weight, physical size, and mounting connections.

The void filler blocks and drum lockout system are designed to minimize the possibility of a nuclear excursion during ground handling and shipping. The void filler blocks are neutron poison blocks designed to be slipped into the cavities in the stationary beryllium reflector normally occupied by the control drums when in their full-in position. The key-lock is a locking system designed to prevent control drum motion even if the actuators are energized.

During the reporting period the design of the safety system was completed and fabrication was initiated.

e. Shield

The flight design reactor shield consists of a gamma shield and a neutron shield. The shield is designed to limit the payload dose assuming a live reactor operating for 10,000 hr at 300 kw. To limit the cost of the neutron shield, only natural LiH is used for shielding. The gamma shield is a tungsten plate shaped to fit directly below the core vessel lower plenum. The neutron shield also serves as the major structural item transmitting the reactor assembly load through the shield to a mating ring attached to the bottom of the neutron shield vessel.

During the reporting period, the design of the shield was completed and fabrication was started. Figures 87, 88, 89, 90, and 91 show the current status of fabrication.

Because of the void regions noted in the SNAP 8 test casting as reported in Section V-F, a structural analysis was conducted on the stability of the lower head of the shield during the casting cycle. It was concluded that the temperature gradients during cooling from the casting temperature could be sufficient to cause plastic deformation of the head. To reduce the problem, it is planned to incorporate approximately eight radial ribs in the lower head of the second casing unit. In addition to providing strength the ribs will also aid in maintaining the integrity of the bond between the LiH and the shield casing.

The thermal analysis of the shield was refined during the reporting period. The Thermal Analyzer Program (TAP) was employed to obtain detailed information on the temperature distribution throughout the shield under steady-state conditions for a net reactor power level of 300 kw. Results are shown in Figure 92. The thermal conductivity of LiH was taken as 2.0 Btu/hr-ft-°F and the environment below the shield was assumed to be at a temperature of

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Figure 87. Shield Cone

700°F due to PCS equipment and radiators. Further refinements in the thermal analysis are anticipated upon incorporation of new heat generation rates resulting from the TDC analysis of neutron fluxes in the shield.

D. SNAP 8 FLIGHT SYSTEM (S8FS)

1. Introduction

The S8FS experiment is the proof test for the electrical generating system (EGS). All components for the nuclear system (NS) will be fully qualified and their arrangement will be in the flight configuration. The NS will be integrated in the spacecraft arrangement with the power conversion system (PCS). Endurance tests will be carried out at the rated 30-kwe power levels.

2. Test Objectives

The objectives of the S8FS test is to prove the adequacy of the SNAP 8 design to meet the overall program objectives, that is: a) demonstrate the rated performance, control stability, and system reliability of the complete EGS up to 10,000 hr at 30 kwe in a simulated space environment and undercharging loads, and b) demonstrate three consecutive, successful startups.

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Figure 88. Bottom of Shield Lower Head

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Figure 89. Shield Mounting Ring

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Figure 90. Shield Honeycomb

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Figure 91. Shield Upper Head

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Figure 92. Steady-State Temperatures
in S8DRM-1 Shield, 300 kw Net

3. System Description

The NS will be flight system equipment which has been modified to scram and restart as required for ground operation. It will be a later model of the S8DS-1. Updating of the design will be governed by S8DS-1 early test results, S8ER operational results, S8DRM-1 fabrication and test experience and information from all continuing fuel and component development programs. Design of the actual system or specific individual components has not yet started, although the entire SNAP 8 program now underway is aimed toward this goal.

4. Test Facilities

Detailed construction drawing design of the test facilities (Figure 93) is progressing at the architectural and engineering office of the Bechtel Corporation. Design is based upon criteria established in AI Specification No. L137-F-70001.

To expedite beneficial occupancy of the facility, Bechtel has broken design into a series of packages. Each individual package when completed is to be fabricated and/or constructed. As the design of each construction increment progresses, it is reviewed in detail. To date, completed designs have been reviewed for all work covering excavation and rough grading, test vault concrete, and miscellaneous mechanical devices including portable and master slave manipulators and a hot cell shield window. Rough excavation has been completed (Figure 94).

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Figure 93. SNAP 8 Flight Prototype
Test Facility
(Building 056)

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Figure 94. SNAP 8 Flight Prototype Test
Facility - Present Status

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