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SNAP 8 QUARTERLY PROGRESS REPORT

APRIL - JUNE 1963

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AEC Research and Development Report

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M-3679(30th Ed.)

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The previous SNAP 8 Quarterly Report, NAA-SR-8392,
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I. PROGRAM OBJECTIVES

SNAP 8 is a nuclear power supply system (NS) designed to provide 35 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 (PCS). 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 (a) develop the SNAP 8 reactor, including its associated controls, and safety features; (b) develop shields for various applications; and (c) test the complete SNAP 8 electrical generating system (EGS), 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	Space electrical power
Test vehicle	As prescribed by mission
Safety criteria	Subcritical during launch Orbital startup Orbital shutdown at end-of-life backed up by reentry shutdown mechanism
Life	10,000 hr
Power conversion	Mercury Rankine cycle Turboalternator being developed by AGC under contract to NASA
Electrical power output	35 kw
Reactor thermal power	600 kw (operated at 70% rated thermal power to produce 35 kw electrical power)
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 at $1300 \pm 20^{\circ}\text{F}$ by stepping reflector control elements in or out as required.
Reactor weight	585 lb
Shadow shield weight	As prescribed by mission

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II. PROGRESS SUMMARY

A. FLIGHT REACTOR DESIGN

Several improvements were made in the reference flight reactor design during the quarter. In general, the following changes were made to increase the reliability of the system.

A holddown device has been included in the fuel element to prevent damage to the ceramic hydrogen barrier during launch shock and vibration.

The electrical drives on the startup control drums have been replaced by "snap-in" spring drives.

The gear-type, control-drum drive motors will probably be replaced by a direct-drive device which eliminates the problems of galling and self-welding at the gears. This change is contingent upon successful development of a direct-drive actuator.

The radius of curvature of the control drums was increased from 4.0 to 4.5 in. to assure adequate control drum reactivity worth.

The optimum reactor operating conditions prior to PCS startup were established. These initial conditions will permit rapid startup of the PCS without imposing severe transients on the reactor.

B. FUEL DEVELOPMENT

Development of the new hydrogen barrier coating (SCB) continues to look very promising. Test results to date indicate that the SCB coating can withstand repeated thermal shocks and will operate at temperatures in excess of the required 1500°F.

C. COMPONENT DEVELOPMENT

Work is continuing on the investigation of self-welding, dry lubricants, high temperature insulation, beryllium oxidation, and emissivity coatings.

The reference design control drum bearings were subjected to a design shock and vibration test without suffering any damage or degradation.

The S8DRM-1 two-region neutron shield was cast successfully. A small pinhole leak, which occurred during the casting process, was repaired in a routine manner.

The mechanical properties of 347 stainless steel were found to be unchanged after 4000 hr of exposure to LiH at temperatures up to 1400°F.

D. SYSTEM TESTS

1. SNAP 8 Experimental Reactor (S8ER)

The S8ER was loaded with fuel using a unitized core loading technique which was developed to eliminate the problems associated with individually, remotely loading 211 fuel elements into the core vessel. The dry critical experiments were completed and the heat transfer system seal welds were made.

2. SNAP 8 Ground Prototype (S8DS-1)

The design of the S8DS was changed to include the 4.5-in. radius control drums and direct drive control drum actuators, in agreement with the reference flight design. The necessity for a new reactor containment vessel was established. The existing reactor containment vessel was found to be inadequate from the standpoint of neutron embrittlement and vacuum capacity. A vacuum of 10^{-5} Torr was found to be necessary to assure long life of the control drum bearings.

Facility construction was completed and acceptance testing was initiated.

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

Assembly of the S8DRM-1 has continued to the point where the wiring harness is now being fitted to the reactor and shield assembly.

4. SNAP 8 Flight System (S8FS-1)

Revised test requirements have resulted in a new facility concept. Since the space radiators will not be included in the system to be tested, the facility will be designed to operate at a high (10^{-5} Torr) vacuum and will not require the large heat removal capacity. A smaller test vault volume is also possible.

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 neutron leakage 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 weight of a typical unmanned-mission shield is calculated to be 1175 lb.) Each major assembly is described in the following paragraphs. Nuclear system parameters are given in Table II. A weight summary of the SNAP 8 nuclear system is given in Table III.

The design reliability goal for the nuclear system has been tentatively established at approximately 93% exclusive of micrometeoroid protection. The goal for nonpuncture probability has tentatively been established at 99.9%.

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 at./cc of (U-Zr) Hx. 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.

The reference design of the fuel element was changed slightly during the past quarter to include a holddown device. This holddown device allows for zero axial clearance during launch but allows sufficient axial clearance during reactor operations to account for fuel element irradiation swelling. This design change should significantly improve the ability of the fuel elements to withstand the launch environment.

B. CORE VESSEL AND INTERNALS (less fuel elements)

The core vessel is a cylinder of 316 SS 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 cylindrical portion of the inlet plenum head. The outlet coolant nozzle is a flued cylindrical nozzle located in the center of the torispherical head. The core vessel contains an upper grid plate of 316 SS; a lower grid plate of Hastelloy-C; a coolant baffle plate of 316 SS; internal BeO reflectors clad with 347 SS; and, of course, the 211 fuel elements previously described.

TABLE II

SNAP 8 NUCLEAR SYSTEM CHARACTERISTICS

Reactor Design		Reactor Parameters		Control System Parameters	
Fuel Elements		Design Operating Conditions		Single control drum worth, % $\Delta k/k$	
Number	211	Reactor thermal power, kw	600	Total drum worth (six drums), % $\Delta k/k$	14.3
Fuel Alloy, wt % U in Zr	10.5	Net electrical power, kwe	35 (@ 425 kw)	Full-in drum position, degrees	180
Uranium Enrichment, at % U^{235}	93.15	Coolant inlet temperature, °F	1100	Full-out drum position, degrees	71
Degree of Hydriding, NH_4 , 1022 a/cm ³	6.0	Coolant outlet temperature, °F	1300	Drum movement per step, degrees	0.52
Fuel rod diameter, in.	0.529	Coolant flow rate, gpm	140	Maximum drum rotation rates, degrees/sec	
Active fuel length, in.	16.5	Operating pressure, psia	40	Startup drums (simultaneous motion)	0.143
Fuel element length, in.	17.005	Core pressure drop, psi	5	Control drums	0.053
Cladding material	Hast-N	Maximum hot spot fuel temperature, °F	1580	Maximum differential drum worth, ¢/degree	6.0
Cladding ID, in.	0.540	Maximum hot spot cladding temperature, °F	1480		
Cladding thickness, in.	0.010	Operating lifetime, hr	10,000	Kinetic Parameters	
Ceramic barrier material	SCB	Peak heat flux, Btu/hr-ft ²	~100,000	Number of delayed neutron groups	6
Ceramic barrier thickness, in.	0.002			Decay constants, sec ⁻¹	
Diametral hydrogen gap, in.	0.007			λ_1	0.0124
				λ_2	0.0305
				λ_3	0.111
				λ_4	0.301
				λ_5	1.13
				λ_6	3.0
Source		Temperature Coefficients, ¢/°F (at design power level)		Effective delayed neutron fraction	0.0077
Type	Po-Be	Spectrum	-0.04 (est.,	Group delayed fractions	
Strength, curies	0.1	Axial expansion	-0.072	α_1	0.03297
Location	top head of vessel	Upper grid plate	-0.027	α_2	0.2194
		Lower grid plate	-0.015	α_3	0.1960
Internal Reflectors		Reflector (prompt, lower plenum)	-0.04	α_4	0.3947
Composition	BeO	Reflector (prompt, upper plenum)	-0.04	α_5	0.1150
Cladding material	347 SS	Reflector (delayed, Be temperature)	-0.06	α_6	0.04193
Cladding thickness, in.	0.018	Total isothermal (@ 1200 °F)	-0.30		
		Total isothermal (@ room temperature)	-0.16	Transport delay times (at 35 kwe), sec	
Inlet Plenum Grid Plate				Nuclear system to boiler	1.2
Material	Hast-C	Reactivity Requirements, % $\Delta k/k$		Boiler	4.2
Thickness, in.	7/16	Xenon	-1.0	Boiler to nuclear system	3.2
Number of coolant holes	420	Samarium	-0.9	Nuclear system	1.2
Diameter of coolant holes, in.	0.125	Burnable poison (Sm ²⁰³)	+2.6		
Spacing of coolant holes, in.	0.570	Fission product poisons	-0.9	Shield (300-kwt reference design)	
		U^{235} burnup	-0.4	Material	
Outlet Plenum Grid Plate		Temperature and power	-2.0	Neutron shield	LiH
Material	316 SS	Hydrogen redistribution	-0.5	Thermal insulation	Min-K
Thickness, in.	7/16	Hydrogen loss	-2.5	Gamma shield	W
Number of coolant holes	420	Contingency	-1.5		
Diameter of coolant holes, in.	0.156	Cold, clean excess reactivity	+7.1	Neutron Shield Cladding	
Spacing of coolant holes, in.	0.570	Total control drum worth	14.3	Material	347 SS
		Shutdown margin	7.2	Thickness	0.270
Coolant				Emissivity	0.9
Material	NaK-78	Nuclear Parameters		Dimensions, Neutron Shield	
Melting point, °F	12	Average thermal neutron flux, n/cm ² -sec	5×10^{12}	Maximum diameter, in.	40
Core Vessel		Median fission energy, ev	0.21	Minimum diameter, in.	31.4
Material	316 SS	Delayed neutron fraction, β_{eff}	0.0077	Height, in.	28.6
Internal diameter, in.	9.214	Mean prompt neutron lifetime, μ -sec	8.1	Dimensions, Gamma Shield	
Thickness, in.	0.105	Uranium loading, Kg U^{235}	7	Diameter, in.	9.5
Length, in.	22.4	Hydrogen reactivity worth, % $\Delta k/k$ per % $\Delta H/H$	0.45	Height, in.	0.85
Distance between grid plates, in.	17.067	Volume fractions in core		Dimensions, Thermal Insulation	
Reflector Control Elements		Fuel	0.782	Thickness, maximum, in.	3-3/4
Material	Be	Cladding	0.061		
Number of startup drums	3	NaK	0.123	Maximum Dose at Payload	
Number of long term control drums	3	Ceramic	0.021	Fast neutrons	
Total number of drums	6	Void	0.013	Scattered, nvt	5×10^{12}
Radius of curvature of drums, in.	4.5	Radial peak to average power	1.23	Direct, nvt	5×10^{12}
Nominal thickness of drums, in.	3.125	Axial peak to average power	1.40	Gammas	
Shimming capability, in.	±1.0			Scattered, rad (C)	1×10^6
				Direct, rad (C)	9×10^6

TABLE III
SNAP 8 NUCLEAR SYSTEM WEIGHT SUMMARY
(1b)

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		270.0
Control motors	16.5	
Startup springs	4.5	
Reflector adjust brackets	1.2	
Ground command 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		1,175.3
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	
Armor for micrometeorite protection	130.0	
Instrument Compartment	17.0	17.0
TOTAL		1,759.2

During the reporting quarter, the reference design of the core vessel and core internals remains essentially unchanged.

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 springs, and the three control drums are driven by linear actuators. 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.

During the reporting quarter, the reflector control system reference design was changed to incorporate a startup drum spring drive system replacing the externally commutated d-c motors. This change was made to increase system reliability when it became apparent that the safety criteria, which dictated the d-c motors, were unduly restrictive.

The control drum stepper motors were replaced with the linear actuator type of $1/2^\circ$, direct-drive stepping devices mounted on the top of the reflector assembly. These are connected by offset bellows so that the linear actuator is within the shield shadow cone. The linear actuators were selected to replace the stepper motors due to their simplicity and hence improved reliability over the stepper motor. The elimination, therefore, of the gear train will remove its self-welding problem from the program. This design modification is, of course, contingent upon the successful completion of the component development effort on the direct drive actuators.

D. SHIELD

The S8DRM-1 shadow shield design is typical of shield requirements for unmanned mission applications of SNAP. This design limits the neutron dose at the payload to 5×10^{12} direct and the gamma dose to 9×10^6 rads (carbon) direct. The payload configuration is assumed to be an 8-ft-diam 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 at a 300-kwt reactor power level. The S8DRM-1 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 .

Analysis has been initiated to investigate the shielding requirements for a variety of potential SNAP 8 missions.

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.

During the reporting quarter, the drum lockout system has been undergoing revision to allow full-length and full-width reflector shims to be added to the drums. This change has not yet been completed. No other significant changes occurred this quarter.

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IV. SYSTEM ANALYSIS

A. NUCLEAR ANALYSIS

1. Code Preparation

Two new Los Alamos transport theory programs are nearing the status of production use. These codes are the one-dimensional DTK program and the two-dimensional DDK program. Excellent success was achieved with the DTK program. Both isotropic and anisotropic versions were used to calculate fluxes for shielding purposes. Calculations were also performed on reactor criticality. Attempts to use the DDK have met with limited success and further efforts are needed.

The DTK cross section library was considerably enlarged during the quarter. The latest set of Hanson and Roach cross sections was run on the FAIM one-dimensional diffusion theory program. Multiplication factors (k_{eff}) for the S8ER were considerably below the experimental values and even more below previous FAIM calculations. The discrepancy appears to be due to changes in the uranium fission cross sections, zirconium cross sections, and assumed spectral distributions for the bound hydrogen cross sections.

2. Shielding Studies

Comparisons have been made of the fluxes in the shield test facility using the one-dimensional transport DTK code and the one-dimensional FAIM diffusion theory program. Figure 1 shows the comparison between the calculated and experimental values. The isotropic DTK calculations gave the poorest agreement with the experiments. Generally good agreement was obtained with the anisotropic DTK calculation and the FAIM program.

Calculations were made of downscattering and absorption heating in the shield.

A series of 19 DTK cases was run to furnish flux information for parametric studies on shields for manned missions.

3. S8ER Calculations

The one-dimensional transport program, DTK, has been used to calculate S8ER criticality. However, results seem to depend strongly on the particular set of cross sections used and no definite conclusions can be drawn at this time. More calculations are planned.

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Figure 1. Shield Test Facility Flux Values

During the quarter the S8ER went critical in the vault with a control drum setting on the fourth drum of 59° compared to the prediction of 51° , a difference of only $-0.25\% \Delta k/k$. Excess reactivity was found to be $6.4\% \Delta k/k$. Since the criticality of S8ER in the vault is so close to that predicted from the critical experiment without the vault, the effect of the vault now seems negligible.

4. Worth of Uranium-234

The uranium-234 in SNAP 8 reactors was found to be worth only $-0.2\% \Delta k/k$ according to two-group hand calculations.

5. Fast Neutron Flux at the Pit Wall

Since neutron doses have been found to have a considerable effect on the nil ductility of reactor pressure vessels, estimates were made of the increase in fast neutron flux due to drums being in the "out" position. According to two-dimensional diffusion theory calculations, rotation of one drum to the outermost position was found to increase the fast flux by a factor of two at the worst point.

6. Impurities in NaK Coolant

The reactivity effects of impurities in the NaK coolant were investigated and found to have little effect on thermal utilization and, thus, on reactivity.

7. Nonuniform Samarium Axial Distribution

Preliminary investigations of the effect of a nonuniform distribution of the samarium prepoison using important weighting techniques showed the distribution to have negligible effect on the samarium worth.

B. THERMAL ANALYSIS

1. S8ER Predicted Operating Temperatures

The present predicted temperatures of the S8ER are based on a theoretical radial and axial core power distribution and on the experimental core outlet coolant flow profile. The experimental coolant flow profiles were obtained on an equivalent hydraulic model. The criteria for shaping the coolant flow was to minimize the maximum ceramic temperature. The outlet coolant profiles for a core power of 300 and 600 kwt, and a core ΔT_c of $200^\circ F$ were determined. The static pressure drop of the central coolant channel operating at a coolant flow of 600 kwt indicated a normalized average coolant flow along the channel of 1.8.

TABLE IV
OPERATING TEMPERATURE LIMITS FOR VARIOUS S8ER POWER LEVELS

			Temperature (°F)		r/R		Z/L	
			Symmetrical	Asymmetrical	Symmetrical	Asymmetrical	Symmetrical	Asymmetrical
S8ER 300 kw	Fuel	Maximum	1400	1380	0.40	0.40	0.90	0.90
		Maximum Average	1390	1370	0.40	0.40	0.90	0.95
	Ceramic	Maximum	1380	1365	0.40	0.40	1.00	1.00
		Maximum Average Peripheral	1375	1360	0.40	0.40	1.00	1.00
	Coolant Outlet	Maximum	1365	1350	0.40	0.40	1.00	1.00
		Minimum	1265	1255	0.90	0.90	1.00	1.00
S8ER 450 kw	Fuel	Maximum	1430	1405	0.40	0.40	0.80	0.80
		Maximum Average	1410	1385	0.40	0.40	0.85	0.85
	Ceramic	Maximum	1385	1365	0.40	0.40	0.90	0.95
		Maximum Average Peripheral	1375	1355	0.40	0.40	1.00	1.00
	Coolant Outlet	Maximum	1360	1345	0.40	0.40	1.00	1.00
		Minimum	1275	1265	0.90	0.90	1.00	1.00
S8ER 600 kw	Fuel	Maximum	1465	1430	0.40	0.40	0.75	0.75
		Maximum Average	1430	1400	0.40	0.40	0.80	0.80
	Ceramic	Maximum	1395	1370	0.40	0.40	0.85	0.90
		Maximum Average Peripheral	1375	1355	0.40	0.40	0.95	0.95
	Coolant Outlet	Maximum	1355	1340	0.40	0.40	1.00	1.00
		Minimum	1270	1260	0.00	0.00	1.00	1.00
S8ER 1000 kw*	Fuel	Maximum	1570	1525	0.40	0.40	0.70	0.70
		Maximum Average	1510	1465	0.40	0.40	0.75	0.75
	Ceramic	Maximum	1445	1410	0.40	0.40	0.80	0.85
		Maximum Average Peripheral	1410	1380	0.40	0.40	0.90	0.90
	Coolant Outlet	Maximum	1370	1350	0.45	0.45	1.00	1.00
		Minimum	1250	1240	0.00	0.00	1.00	1.00

*Coolant inlet temperature = 1050°F

The outlet normalized coolant flow at the central coolant channel for a 600-kwt condition is about 1.6. Therefore, the inlet normalized coolant flow at 600 kwt was assumed to be 2.0 and a probable normalized inlet flow shape was determined. The axial and radial core temperatures were determined using a digital code assuming the coolant flow shaping varied linearly from the inlet to the outlet coolant profile.

Table IV gives the limiting temperatures prevailing for 300, 450, 600, and 1000 kwt, with an inlet coolant temperature of 1100°F and an average coolant temperature increase of 200°F.

The radial and axial location in the core of these maximum temperatures is also given. Two fuel rod configurations are considered: (a) a symmetrical case in which the fuel rod is centrally located within the cladding tube, and (b) an asymmetrical case in which the fuel rod touches the cladding at the point where two fuel elements are touching. A hot channel factor of 1.06 is included in the symmetrical case.

2. SNAP 8 Hydraulic Studies

a. S8ER Flow Profile

The flow profile of the NaK coolant must be shaped in order to obtain optimum heat transfer rates. Results from hydraulic experiments led to the development of a baffle plate for the reactor lower plenum. This baffle plate produces a flow shaping that is almost identical to an ideal flow profile.

Ideally, the normalized peak-to-average flow at the centerline of the core was specified as 1.45 or more for the 600-kw S8ER. The resulting baffle plate, which has four concentric rings of graduating depth in the space between grid plate and baffle plate, yields peak values ranging between 1.46 and 1.67. These higher flow shapings are necessary to maintain flow at the mid-radius regions. The minimum-to-average flow has an ideal value of 0.76, but the axis perpendicular to the inlet has values from 1.08 to 1.26. This high flow is thought to be caused by crossflow to the outer radius and shows up on the particular axis by virtue of the hexagonal geometry.

The four concentric rings shape the velocity profile by restricting the radial flow of the coolant which flows through the center hole of the baffle plate.

The tests were made using a water hydraulic model consisting of a dimensionally identical core model employing Reynolds analogy for hydrodynamic similarity. It was required to remove the upper plenum in these tests and to invert the core. Thus, flow measurements were made at the exit plane of the core and the flow profile obtained was for the outlet plane and was not an average through the core.

b. S8ER Pressure Drop

The above tests yielded flow profiles at the coolant outlet. To determine the inlet shaping, using the designed baffle plate, pressure probes were inserted along the core coolant channels. The resulting peak-to-average flow at the inlet was estimated to be 2.0.

c. Basic Hydraulic Study Using the Salt-Interface Velocity Technique

Basic studies were initiated to obtain mass flow data using two kinds of flow channels: (a) a single tricuspid channel, simulating a SNAP 8 coolant passage; and (b) a single circular channel, which is simpler in cross section and better understood.

Single-flush, single-protruding, and two parallel-plate electrodes have been alternately used as sensing devices at various positions along the channels. They sense a slug of salt solution which is injected at the inlet stream of water, and give information on velocity. Thus, it will be possible to determine the flow profile at different points along the coolant channel, using the experimental method that gives the most reproducible data.

d. S8DS Core Hydraulics

Dummy fuel elements and reflectors have been fabricated and part of them are being instrumented with static pressure probes to measure any radial variation of pressure through the core. The inlet plenum has been modified by incorporating the latest elliptical inlet pipe configuration and will use new flow control equipment being acquired.

From the S8ER experiments, it has been established that a solid inlet baffle with a smooth transition can probably replace the present baffle plate which has concentric rings. In addition, it will be determined if an outlet baffle can produce proper shaping of the flow. The final baffle configuration should yield an optimum flow shaping with a minimum pressure drop through the S8DS core.

C. CONTROL ANALYSIS

The control analysis of the startup of SNAP 8 continued during the report period. The AIREK III digital kinetics code was modified for the analysis of the nuclear startup. A survey of PCS startup parameters was made to determine the permissible range of the startup parameters. A re-examination of the analog models was completed, as well as a comparison of AI and AGC simulations.

1. AIREK III

The latest version of the AIREK digital kinetics code has been modified and tested successfully for use in the SNAP 8 startup analysis. The new version of the code has improved accuracy and speed and, at the same time, is more general than the previous versions. In addition, the code contains subroutines for presenting the results in graphical form.

The modifications of AIREK III for the startup study are the following:

- a) The code prints out the precursor concentrations. These are used for initial conditions at the beginning of the analog runs.
- b) A stepper routine was added to simulate the addition of reactivity by the control drums. The reactivity varies as a quartic function of drum angle.
- c) The printout was modified to conserve machine time and give detailed output in the regions of interest (i. e. , sensible heat generation).

In addition to the reactivity subroutine modification for drum steps, reactivity may be put in as (1) tabular, (2) tabular plus stepper, (3) tabular plus normal (e. g. , a ramp), (4) stepper plus normal, and (5) stepper plus normal plus tabular.

The code furnishes graphical results of power, period, energy, and precursor concentrations as a function of time as well as tabular results, if desired.

2. Comparison of Analog Model Simulations

The accuracy of a discretized model, presently in use at AI, is limited by the number of nodes used in the simulation. A comparison of outlet coolant temperatures was made with 3-, 5-, and 10-node simulations for a 20°F step insertion of inlet coolant temperature at power levels of 400 and 6 kw. The results are shown in Figures 2 and 3. It can be seen from these figures that the

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Figure 2. S8DS Core Model Response to Step in Inlet Temperature ($n_o = 400 \text{ kw}$)

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Figure 3. S8DS Core Model Response to Step in Inlet Temperature (n₀ = 6 kw)

5-node results are in close agreement with the 10-node results. The relative agreement of these two cases indicates that the 5-node simulation is sufficiently accurate for analog work. It was also determined that the assumption of a uniform power distribution instead of the cosine distribution is valid where the outlet coolant temperature response is desired (Figure 2).

3. Frequency Response of the 5-Node Model

A frequency response study was initiated on the SNAP 8 core simulation presently being used in conjunction with the electronic analog computer investigations at AI. This study is basically a time solution of space discretized (5 axial nodes plus entrance fuel node)* energy balance equations, including variable flow and power distribution. Small oscillations of core inlet coolant temperature and core coolant flow rates were considered for two base power and flow levels. These were 400 kw at 32,000 lb/hr NaK flow and 6 kw at 3,200 lb/hr. Frequency response data are shown in Figures 4 through 15.

4. PCS Startup

Recent studies on the analog simulation of the SNAP 8 electrical generation system have resulted in the optimization and selection of various system parameters for the power conversion system startup. The limitations imposed on the startup transient include:

- a) Rate of change of reactor fuel or coolant temperature must not exceed $150^{\circ}\text{F}/\text{min}$,
- b) Maximum value of reactor outlet coolant temperature must not exceed the scram level of 1375°F ,
- c) Minimum value of reactor outlet coolant temperature at rated Hg flow must not be less than 1270°F , and
- d) Peak reactor power must not exceed the scram level of 130% of rated power (rated power is presently 400 kw).

The system was simulated on a combination of two PACE 231-R analog computers. The equations used for the reactor include: 6-group reactor kinetics, 5-node plus entrance-node reactor heat transfer, reactivity feedback from fuel and grid plate temperatures, and a simulation of control system action. The

*"Transient Response of Two Reactor Heat Transfer Models," D. G. Mason, NAA-SR-7938 (to be published).

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Figure 4. Frequency Response of Normalized Power to Sinusoidal Changes in Inlet Coolant Temperature for S8DS Analog Core Model, 6 kw

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Figure 5. Frequency Response of Average Fuel Temperature to Sinusoidal Changes in Inlet Coolant Temperature for S8DS Analog Core Model, 6 kw

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Figure 6. Frequency Response of Outlet Coolant Temperature to Sinusoidal Changes in Inlet Coolant Temperature for S8DS Analog Core Model, 6 kw

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Figure 7. Frequency Response of Normalized Power to Sinusoidal Changes in Inlet Coolant Temperature for S8DS Analog Core Model, 400 kw

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Figure 8. Frequency Response of Average Fuel Temperature to Sinusoidal Changes in Inlet Coolant Temperature for S8DS Analog Core Model, 400 kw

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Figure 9. Frequency Response of Outlet Coolant Temperature to Sinusoidal Changes in Inlet Coolant Temperature for S8DS Analog Core Model, 400 kw

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Figure 10. Frequency Response of Normalized Power to Sinusoidal Changes in Coolant Flow for S8DS Analog Core Model, 6 kw

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Figure 11. Frequency Response of Average Fuel Temperature to Sinusoidal Changes in Coolant Flow for S8DS Analog Core Model, 6 kw

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Figure 12. Frequency Response of Outlet Coolant Temperature to Sinusoidal Changes in Coolant Flow for S8DS Analog Core Model, 6 kw

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7568-55203

Figure 13. Frequency Response of Normalized Power to Sinusoidal Changes in Coolant Flow for S8DS Analog Core Model, 400 kw

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7568-55204

Figure 14. Frequency Response of Average Fuel Temperature to Sinusoidal Changes in Coolant Flow for S8DS Analog Core Model, 400 kw

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Figure 15. Frequency Response of Outlet Coolant Temperature to Sinusoidal Changes in Coolant Flow for S8DS Analog Core Model, 400 kw

coolant transport delay for the primary loop piping was approximated by two separate 9-node cascaded delays with the delay times dependent on NaK flow. The heat exchanger simulation used was developed by AGC. It consists of a simulation of the preheater, boiler, and superheater sections and the model describes the inlet and outlet temperatures of NaK and Hg for each section. Provision is made for varying NaK and liquid Hg flow. Hg vapor flow and pressure are determined in this simulation.

The effects of variations of the following system parameters were investigated over the ranges indicated. (Initial conditions are defined as the conditions existing immediately before Hg flow begins.)

- a) Initial NaK flow—10 to 80% of rated flow (rated NaK flow = 8.89 lb/sec),
- b) Initial core temperature rise — 12.7 to 254° F,
- c) Initial power — 10 to 160 kw,
- d) Time for first Hg flow rise (0 to 40%) — 0 to 90 sec,
- e) NaK rise time — 0 to 90 sec,
- f) Control step interval — 120 to 360 sec,
- g) Hg rise time (40 to 100%) — 300 to 600 sec,
- h) Delay of NaK rise — 0 to 75 sec,
- i) Hg injection level — 20 to 50% of rated flow (rated Hg flow = 2.53 lb/sec),
and
- j) Fuel temperature coefficient — -0.05 to -0.15¢/° F.

Within the range studied for each of the variables, there is a "permissible" range throughout which all system limitations can be met. Using this range of values, restricting the system to low initial power levels (i. e., low heat losses and better thermal efficiency) results in an "optimum set" of parameters (Table V). The "optimum" conditions result in a maximum reactor outlet coolant temperature of about 1360° F, a maximum rate of change of NaK temperature within the core of about 110° F/min, a peak reactor power of about 117% of rated power, maximum rate of change of fuel temperature <150° F/min, and a minimum reactor outlet coolant temperature of about 1280° F.

TABLE V
OPTIMUM REACTOR OPERATING CONDITIONS

Parameter	Permissible Range	Optimum
Initial NaK Flow	$\geq 40\%$ of rated flow	50%
Initial core temperature rise	$50^\circ\text{F} < \Delta T < 100^\circ\text{F}$	51°F
Initial power	fixed by above	50 kw
Hg flow rise time for 0 to 40% of rated	≥ 90 sec	90 sec
NaK rise time	≥ 60 sec	70 sec
Control step interval	≥ 200 sec	200 sec
Hg flow rise time from 40 to 100%	≥ 300 sec	500 sec
Delay of NaK rise	≤ 75 sec	40 sec
Hg injection level	$\leq 50\%$ of rated flow	40%

Some of the results of the parameter studies pertaining to PCS startup are presented graphically in Figures 16 through 22. The curves labeled " T_{c5} -Nominal" and the curves in Figure 20 are based on an initial reactor outlet NaK temperature of 1300°F . Since the present control system design specifies a temperature deadband upper limit of $1320 \pm 3^\circ\text{F}$, it is possible for the initial outlet temperature to be as high as 1323°F . Because one of the limitations on the PCS startup transient is that reactor outlet temperature must not exceed 1375°F , a limited number of analog runs was made with an initial outlet temperature of 1323°F . The effect of this initial difference was then added to " T_{c5} -Nominal" and plotted as " T_{c5} -estimated deadband effect."

D. SHIELDING ANALYSIS

1. Embrittlement of S8DS Reactor Pit Liner

During the previous period, an extensive analysis was made of the nil ductility transition (NDT) temperature of the carbon steel containment vessel for the reactor pit in the S8ER test facility. Contributions to this study by shielding personnel included (a) the determination of the integrated fast neutron flux at various locations on the surface of the vessel, and (b) the nuclear heat generation rates in the vessel.

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Figure 16. Effect of Initial Core Temperature Difference

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Figure 17. Effect of Initial Power Level

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Figure 18. Effect of Initial Mercury Flow Rise Time

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Figure 19. Effect of Initial NaK Flow on Rate of Change of Reactor NaK Temperature

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Figure 20. Effect of Initial NaK Flow on NaK Temperature at Reactor Outlet

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Figure 21. Effect of Initial NaK Flow, Initial Core Temperature Difference = 101.6°F

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Figure 22. Effect of Initial NaK Flow, Initial Core Temperature Difference = 50.8°F

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Figure 23. Nuclear Heating in S8DS Reactor Pit Wall

During this report period, the above-listed calculations were modified to correspond to the reactor pit liner in the S8DS test facility. The integrated flux due to fast neutrons (>1 Mev to effect the NDT temperature) was estimated to be 6.7×10^{11} nvt/kwh. This estimate corresponded to (a) a dose point at a radius of 30 in. in the core midplane and (b) an effective reflector thickness of 3 in. (for a reflector control drum in the outermost position, the dose increased by about a factor of 2).

The calculated values of the nuclear heat generation rates in the S8DS reactor pit wall are presented in Figure 23. Due to a lack of information on the chemical composition of therman, this material was assumed to behave the same as concrete.

2. Unmanned SNAP 8 Spacecraft Shield Design

The shielding aspects of an unmanned SNAP 8 spacecraft design were investigated. Calculations were made of the neutron and gamma doses and spectra corresponding to a 600-kwt shield designed to protect a 20-ft-diameter payload located 40 ft from the center of the core of a SNAP 8 reactor. Two radiation level criteria — one to ensure proper functioning of payload equipment, the other to ensure proper functioning of PCS components — were imposed:

Case 1: A dose of 1×10^{11} nvt fast neutrons and 1×10^6 rads (C) gammas at the payload.

Case 2: A dose of 5×10^{12} nvt fast neutrons and 5×10^7 rads (C) gammas at a distance of 1 ft from the shield.

The resulting shield design specifications are summarized in Table VI. Table VII presents the radiation leakage spectra for these designs while Figures 24 and 25 represent the spatial variation of the fast neutron and gamma doses between the payload and the bottom of the shield.

As indicated by the data in the two figures, the shield design which satisfies the payload dose restrictions (Case 1) results in radiation levels which are about twice as high as the PCS dose restrictions (Case 2). The shield weight differential to satisfy the latter requirements is about 360 lb, as seen in Table VI.

Limited investigations were made of the doses due to two sources of radiation external to the SNAP 8 reactor-shield assembly: (a) gammas from activated NaK in the PCS boiler and (b) fast neutrons which scatter from portions

TABLE VI
SNAP 8 SHIELD DESIGN DATA

Shield Designation Number	1	2
<u>Payload Dimensions</u>		
Diameter, ft	20	20
Axial distance from reactor core center, ft	40	40
<u>Reactor Operating Conditions</u>		
Reactor power level, kw	620	620
Power delivered (by NaK) to PCS boiler, kw	600	600
Reactor operation time, hr	10,000	10,000
<u>Radiation Levels Due to Reactor Sources</u>		
Fast neutron dose at payload, nvt	1×10^{11}	4.3×10^{10}
Fast neutron dose at 1 ft below shield, nvt	1.2×10^{13}	5×10^{12}
Gamma dose at payload, rad (C)	1×10^6	5.1×10^5
Gamma dose at 1 ft below shield, rad (C)	1.0×10^8	5×10^7
<u>Shield Dimensions</u>		
Length of conical neutron shield (LiH), in.	30.6	33.0
Length of spherical segment neutron shield (LiH), in.	3.9	3.9
Maximum diameter of conical shield, in.	46.8	47.8
Minimum diameter of conical shield, in.	33.2	33.2
Axial distance between core center and shield, in.	16.3	16.3
Thickness of exterior gamma shield (W), in.	0.85	0.85
Diameter of exterior gamma shield, in.	9.7	9.7
Thickness of interior gamma shield (W), in.	0.3	0.6
Diameter of interior gamma shield, in.	37.6	37.6
<u>Shield Assembly Weight</u>		
Neutron shield (lithium hydride), lb	1112	1228
Gamma shield (tungsten), lb	263	492
Shield vessel and structure, lb	399	414
Cable harness, lb	15	15
Destruct charge, lb	11	11
Total shield assembly weight, lb	1800	2160

TABLE VII
SPECTRA OF RADIATION LEAKAGE
FROM SHIELDS

Shield Designation Number	Shield Leakage (%)	
	1	2
Gamma Dose Leakage Spectrum		
0 to 1 Mev	0.1 [*]	0.0 [†]
1 to 2	6.8	3.9
2 to 3	48.7	52.4
3 to 4	6.4	4.2
4 to 5	2.9	1.8
5 to 6	2.1	1.4
6 to 7	12.6	13.9
7 to 8	17.0	19.5
8 to 9	3.1	2.6
9 to 10	0.2	0.2
Total	99.9	99.9
Neutron Leakage Spectrum		
3 to 10 Mev	14.9 [§]	24.4 [§]
1.4 to 3	20.1	26.1
0.9 to 1.4	7.9	8.7
0.4 to 0.9	10.5	10.7
0.1 to 0.4	9.7	7.6
17 to 100 kev	8.1	5.9
3 to 17	6.5	4.2
0.55 to 3	5.8	3.6
0.1 to 0.55	5.5	3.2
30 to 100 ev	3.5	1.9
10 to 30	2.7	1.4
3 to 10	2.2	1.1
1 to 3	1.3	0.6
0.4 to 1	0.7	0.3
0.1 to 0.4	0.4	0.2
(thermal) to 0.1	0.2	0.1
Total	100.0	100.0

*Spectrum at payload

†Spectrum at 1 ft below shield

§Spectrum at bottom of shield.

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Figure 24. Variation of Fast Neutron Dose Between Payload and Bottom of Shield (10,000 hr at 620 kwt)

Figure 25. Variation of Gamma Dose Between Payload and Bottom of Shield (10,000 hr at 620 kwt)

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of the NaK pipes which pass around the shield and are thus outside of the shielded conical angle. Lack of data regarding the design of the PCS boiler prompted the use of the overly conservative assumption that there was no self-absorption in the NaK or in the boiler walls. A second assumption was made that the entire Na^{24} activity (260 curies) in the reactor coolant was concentrated at a point 3 ft distant from the PCS dose point. Under these assumptions it was calculated that the dose from the activated NaK was about 4% of the design value at the payload and about 10% of the design value at the PCS. Neutron scattering from exposed portions of the NaK pipes was estimated to produce very high radiation levels over the outer portions of the payload surface, varying from about 60% of the design dose at the midradius to about 60 times the design dose at the full radius of the payload.

3. Manned SNAP 8 Spacecraft

It has been established recently that NASA does not intend to define a specific mission objective for SNAP 8 until a series of mission studies is completed about two

years hence. Until that time, then, no well-defined radiation shielding criteria can be established. On the other hand, it may be anticipated that once the specific mission has been selected, a limited time will be available to design a shield.

Thus, AI is developing a diversified shielding capability which will permit the rapid design of SNAP 8 shields for any proposed mission.

4. Shield Experiments

The effect of transverse leakage as a correction to the measurements of the lithium hydride attenuation coefficient was studied. The attenuation of fast neutrons as detected by a Hurst dosimeter was measured using slabs of 3-ft and 5-ft dimensions in the transverse direction. The neutron intensity is shown as a function of shield slab thickness in Figure 26. The data is normalized to

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Figure 26. Neutron Intensity vs Shield Slab Thickness

TABLE VIII
REACTOR VESSEL ANALYSIS MARGINS OF SAFETY

Part	Material	Design Condition	Margin of Safety*
Vessel — Lower Head	316 SS	T = 1130° F P = 62.5 psi	+0.73 (creep)
Vessel — Lower Body	316 SS	Launch Loads	High
	316 SS	Operational T = 1300° F P = 62.5 psi	+0.34 (creep)
Upper Head at Outlet	316 SS	M = 600 in.-lb T = 1330° F	+0.50 (creep)
Vessel — Upper Head Joint	316 SS	M = 600 in.-lb T = 1330° F	+0.93 (creep)
Vessel — Upper Joint	316 SS	M = 600 in.-lb T = 1330° F	High (creep)
Bottom Grid Plate	Hastelloy-C	Launch Load	High (tensile)
Reflector Support Bracket — Bolt	Inconel-X	Launch Load	High (bending shear)
	NAS 501 Bolt	Launch Load	High (shear)
Bracket — Scram Pivot	Inconel-X	Launch Load	High (bending shear)
Vessel Support	316 SS	Launch Load	High (tensile shear)
Vessel Inlet	316 SS	M = 600 in.-lb	+0.82 (creep)

*"High" Margin of Safety (M. S.) signifies all M. S. higher than 100%.

the same incident flux on the shield. The results of the previous measurements on combinations of 3-ft and 5-ft slabs is also shown. From these results it is concluded that transverse leakage is not a significant correction to the shield attenuation coefficient measurements.

E. STRESS ANALYSIS

1. SNAP 8 Reactor Vessel

A detail stress analysis of all major structural components, covering all significant conditions of operation, was completed. The stress analysis report establishes the adequacy of the S8DRM-1 reactor vessel to retain its dimensions, integrity, and ability to function under anticipated loadings and environmental conditions. A summary of the margins of safety is presented in Table VIII.

Margins of safety are based on the (a) ultimate strength or 1.25 times the yield strength of the material, whichever is lower, and (b) the creep strength of the material. The allowable creep strengths used in the analysis were: (a) the stress producing a creep rate of 1%/10,000 hr; or (b) the stress for creep rate of 1%/100,000 hr.

The design loads used in the analysis were obtained by multiplying the maximum probable load by a factor of safety of 1.25.

2. SNAP 8 Reactor Shield

A similar analysis of the S8DRM-1 shield was completed and the margins of safety are presented in Table IX.

TABLE IX
REACTOR SHIELD ANALYSIS MARGINS OF SAFETY

Part	Material	Design Condition	Margin of Safety*
Shield, Upper Structure, Edge Discontinuity Stress	316 SS	Launch Loads	+0.40 (yield)
Inner Head and Cap Assembly Membrane Stress-Surface	347 SS	Launch Loads	+ High (yield)
Inner Head and Cap Assembly Edge Discontinuity Stress	347 SS	Launch Loads	+ High (yield)
Shield Body Induced Hoop Stress	347 SS	Launch Loads	+ High (yield)
Inner Head and Cap Assembly Critical Buckling Due to Shield Fabrication	347 SS	Shield Fabrication T = 1330°F Pu = 1.50 psi	+ High (buckling)
Head Assembly Lower Ring Stress	347 SS	Launch Loads	+0.22 (yield)
Head Assembly Lower Edge Discontinuity Stress in Diaphragm	347 SS	Launch Loads	+0.33 (yield)
Shield Attachment Ring Bolt Stress		Launch Loads	+ High (yield)
Shield Attachment Ring	347 SS	Launch Loads	+0.85 (yield)

*"High" Margin of Safety (M. S.) signifies all M. S. higher than 100%.

V. FUEL ELEMENT DEVELOPMENT AND FABRICATION

A. INTRODUCTION AND SUMMARY

It is the objective of the fuel element development to develop reliable fuel elements for the 600 kw_t SNAP 8 reactor. The performance goals established for the fuel elements are:

- 1) The peak power density of the fuel is 5.5 kw_t/ft of fuel rod length at a peak central temperature of 1600°F for 10,000 hr. The peak fuel burnup is about 0.4 metal at. %.
- 2) The hydrogen loss from the fuel elements must be less than 5% of that in the fuel alloy during the 10,000 hr the reactor is in operation. Thus, the hydrogen barrier coating and closure seal must withstand all the prelaunch thermal and mechanical checkout; the acceleration, vibration, and shock due to launch; the thermal shock of ~300°F/min during startup; and a 1500°F peak cladding operating temperature for 10,000 hr in a total flux of 2×10^{21} nvt fast neutrons. The fuel element cladding must withstand all of the above plus resist corrosion of the NaK environment and maintain dimensional integrity over the life of the core.
- 3) Each fuel element should have a 99% reliability of meeting these performance goals. Some redundant fuel elements are provided in the core so that failure of the hydrogen barrier in a few elements would not cause failure of the reactor.

The first step toward achieving these goals was made with the development and fabrication of the S8ER fuel elements. These elements contained the AI-8763D ceramic hydrogen diffusion barrier coating. Although this coating was the best available at the time the S8ER was fabricated, there was concern as to the ability of the coating to withstand repeated thermal shocks and its ability to withstand thermal cycling in the coating softening range of 1400 to 1450°F.

The suspected inability of the AI-8763D hydrogen barrier to withstand thermal cycling and shocks has been verified on 64 fuel elements tested to date. These elements have exhibited significant increases in hydrogen leak rates as a function of thermal cycling.

This limitation appears to have been solved with the new hydrogen barrier, SCB. Additional tests have been conducted with no coating failures to date. Previous quarterlies have reported the formulation and early testing of the SCB coating. Additional tests conducted during this quarter also show the coating to be equal to the AI-8763D with relation to hydrogen retention and considerably superior as far as ability to withstand thermal shocks and cycles. The high temperature stability in hydrogen of the SCB coating is also much greater than the AI-8763D. One closed-end cladding tube has now been at 1900°F in 1 atm. of hydrogen for over 2360 hr with essentially no change in permeation rate.

The major development effort is now being placed on forming and qualifying the final closure seal to make complete fuel elements using the SCB coating. The ability to make a hydrogen-leak-tight seal has already been demonstrated. However, the detailed process variables must still be determined.

Other areas of fuel element development and fabrication progressed fairly close to schedule. Development work on the single piece closure seal for use with the AI-8763D coating has been completed. Similar developments with the SCB coating reached an advanced stage. This closure was incorporated into the S8DRM-1 fuel element design which was released for fabrication during the quarter.

Irradiation of NAA-53-3, the first carbon-modified fuel experiment, was completed and the capsule was returned to the AI hot cell. NAA-68-1, the AI-8763D coated membrane irradiation, is still in the ETR and had received a calculated fast flux exposure of 3.11×10^{20} nvt at the end of the first 1-1/2-in. cycles. NAA-77-1, the SCB coated membrane for inpile permeation testing is in the latter stages of assembly, while the subscale fuel elements for the NAA-115-1 and NAA-117-1 irradiations have been 75% completed. The hardware and control systems for the NAA-115-1 capsule were ordered. The specific details of these and other experiments are covered in detail in later sections of this report.

B. FUEL ELEMENT DEVELOPMENT

1. Fuel

a. Fuel Structural Studies

In order to better understand the role played by various constituents on the mechanical properties and irradiation resistance of the fuel, a study of microstructure has been undertaken. This study is aimed primarily at the identification of precipitate phases. Three distinct types of precipitates have been found using optical metallography and electron microscopy techniques. Table X lists the pertinent features.

TABLE X
RESULTS OF FUEL MICROSTRUCTURE STUDIES

Description	Size	X-ray Diffraction Composition
Transgranular platelets	5 to 10 micron	ZrC
Inter- and transgranular spheroids	0.5 to 1 micron	U-UO ₂ mixture
Spheroids	0.05 to 1 micron	Undetermined

Work has recently begun utilizing the electron microprobe analyzer to verify the composition of precipitates as determined by x-ray diffraction.

b. Fuel Nondestructive Tests

A radiographic technique was developed for inspecting hydrided fuel rods for internal flaws. The 2-Mev x-ray machine at the Southern California Cancer Center was employed as the radiation source. A radiographic thickness sensitivity of 2% (0.010 in.) and a detail sensitivity of 3% (0.040-in.-diam x 0.008-in.-deep hole) was achieved using high definition film. Radiographic sensitivity was measured with a special hydrided fuel penetrometer containing milled steps and drilled holes. The flat steps were designed to disclose thickness sensitivities from 1.0 to 7.2%. The drilled holes were included to measure detail sensitivity in the range of 0.7 to 4.0%. Trial exposures were made with the x-ray machine adjusted to both 1.5 and 2.0 Mev. As was expected, slightly higher film contrast was obtained at the lower energy level. However, this increased

contrast was offset by decreased detail sensitivity at 1.5 Mev because the focusing coil in the x-ray machine had been adjusted for minimum focal spot size (for maximum detail sensitivity) at 2 Mev. This radiographic technique was capable of disclosing shallow cracks and extrusion marks in fuel rods with known defects. However, the radiographic images of the cracks became increasingly indistinct as the plane of the crack was oriented out of parallelism with the x-ray beam.

All fuel rods for the irradiation experiments, NAA-115 and NAA-117, were examined using this new technique. The films from 6 exposures, in 30° increments from 0 to 150° on each rod, showed no defects in the specimens.

Ultrasonic testing techniques are being developed for rapidly examining as-cast U-Zr fuel billets. Contact-type search units have successfully disclosed shrinkage cavities in the top of these billets. These same billets have been examined after extrusion and the elongated remnants of the cavities were readily identified. Eventually, this ultrasonic test will be performed in the melt room, not only to permit more accurate cropping of defective material from the top of billets, but also to aid in the development of improved casting techniques resulting in a better product at lower cost.

All electronic and nuclear components have been procured for the prototype neutron-scatter gage for measuring the hydrogen content of hydrided fuel rods. In addition to the transistorized scaling equipment, two new neutron detectors containing 10 atmospheres of He^3 are being compared with the detectors containing BF_3 . A biological shield to protect personnel from the seven-curie Pu-Be neutron source has been mocked-up using plywood and paraffin. The inside dimensions of the shield are 36 x 36 x 18 in. It is surrounded by 8 inches of paraffin. Experiments are underway to determine how much neutron scattering is caused by the shield itself.

An experiment was performed to correlate uranium analyses in fuel samples as determined by the x-ray fluorescence method and the autoradiometric uranium analyzer (AUA) method. The sample areas for both methods were the ends of selected hydrided fuel rods which contained from 6.5 to 10.3 wt % uranium. Previous comparisons of analysis by the two methods had produced a very low correlation coefficient of 0.307. However, the AUA analyses had been performed on the sides of hydrided fuel and the x-ray fluorescence analyses had sampled cross-sectional areas of wafers cut from unhydrided rods. Using the

data from the analyses of the same samples, a correlation coefficient was calculated to be 0.914 (data obtained during normal working hours). When x-ray fluorescence measurements were made between 3 to 6 a. m. (lower background interferences), the correlation coefficient was calculated to be 0.976. The calculated precision of the two methods was as follows: AUA method — 0.023 wt % U; x-ray fluorescence method, 3 to 6 a. m. — 0.109 wt % U; and daytime — 0.216 wt % U. Additional statistical tests were made which showed that the variance (precision squared) in uranium analyses using the AUA method was significantly smaller than the variance using the x-ray fluorescence method.

c. Physical Property Measurements

Further studies of stress-strain diagrams have been made on the fuel. From this work it has been demonstrated that the hydrided fuel in the epsilon phase exhibits some plasticity at 1250°F and above. At 1450°F, for example, the fuel shows 0.5% deformation at rupture. Metallographic and fractographic studies of tensile samples have given some insight into the modes of failure of the hydrided material. The delta and epsilon phase zirconium — 10 wt % uranium hydrides fail transgranularly. Cracks tend to propagate at an oblique angle to the epsilon twin bands. This preferred direction of crack propagation explains why the columnar grained material of nonmodified alloys is weaker and more notch-sensitive than carbon modified alloys. The twin bands in the columnar grains are usually oriented at oblique angles to the radial direction of the rod. Since the long axis of the columnar grains is in the radial direction, cracks can propagate long distances without encountering grain boundaries, which appear to be effective barriers to crack propagation. Such cracks, which probably initiate at surface imperfections, apparently gain enough momentum to pass through the equiaxed grains in the rod core without changing direction significantly.

2. Hydrogen Barrier

a. Nucleated Glass (SCB)

Permeation, microstructure, and property studies show that SCB hydrogen barrier coatings are more stable and are at least as impermeable to hydrogen as coatings presently in use. SCB is, therefore, a prime candidate for the S8DS-1 fuel elements. Before a new coating can be adopted, it must be

proven dependable by performance testing. Coating techniques must also be developed to produce a product which meets quality control standards. Both of these programs were initiated this quarter.

Progress has been made in eliminating small defects in the SCB coating at the blind end of the cladding tubes. Reduction of these defects is attributed to a longer drying time for the bisque prior to firing. Improved quality of the chromized layer on the cladding tubes may also be a contributing factor. Process changes in the coating technique, such as a heavy sandblasting, chemical cleaning, and etching of the metal substrate, did not appear to improve the coating quality.

Excessive bowing of the cladding tubes during firing of the SCB coating has been eliminated by a new firing rack design. The rack supports the cladding tubes horizontally at two points, 10-1/2 inches apart. Firing of the tubes in the vertical position has also been effective in preventing bowing.

Chromized Hastelloy-N cladding tubes undergo excessive oxidation of the outside surface during the SCB firing operation at 2100°F. Two techniques were found to be successful in reducing this oxidation:

- 1) A commercial protective coating can be painted on the tubes before firing.
- 2) A special stainless-steel jacket was designed to enclose the tubing during firing.

Initial tests show that both methods are successful and practical. The steel jacket also holds the tube rigid and prevents bowing. However, tubes within the steel jacket cool at a much slower rate after firing and it has not yet been established whether or not this slow cooling rate affects the coating quality.

A study was made to determine if an organic suspending medium for SCB frit had merits over the conventional water-clay system. This organic solution is a commercial product used successfully by Battelle Memorial Institute (BMI) in their coating development subcontract work. Use of the organic solution eliminates water, which could be absorbed on the frit particles and produce bubbles and defects in the coating. The disadvantages to its use are that it requires a longer and more elaborate drying schedule, and the slip cannot be applied in layers greater than one-half mil.

Cladding tubes were coated with one, two, and four layers using this organic slip. Test results and coating evaluations indicated that the organic slip had no advantage over the conventional method of application.

b. Nucleated Glass (SCB) Tests

A total of twenty-two (22) additional SCB-coated cladding tubes have been permeation tested for use in endurance and thermal cycling tests. Table XI lists the hydrogen permeation rates for these cladding tubes at 1500°F under one atmosphere hydrogen. About half of the tubes contained SCB coatings which were crystallized prior to test. The remaining tubes had noncrystallized coatings. Three of the tubes in Table XI had high permeation rates in the as-fabricated condition and were rejected. Initial permeation rates on the remaining nineteen (19) tubes show some difference between crystallized and noncrystallized coatings. Although previous tests have shown a decrease in the permeation rate of a given membrane when crystallized in hydrogen, the average as-fabricated leak rate from membranes crystallized in air was higher in these tests. The average of nine crystallized coatings is 2.48×10^{-3} cc/hr/cm², while the ten noncrystallized coatings averaged 1.62×10^{-3} cc/hr/cm². These results may be due to more minor defects in the crystallized coating. The noncrystallized coating appears more favorable because of the above results and the initial closure studies, which showed that it was easier to form the glass-to-glass seal using the noncrystallized material.

Ten of these coated tubes have been placed in endurance testing furnaces and are under long-term testing at 1500°F. Half of the tubes contain crystallized coatings while the other half are in the noncrystallized condition. The hydrogen permeation rates of these tubes can be monitored individually or in clusters of five without removing them from the furnaces. The average hydrogen permeation rate on the ten tubes after approximately 700 hours of operation at 1500°F was 2.6×10^{-3} cc(STP)/hr/cm².

Experiments have been initiated on the mechanism of hydrogen permeation through the SCB coating. In the first test, an SCB permeation membrane was taken to 1900°F and the permeation rate measured at 7.5 and 10 atmospheres of hydrogen pressure. At this temperature and these pressures, the Hastelloy-N cladding wall underwent about 1% creep. The membrane failed on the flat end of the top cap which had been machined to a 0.010-in. thickness.

TABLE XI
HYDROGEN PERMEATION RATES OF
SCB-COATED CLADDING TUBES

Tube No.	Coating Condition*	Permeation Rate, ϕ [cc(STP)/hr/cm ² at 1500°F and 1 atm H]
E-781	NC	1.1×10^{-3}
773	NC	1.7×10^{-3}
827	NC	6.6×10^{-2} (disqualified)
809	NC	2.5×10^{-3}
835	NC	1.2×10^{-3}
848	NC	1.7×10^{-3}
839	NC	1.9×10^{-3}
890	NC	1.1×10^{-3}
900	NC	1.5×10^{-3}
800	NC	1.9×10^{-3}
838	NC	1.6×10^{-3}
822	C	2.4×10^{-3}
764	C	6.5×10^{-2} (disqualified)
717	C	7.0×10^{-3} (disqualified)
808	C	2.1×10^{-3}
762	C	2.4×10^{-3}
777	C	1.8×10^{-3}
846	C	2.0×10^{-3}
726	C	1.8×10^{-3}
844	C	3.3×10^{-3}
893	C	2.5×10^{-3}
759	C	3.0×10^{-3}

*NC = Noncrystallized
C = Crystallized

Examination of the coating on the wall of the tube revealed that it was still in excellent condition. It is doubtful that failure would have occurred if the membrane contained a reference top cap having 0.080-in. thickness.

A plot of the permeation rate vs pressure obtained for this membrane before failure occurred is shown in Figure 27. The plot shows a straight-line function with a pressure coefficient equal to one. This suggests that hydrogen

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Figure 27. Permeation Rate vs Hydrogen Pressure for SCB at 1900°F

permeates through the membrane as the molecular species. Experimental work on composite enameled membranes of Solaramic S14-35A and AI-8763D have always shown pressure coefficients of about 0.75 for temperatures of 1200 to 1500°F. It is postulated that the deviation from unity is associated with softening of the enamel.

Another SCB membrane has been on long-term permeation test for 3760 hr at 1500 and 1900°F under 1 atm of hydrogen pressure. The membrane has been at 1900°F for 2360 hr and the current permeation rate at this temperature is 1.3×10^{-2} cc(STP)/hr/cm² at 1900°F and 1 atm H₂.

Thermal cycling tests on two SCB membranes which were reported in detail last quarter were concluded with 102 cycles between 1500°F and room temperature. The permeation rates after completion of the cycling tests were 2.5×10^{-3} and 3.2×10^{-3} cc(STP)/hr/cm² at 1500°F and 1 atm H₂. Initial permeation rates on these two membranes were 2.2×10^{-3} and 3.3×10^{-3} , respectively. Two new SCB-coated membranes, one crystallized and one noncrystallized, are now under test and have received 25 cycles. The permeation rates on these membranes have not changed as a result of the testing. The membranes will receive a total of 100 cycles before the testing is completed.

c. Glass Enamel (AI-8763D) Tests

Five membranes containing the AI-8763D glass enamel have completed 10,500 hr at temperatures ranging from 1200 to 1500°F. Internal hydrogen pressures during this time ranged from 1 to 5 atm (abs). The latest permeation measurements of these membranes (grouped together in a cluster during

testing) indicated a hydrogen loss rate of 1.1×10^{-3} cc(STP)/hr/cm² at 1450°F and 4 atm of internal hydrogen pressure. The membrane cluster has been removed from its previous test system and reloaded in a new test system. The new system is capable of measuring the permeation of the membrane individually or as a group.

d. Nondestructive Tests Related to Coating

Both danger coefficient and neutron activation techniques have been undergoing development and evaluation in order to perfect a nondestructive method for measuring the total and incremental poison content of barrier coatings. Using data from danger coefficient experiments, a statistical analysis was completed to correlate L77 reactor control and "worth" with the weight of Sm₂O₃ in each coated tube. Five tubes containing from 0 to 200% of the specified amount of Sm₂O₃ in the slip were employed in the evaluation. A correlation coefficient of 0.995 was estimated from danger coefficient (percent reactivity or control rod "worth") and weight of poison data. This correlation coefficient value indicated close to a 1:1 relationship between the two methods.

e. Alumina Coating

A Hastelloy-N tube coated with aluminum oxide was produced by the Stanford Research Institute (SRI) at the close of their contract in April. The tube was coated by a vapor deposition process, using the reaction:

The tube, although of poor quality, was supplied for testing. The coating appeared porous and nonadherent. Permeation tests verified this observation with a rate of 2.2×10^{-1} cc(STP)/hr/cm² at 1500°F and 1 atm H₂.

A program to further investigate the alumina coating and develop production processes was expanded at AI during this quarter. Temperatures and reactant concentrations for depositing α -Al₂O₃ have been established. Using this as the starting point, techniques are being developed for deposition in a blind end tube. The hot aluminum chloride vapor is carried by carbon dioxide and hydrogen into the hot reaction zone. Here the exchange reaction takes place with the Al₂O₃ depositing on the walls. The spent reactants are carried

away by a quartz exhaust tube. Seven vapor deposition runs have been made. The conditions of these experiments are shown in Table XII.

TABLE XII
RESULTS OF Al_2O_3 VAPOR DEPOSITION TESTS

Run Number	Deposition Time (min)	Furnace Temperature ($^{\circ}\text{F}$)	Vapor Pot Temperature ($^{\circ}\text{F}$)	Time (min)	Flow Rate H_2 CO_2 (cc/min)	
1	40	2110	245	40	100	10
2	93	2160	150 to 340	60 33	100 200	10 20
3	150	2000	240 to 332	60 90 30	100 200 100	10 20 10
4	180	2000 last 2200 hr	197 to 375	30 30 90	200 300 400	20 30 40
5	120	2060	225 to 376	120	500	50
6	120	1600	292 to 300	120	500	50
7	288	2150	345 to 366	288	500	50

The coatings on the tubes were examined with a borescope. Runs 1, 2, and 7 had a uniform coating. Run 5 appeared to have some flow lines. Runs 3, 4, and 6 showed evidence of spalling at the reactant inlet portion of the tubes, although the lower portions appeared uniform. The deposits were examined both metallographically and by x-ray diffraction. With careful polishing, it was possible to determine the thickness of the coating on Tube No. 7. It was a uniform 0.4 mils. Earlier attempts to identify alpha-alumina were unsuccessful as the x-rays penetrated the coating and, thus, gave a very diffuse pattern.

The membranes from Runs 2 and 7 were hydrogen permeation tested at 1500°F , and one atmosphere of hydrogen. The leak rates were 1.5×10^{-1} and 1.4×10^{-2} cc(STP)/hr/cm² for 2 and 7, respectively. These permeation rates are a factor of 100 better than those for bare metal under comparable conditions.

f. Advanced Coatings

Three silicon-carbide-coated tubes have been purchased from a vendor for evaluation of this material as a hydrogen barrier. Molybdenum tubing was used as the metallic substrate to more closely match the thermal expansion of the SiC, and to withstand application temperatures in excess of 2300°F. The molybdenum metal was found to become embrittled by the process and attempts to close the end of the coated tubes for permeation testing have been unsuccessful. Mechanical end seals have been designed to eliminate weld-

ing and machining of the brittle molybdenum. The hydrogen barrier qualities of the SiC will be determined next quarter.

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Figure 28. Permeation Rate vs Reciprocal Temperature, Aluminized Hastelloy-N

The process of applying an aluminum-nickel intermetallic diffusion layer on Hastelloy-N claddings is still being investigated. Problems have been encountered in applying uniform coatings to full-length blind-end cladding tubes. Interest in this process is as a possible substitute for the present chromizing process.

The potential of this diffusion coating as a hydrogen barrier was investigated. It was found to be somewhat better than bare metal but not effective for our requirements. Tests were carried out at 1 atm H₂ pressure from 600 to 1500°F. Soaking at 1500°F caused a change in the activation energy of permeation as shown in Figure 28.

3. Closures

a. AI-8763D Coating Closures

The one-piece cup-cap glass-to-glass closure was chosen for the S8DRM-1 fuel elements this quarter. To verify the developmental results

obtained last fall, an additional group of closures were fabricated. The previous set of 13 closures had an average leak rate of $0.02 \text{ cm}^3/\text{hr}$ at 1400°F and 2 atm H_2 . The results of the current set of 14 are shown in Table XIII. The average is $0.03 \text{ cm}^3/\text{hr}$ under the same test conditions. Two S8ER blend samples were prepared at the same time; the leak rates were similar. However, the one-piece design is more consistent and easier to handle.

TABLE XIII
HYDROGEN PERMEATION THROUGH REFERENCE
DESIGN CLOSURES

S8ER Design Sample No.	Reference Design Sample No.	Permeation Rate, ϕ (cc/hr at 1400°F and 2 atm H_2)
	31	0.02
	32	0.00
	33	0.01
	34	0.10
	35	0.06
	36	0.11
	37	0.01
	38	0.00
	39	0.00
	40	0.00
	41	0.00
	42	0.00
	43	0.04
	44	0.09
45		0.01
46		0.18

Average for
31 to 44 = 0.03 cc/hr

Five closure samples from the previous group assembled were submitted to a cluster endurance test at 1500°F . The test had been underway 2000 hr when failure was detected. Each sample was then retested individually, and the rate was higher in all cases, one was off-scale ($>9 \text{ cc/hr}$). When these were cut open, spalling of the coating was apparent. It is suspected that spalling was caused by underfiring of the second (middle) coat as some of the early closure work on AI-8763D ceramic gave similar evidence.

Three closures of the current group have been placed on endurance test. The data in Table XIV indicates these closures are increasing in the same manner as the first group. This increasing leak rate is probably due to the AI-8763D coating.

TABLE XIV
ENDURANCE TESTS OF REFERENCE DESIGN CLOSURES

Blend Number	Previous Permeation Rate*	Time (hr)†	Permeation Rate [cc(STP)/hr]	Time (hr)	Permeation Rate [cc(STP)/hr]	Time (hr)	Permeation Rate [cc(STP)/hr]	Time (hr)	Permeation Rate [cc(STP)/hr]
31	0.02	48	0.06	119	0.09	211	0.68	305	1.01
32	0.00	4	0.12	127	0.13	221	0.18	331	0.20
33	0.01	4	0.06	30	0.07	213	0.31	305	0.43

*Element hydrogen permeation rate at a nominal 1400°F, cc(STP)/hr

†Total number of continual hours at 1400°F

b. Assembly Closure Welding

Qualification of the electron beam fuel element closure welding process is underway. A limited number of tests have been made to establish the final S8DRM-1 end-plug cladding-tube configuration. The 0.015 in.-weld lip could not bridge over the oxide film left by the shrink-fit operation. However, when the lip was increased to 0.035 in., there was a large enough pool of molten metal to bridge the oxide film. The modified final closure technique has thus far been used to produce eight good welds. Each was welded in one pass with good weld appearance and reproducibility.

Using the electron-beam final closure weld, nine S8DRM-1 prototype elements were assembled with the reference design end-closure and the collar button hold-down device. The results of subjecting these elements to the following series of tests are shown in Table XV.

- 1) X-ray for location of all parts,
- 2) Permeation test for leak rate,
- 3) X-ray for shift in position of parts,
- 4) Qualification test for ability to withstand shock and vibration,
- 5) X-ray for location of all parts, and
- 6) Permeation test for final leak rate.

Two of the nine elements were gross leakers as determined by the first permeation test. These elements were cut open and the closure end pressurized and tested separately. Both tests showed a very small percentage of the element leak to be in the closure. Loss of the end gap in some elements

TABLE XV

PERMEATION HISTORY OF NINE S8DRM-1 PROTOTYPE ELEMENTS

Element Number	Collar Button Fit-up As Assembled	Average Permeation Rate, ϕ_1 (cc/hr at 1400° F)	Collar Button Fit-up After Permeation Test	Collar Button Fit-up After Shock and Vibration	Permeation Rate, ϕ_2 (cc/hr at 1400° F)	Change of Permeation Rate (%)
E541N	5-mil gap, CB discs are parallel ($\sim 1^\circ$)	1.6	0 gap, 1° tilt to bottom disc	No change	1.8	+12
E546	5-mil gap, CB discs are parallel	0.27	0 gap, 3° tilt to bottom disc	No change	0.57	+100
E553N	0 gap, discs are parallel	1.9	0 gap, 1° tilt to bottom disc	Very slight bow in pin	2.3	+17
E554N	5-mil gap, discs are parallel	1.2	0 gap, 1° tilt to bottom disc	No change	1.1	-8
E558N	0 gap, $1/2^\circ$ tilt to bottom disc	0.44	0 gap, 3° tilt to bottom disc, slight	3-mil gap between top disc and fuel rod, slight bow in pin, slight movement of rod when element is shaken	0.59	+34
E592N	0 gap, discs are parallel	0.53	0 gap, 2° tilt to bottom disc, slight bow in pin	No change	0.69	+30
E594N	5-mil gap, 1° tilt to bottom disc	0.67	0 gap, $1/2^\circ$ tilt to bottom disc	Slight concavity to top disc, though it is still in contact with fuel	1.2	+79
E565N	5-mil gap, discs are parallel	9.0	0 gap, 1° tilt to bottom disc	Element decanned, blend tested alone, $\phi = 0.11$ at 1400° and 2 atm	-	-
E587N	0 gap, 1° tilt to bottom disc	9.0	0 gap, 2° tilt to bottom disc	Element decanned, blend tested alone, $\phi = 0.04$ at 1400° and 2 atm	-	-

during the first permeation test may be due to axial shrinkage of the swage-straightened tubing. It can be seen that the collar buttons withstood the shock and vibration testing. After the ramp heating tests, the cladding will be examined to determine the cause of any gross leak rate changes.

c. Nondestructive Tests for Closures

A high resolution ultrasonic testing technique has been shown to be a feasible method for measuring the amount of weld penetration in end closures. High frequency sound from a focused ultrasonic transducer is aimed at the weld zone parallel to the axis of the fuel rod. Complete nonfusion in the weld has been readily detected using the initial experimental mockup. The ability to measure varying degrees of weld penetration will be determined as soon as weld samples are prepared. To date, only electron beam welds have been inspected, since this welding technique does not melt and round the corner of the end cap. The ultrasonic sound beam will not enter the end cap properly if the corner is rounded as is the case during tungsten inert gas (TIG) welding.

The electrical resistance method has not been successful in efforts to detect variations in weld penetration. A series of electron-beam butt-welds in tubing with weld penetration from 30 to 100% were examined using a four-contact probe with an a-c milliohmmeter and a d-c resistance bridge. The four-contact probe contained two current and two potential contacts. No correlation between resistance readings and weld penetration has been demonstrated to date. The primary variation in resistance readings has been attributed to wall thickness changes.

Special radiographic techniques using correction forms are being designed for future evaluation to aid in detecting variations in weld penetration of end closures. The greatest problem to be overcome is the control of scattered radiation during tangential exposures.

4. Cladding

a. Corrosion and Cleaning Studies

The construction of a thermal convection NaK loop has been completed and the first complete fuel element is on test. The purposes of the loop are:

- 1) To measure permeation rate of the fuel elements in a NaK environment (this test is of primary interest because all qualification testing is being performed in a vacuum environment);

- 2) To determine if the hydrogen in the NaK changes the corrosion rate of NaK on Hastelloy-N; and
- 3) To determine if there is any significant difference in permeation rate, and between fuel elements that have had the chrome oxide removed and those that have not been precleaned.

The tests are being conducted near reactor thermal gradient conditions: inlet 1150°F, outlet 1400°F. The full series of tests including 1, 2, 4, and 8 days should be completed within the next few months.

b. Nondestructive Tests for Cladding

Two replicating materials were investigated in order to develop non-destructive means of measuring the depth and configuration of artificial defects machined into cladding tubes used for eddy current and ultrasonic testing standards. One of the replicating materials was a heat-curing flexible plastic while the other was a self-curing rigid plastic. A series of casts of machined notches was made using both materials. These casts were evaluated for both accuracy and reproducibility using a measuring microscope. On outside notches, it was found that both the flexible and the rigid materials were able to reproduce the dimensions of the notches within ± 0.0002 in. for notches ranging in depth from 0.001 to 0.006 in. Further tests disclosed that only the flexible plastic could be stripped from inside surface notches. It was concluded that both of these materials will greatly aid in the evaluation of the size of artificial defects and the qualification of eddy current and ultrasonic nondestructive testing standards.

Eddy current techniques for measuring the wall thickness of Hastelloy-N cladding tubes have been successfully employed in production inspection. Instrument sensitivities of ± 0.0002 in. have been achieved. This particular instrument is easier to use than ultrasonic instrumentation.

c. Physical Property Measurements

The application of SCB coating to Hastelloy-N cladding requires that the cladding be subjected to a temperature of 2100°F in air for about 45 min. Prior to coating, the Hastelloy-N is chromized at 1800°F for about 2 hr and then vacuum outgassed for 12 hr at this temperature. In order to be assured that no deleterious effects on mechanical properties of the cladding result from these heat treatments, a study of the short-term tensile properties of the Hastelloy-N after various stages of processing is underway.

Room temperature tensile tests were made on samples of Hastelloy-N in the following conditions: (1) as-received, (2) chromized and vacuum outgassed, and (3) chromized, outgassed, and coated on one side with three coats of SCB enamel. Table XVI shows the results of these tests. There were no significant adverse effects on short-term mechanical properties as a result of these processes. All of the strength values are well within the AI specifications for Hastelloy-N. The changes that do occur are probably due to grain growth in the material. Further studies are planned to determine the creep properties of Hastelloy-N as a result of the coating process.

TABLE XVI
TENSILE TEST RESULTS FOR HASTELLOY-N
(0.067-in.-thick sheet)

Extent of Processing	Ultimate Tensile Strength (psi)	0.2% Yield Strength (psi)	Reduction of Area (%)	Elongation in 2 in. (%)
As-received	117,600 ± 600*	46,100 ± 200	26.9 ± 0.6	33.1 ± 0.9
Chromized and out-gassed	122,700 ± 3000	47,100 ± 200	29.8 ± 1.9	38.7 ± 5.5
SCB-coated	118,400 ± 2500	43,100 ± 700	47.6 ± 2.6	57.8 ± 2.4

*All ± values are standard deviations of the average

5. Performance Tests

a. Development

Further data have been obtained which indicate that the developmental S8ER fuel elements containing the AI-8763D coating tend to deteriorate under thermal treatment. Last quarter, it was reported that sharp increases in permeation rate were found after 100- to 200-hr operation at 1400°F. This was demonstrated by data from thermal endurance and control permeation tests. Thermal endurance and thermal cycling tests were performed this quarter to obtain additional data on the effects of thermal treatment.

Four fuel elements were given multiple thermal cycles between room temperature and 1400°F at 50°F/min. Four other elements were maintained at 1400°F for several hundred hours without thermal cycling. (The elements used

were previously given vibration, shock and acceleration exposure, as reported last quarter.) The effects of these thermal treatments are shown by the permeation test data in Table XVII. The data show that permeation rates gradually increased with accumulated time at 1400°F. The data also show that thermal cycling plus time at 1400°F can cause large increases in permeation rate. None of the data from previous thermal tests indicated similar gross damage. Although mechanical inputs did not immediately affect the fuel elements, it is apparent from the test data that they have become more susceptible to damage by subsequent thermal treatment.

A summary of the data on 19 S8ER developmental fuel elements (normal uranium) indicated the following:

- 1) Thermal aging at 1400°F caused the hydrogen permeation from each element to increase. The increase in hydrogen permeation was due to defects in the ceramic coating, rather than general coating deterioration.
- 2) The increase in hydrogen permeation caused by thermal aging was accelerated and increased in magnitude by thermal cycling between room temperature and 1400°F.
- 3) Exposing the fuel elements to the qualification level of vibration, shock, and acceleration caused little or no immediate increase in hydrogen permeation.
- 4) Four elements previously exposed to the qualification level of vibration, shock, and acceleration were tested at 200% of the qualification level. One fuel element exhibited a severe increase in permeation after this test.
- 5) Thermal cycling of elements previously exposed to vibration, shock, and acceleration caused severe increases in hydrogen permeation rates. This effect of thermal cycling was found for elements exposed to either high or low levels of mechanical inputs.

b. Qualification

This program is being performed to determine if the fuel elements can withstand the qualification levels of mechanical and thermal inputs without

TABLE XVII

HYDROGEN PERMEATION OF S8ER DEVELOPMENTAL FUEL ELEMENTS

Cumulative Time at 1400°F (hr)	Cumulative Thermal Cycles	Permeation Rate, ϕ [cc(STP)/hr] at 1400°F				Cumulative Time at 1400°F (hr)	Cumulative Thermal Cycles	Permeation Rate, ϕ [cc(STP)/hr] at 1400°F			
		E-127-N*	E-131-N	E-163-N	E-264-N*			E-124-N	E-194-N	E-210-N	E-228-N
20	1	0.88	1.68	1.48	0.55	20	1	0.62	0.49	1.29	0.54
48	2†	1.26	1.53	0.73	0.35	48	2†	0.22	0.35	0.98	0.41
72	3	1.08	1.18	0.94	0.35	72	3**	0.20	3.18	0.84	0.24
97	4	2.27	2.10	0.98	0.56	95	3**	0.41	3.57	0.84	0.29
134	-	>6.00	-	-	-	122	3**	0.55	3.77	0.99	0.45
180	5	-	3.98	1.64	0.55	192	3**	0.81	4.99	1.34	0.76
200	-	-	>6.00	2.24	0.60	214	3**	0.91	5.05	1.46	0.84
220	6	-	-	-	1.41	244	3**	-	5.05	1.63	0.94
243	6	-	-	3.33	2.56	294	4	-	>6.00	2.32	-
316	6	-	-	-	5.43	364	-	-	-	-	1.12
322	6	-	-	-	>6.00	413	-	-	-	-	1.64††
361	6	-	-	4.14	-	-	-	-	-	-	-
410	6	-	-	4.65	-	-	-	-	-	-	-
430	6	-	-	4.65	-	-	-	-	-	-	-
500	6	-	-	3.73	-	-	-	-	-	-	-
558	6	-	-	3.78	-	-	-	-	-	-	-

*Received two additional thermal cycles and 4 hr at 1400°F during Handing Test.

†Permeation test after 100% of qualification mechanical inputs.

**Permeation test after 200% of qualification mechanical inputs.

††Permeation rate after thermal cycle 4.

damage to the ceramic. Irradiation effects are not included in the qualification inputs. However, the data obtained from this test program can be used in conjunction with irradiation data to determine the reliability of fuel elements.

The qualification inputs have been designed to simulate the various reactor environments (S8ER, S8DS, S8FS, and S8FR), and are as follows:

- 1) Thermal Soaks: Several slow heating and cooling cycles at less than 300°F/hr with a 10-hr dwell at 1300°F — this input simulates a prelaunch system checkout.
- 2) Launch Vibration, Shock, and Acceleration (omitted in the S8ER and S8DS Programs): Simulates reactor launch conditions.
- 3) Ramp Heat: Numerous fast heating (300°F/min to 850°F, then 150°F/min to 1400°F), and cooling (50°F/min) cycles — simulates reactor startup conditions.
- 4) Thermal Cycle: 300 cycles about 1400°F (1350°F to 1450°F) at 50°F/min — simulates reactor operation.
- 5) Thermal Endurance: 10,000-hr operation at 1400°F — simulates reactor operation.

In addition, the elements receive a hydrogen permeation test at 1400°F after completion of each input 1 through 5. Also, permeation is monitored continually while the elements are under thermal endurance. The elements are under vacuum ($\sim 10^{-5}$ mm Hg) during each of the above inputs with the exception of vibration, shock, and acceleration.

The status of the equipment which is used for performance testing the SNAP 8 fuel elements is indicated below:

- 1) Thermal Soak: This system is operational and has been shown to give reproducible inputs.
- 2) Launch, Vibration, Shock, and Acceleration: This system is operational, but the present test vessel can only hold 19 elements during testing. A simulated core tank, which can hold 218 elements, is being fabricated.
- 3) Ramp Heat: One system is operational and two additional systems are being installed.

TABLE XVIII

S8ER FUEL ELEMENT QUALIFICATION TEST RESULTS

Element Number	N_H^*	Carbon [†] Additive (wt %)	Acceptance $\phi_{1400}^{\S}$	Pre-Thermal Soak ϕ_{1400}	Post-Thermal Soak ϕ_{1400}	Post-Ramp Heat ϕ_{1400}	Post-Thermal Cycle ϕ_{1400}	Endurance Test ϕ_{1400}
N-760	6.01	0.18	0.05	0.13	0.94	1.24	-	-
N-804	6.20	0.18	0.18	0.20	0.15	0.64	-	-
N-811	6.11	0.35	0.58	0.56	1.15	2.40	-	-
N-841	6.10	0.17	0.09	0.11	0.13	0.68	-	-
N-886	6.08	0.36	0.25	0.27	0.72	-	-	-
N-897	6.15	0.18	0.30	0.28	0.38	0.41	-	-
N-802	5.98	0.35	0.46	0.47	0.68	1.61	-	-
N-810	6.13	0.18	0.27	0.24	1.01	1.58	-	-
N-821	6.20	0.18	0.11	0.07	0.28	0.75	-	-
N-880	6.15	0.18	0.07	0.05	0.07	0.68	-	-
N-881	5.97	0.38	0.31	0.57	1.04	2.37	-	-

* N_H = Number of hydrogen atoms $\times 10^{23}$ /cc of fuel.

[†]Refers to the weight percent of carbon additive in the fuel.

[§] ϕ_{1400} = Fuel element hydrogen permeation, cc(STP)/hr at a nominal 1400°F.

- 4) Thermal Cycle: The systems will be the same as those used for ramp heating.
- 5) Thermal Endurance: One system is operational and another is being installed.
- 6) Permeation Testing: A nine-furnace system is operational and a sixteen-furnace system is being installed.

All of the elements presently being tested are basically of the S8ER design and contain the AI-8763D coating. Some of the elements are being subjected to advanced qualification tests (S8DS and S8FS) to establish the level of development toward meeting the overall performance goals. The elements have been divided between the tests as shown below:

- 1) Eleven elements in S8ER qualification testing.
- 2) Twelve elements in S8DS qualification testing.
- 3) Four elements in S8FS, S8FR-1 and -2 qualification testing.

Four elements (undergoing S8FR qualification inputs) were modified to include increased axial and radial gaps between the fuel rod and the ceramic coating to simulate initial space required for subsequent fuel swelling. This clearance, at room temperature, amounts to a loose fit between the fuel rod and coating.

Of the 33 S8ER design elements to be qualification tested, 27 have been received. The progress of these elements in their respective programs is shown in Tables XVIII, XIX, and XX. Each table lists N_H (number of hydrogen atoms $\times 10^{-22}$ /cc of fuel), and the weight percent of the carbon additive in the fuel. This data is listed because both N_H and carbon content affect the dissociation pressure of the hydrogen in the fuel and hence the permeation rate.

All of the S8ER design elements to be tested with S8ER inputs (cf. Table XVIII), have been received and will shortly enter the thermal endurance phase of the program. The majority of these elements tested to date show significantly increased permeation rates as a result of the preliminary treatments.

All of the S8ER design elements to be tested with S8DS inputs (cf. Table XIX) have been received and are undergoing test. Of the elements received under this program, half have been liquid-honed (a high velocity

TABLE XIX
S8DS FUEL ELEMENT QUALIFICATION TEST RESULTS

Element Number	N_H	Additive (wt %)	Acceptance ϕ_{1400}		Pre Thermal Soak ϕ_{1400}	Post Thermal Soak ϕ_{1400}	Post Ramp Heat ϕ_{1400}	Post Thermal Cycle ϕ_{1400}	Post 2nd Ramp Heat ϕ_{1400}	Endurance Test, ϕ_{1400}
			Pre Clean	Post Clean						
N-798	5.96	0.37	0.27	0.34	0.33	1.21	-	-	-	-
N-814	6.11	0.16	0.08	0.13	0.20	0.26	-	-	-	-
N-858	6.18	0.18	*	0.17	0.16	0.54	-	-	-	-
N-870	6.11	0.34	*	0.25	0.22	0.62	-	-	-	-
N-887	6.13	0.18	*	0.10	0.12	-	-	-	-	-
N-918	6.08	0.35	*	0.89	0.68	-	-	-	-	-
N-951	6.04	0.36	*	1.42	0.94	-	-	-	-	-
N-962	6.08	0.18	*	0.09	0.71	-	-	-	-	-
N-939	6.18	0.18	0.17	0.33	1.70	-	-	-	-	-
N-960	6.10	0.18	0.14	0.15	0.08	-	-	-	-	-
N-916	6.12	0.28	0.34	0.50	-	-	-	-	-	-
N-937	6.15	0.37	0.17	0.10	-	-	-	-	-	-

* ϕ_{1400} = Fuel element hydrogen permeation, cc(STP) 1 hr at a nominal 1400°F

TABLE XX
S8FS-FR-1 AND-2 FUEL ELEMENT QUALIFICATION
TEST RESULTS

Element Number	N _H	Additive (wt %)	Acceptance ϕ 1400*		Pre V, S, and A ϕ 1400 [†]	Post V, S, and A ϕ 1400	Post Thermal Soak ϕ 1400	Post Ramp Heat ϕ 1400	Post Thermal Cycle ϕ 1400	Post 2nd Ramp Heat ϕ 1400	Endurance Test ϕ 1400
			Preclean	Postclean							
N-861	6.20	0.21	0.17	0.42	0.54	0.75	0.98	-	-	-	-
N-884	6.23	0.21	0.34	1.13	1.04	1.14	-	-	-	-	-
N-972	6.20	0.21	0.14	0.32	0.81	-	-	-	-	-	-
N-963	6.11	0.18	0.13	0.23	0.28	-	-	-	-	-	-

* ϕ 1400 = Fuel element hydrogen permeation, cc(STP) 1 hr at a nominal 1400°F

[†]Vibration, shock, and acceleration

abrasive cleaning process to remove the oxide layer present on the element cladding; the purpose of the cleaning is to reduce initial NaK contamination). The liquid-honed elements were permeation tested at 1400°F before and after cleaning to show any changes in permeation rate. Permeation testing to date of these elements also showed that the permeation rates are increasing.

Four of the ten S8ER design elements to be tested for S8FS-FR-1 and -2 qualification (cf. Table XX), have been received and are in the early stages of the program. To date, these elements showed increases in permeation rate similar to those of the other elements.

A typical temperature profile for an element undergoing thermal soak (one cycle) is shown in Figure 29, along with the required heating and cooling rates. All subsequent cycles of this element (and subsequent elements) had identical profiles.

Figures 30 and 31 show typical temperature profiles (S8ER Qualification Program) for the ramp heating input. The system is first checked out for the desired heating and cooling rates by use of a reject element with a series of six thermocouples along its length (Figure 30). The qualification elements have thermocouples spot welded only at the ends. Spot welding thermocouples to the solid metal end caps minimizes the possibility of ceramic damage. A typical ramp heat profile for an S8ER qualification element is shown in Figure 31. During ramp heating the maximum temperature attained and the maximum axial temperature differential for a series of these elements is shown in Table XXI. The axial temperature differential of the elements ramp-heated to date vary from element to element. This variation is not a fault of the system itself, but

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Figure 29. Fuel Element Temperature During Thermal Soak

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Figure 30. S8ER Fuel Element Temperature Profile During Ramp Heat System Checkout

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Figure 31. S8ER Qualification Test Fuel Element Temperature During Ramp Heating

TABLE XXI
RAMP HEATING DATA

Element	Maximum Temperature Recorded (°F)	200°F	400°F	600°F	800°F	1000°F	1200°F	1400°F
E-894-N Profile (Element)	1415	25	50	20	10	20	20	30
841	1410	1	35	25	5	35	30	45
886	1405	5	30	20	5	25	30	50
804	1400	1	10	10	2	20	25	50
760	1405	1	15	10	3	5	10	10
811	1415	5	40	35	20	20	25	50
897	1400	1	35	25	5	25	25	30

appears to be an unavoidable variation between elements. This temperature differential is believed to be a result of differences in the radial clearance between the fuel and its cladding.

The data presented in Table XVIII show the fuel elements will probably meet the requirements for S8ER of ~2000 hr of operation. As expected, the data, including Tables XIX and XX, show that the fuel elements of the S8ER design are not capable of meeting the ultimate SNAP 8 fuel element performance goals. Significantly increased permeation rates were found as noted below:

- 1) Of 24 elements which were tested upon receipt (pre-thermal soak), 3 showed significantly higher permeation.
- 2) Of 16 elements which were tested after thermal soaking, 12 showed significantly higher permeation.
- 3) Of 4 elements which were tested after ramp heating, 4 showed significantly higher permeation.
- 4) Of 2 elements which were tested after vibration, shock, and acceleration, 1 showed significantly higher permeation.

Considering the "loose-fit" assembly of the elements which were vibrated, shocked, and accelerated, it is surprising that this input did not completely destroy the hydrogen barrier of these two elements.

Although there have been significant increases in the permeation rates of the elements caused by one or more of the inputs, only three are close to "failure," E-881-N, E-811-N, and E-939-N. Reactor design calls for an average element permeation of ≤ 1.60 cc(STP)/hr.

c. Environment

Six normal uranium fuel elements were received for environmental testing. The purpose of the environmental test program is to establish the reliability of SNAP 8 fuel elements as a function of the severity of the non-nuclear environment, finally yielding the mode of failure and ultimate capability of the elements, thus establishing the safety factor over the qualification level. Emphasis is on the effect of increased inputs of vibration, shock, and acceleration; and of increased rates of ramp heating. The alloy composition and acceptance test leak rates of elements received for this program are shown in Table XXII.

TABLE XXII
ENVIRONMENTAL TEST RESULTS

Element Number	Carbon (wt %)	Hydrogen (wt %)	N _H	Uranium (wt %)	Cleaned	Permeation Rate, ϕ Acceptance Test [cc (STP)/hr at 1400°F]
E-904-N	0.36	1.67	6.08	10.17	No	0.07
940	0.35	1.66	6.04	10.17	No	0.17
947	0.18	1.69	6.13	10.18	No	1.12
976	0.33	1.67	6.08	10.16	Yes	0.20
993	0.36	1.68	6.08	10.09	No	0.38
996	0.40	1.65	6.01	10.06	Yes	0.29

d. Performance Evaluation

Efforts were initiated to conduct a complete theoretical study of the performance capabilities and limitations of the SNAP 8 fuel element containing the AI-8763D coating. In this respect, the permeation test data for the AI-8763D five-membrane cluster experiment were evaluated. Utilizing a Langmuir adsorption correlation technique, the permeation data for temperatures of 1200, 1300, and 1400°F were found to be very adequately correlated by the following equation:

$$\phi = \frac{13.46 P T^{5/2} e^{-21020/T}}{T^{5/2} + 3.881 \times 10^9 P e^{-10340/T}} \quad , \quad \dots (1)$$

where

ϕ = membrane hydrogen permeation, cc(STP)/hr/cm²;

P = hydrogen pressure, atm; and

T = membrane temperature, °R.

(Note: In the above equation, 13.46 is defined to be the "aging constant." Inasmuch as the cluster data for which the equation was derived were obtained after many hours at elevated temperature, the reader is cautioned concerning use of this constant for unaged membranes.)

It is disappointing to report that the above equation does not correlate the cluster data obtained at 1450 and 1500°F. The test data denote an unexplained jump in permeation rate with a change in temperature from 1400 to 1450°F. Thus, use of the above equation for a cladding temperature in excess of 1400°F gives computed rates significantly lower than those experimentally observed in this higher temperature region.

Isothermal ($\leq 1400^\circ\text{F}$) permeation for SNAP 8 fuel elements coated with AI-8763D ceramic can thus be expressed by the following equations:

$$\phi_E = \phi_{Bl + D} + \phi_{Cl} \quad , \quad \dots (2)$$

$$\phi_E = K_{Bl + D} \sqrt{P} e^{-12760/T} + K_{aging} \frac{P T^{5/2} e^{-21020/T}}{T^{5/2} + 3.881 \times 10^9 P e^{-10340/T}} \quad , \quad \dots (3)$$

where

ϕ_E = total fuel element permeation, cc(STP)/hr;

$\phi_{Bl + D}$ = combined total blend and defect permeation, cc(STP)/hr;

ϕ_{Cl} = total permeation through defect-free AI-8763D ceramic coated cladding, cc(STP)/hr;

$K_{Bl + D}$ = combined blend and defect constant;

K_{aging} = total aging constant (i. e., K_{aging} in Equation 3 is for the total ceramic coated cladding area);

P = hydrogen dissociation pressure, atm; and

T = cladding and fuel temperature, °R.

The hydrogen dissociation pressure is obtained from:

$$P = \exp (-8.8455 + 88.9901 x - 78.8961 x^2 + 21.3731 x^3) \cdot \left[\exp -\frac{10^4}{T} \left(12.9720 - 9.7707 x + 2.4984 x^2 \right) \right], \quad \dots (4)$$

where x = the "effective" (H/Zr) ratio obtained using the relationship

$$x = \frac{108,600 H}{(100 - H) (1200 - 103.22 C - 12 U)} \quad \dots (5)$$

In Equation 5,

H = wt % hydrogen in the fuel,

C = wt % carbon in the fuel alloy prior to hydriding, and

U = wt % uranium in the fuel alloy prior to hydriding.

Let us consider the implication of Equation 3 with the associated Equations 4 and 5. $K_{\text{Bl} + \text{D}}$ is actually a "quality constant" which can be different for each ceramic-coated clad assembly. For a "perfect" assembly, $K_{\text{Bl} + \text{D}} = 0$, and the sole permeation is through defect-free ceramic coated cladding. Computation of $K_{\text{Bl} + \text{D}}$ for an element is easily accomplished given ϕ_E , T , H , C , U , and K_{aging} . A Fortran computer program was written and successfully checked out for calculation of $K_{\text{Bl} + \text{D}}$ values for each S8ER fuel element. An evaluation study was initiated to determine the proper value of K_{aging} to use in these calculations.

6. Irradiation Tests

a. Fission Gas Mobility Studies

These studies are gaining information on the mechanism of radiation swelling in the SNAP fuel alloy by direct observation of the mobility of the fission gases with changes of temperature and hydrogen concentration. It is

almost universally assumed that motion of the fission gas allows it to coagulate in pockets. The gas can then exert pressure and result in volume instability (swelling). These studies will afford initial screening against fuel compositions of swelling characteristics unsuitable for SNAP 8 use.

In this study, small thin sections of zirconium hydride fuel of various hydrogen and carbon levels are being subjected to parts-per-million levels of fuel burnup at low temperatures in the AI Shield Test Reactor. Following irradiation, the samples are being annealed at progressively higher temperatures or under other circumstances that may promote the release of Xenon 133 (5.3-day half-life). Single channel pulse height analyzers provide adequate resolution of the emitted gamma rays from this isotope, and a 100-channel analyzer permits its identification.

Low gas-release rates can be associated with internal trapping of the generated rate gas atoms. High release rates could be associated with a myriad of fuel material parameters. The most obvious variables are temperature, hydrogen content, carbon content, grain size, particle size distribution, contamination level (oxygen, nitrogen), and phase changes. With these variables, the amount of testing required to demonstrate stability by full-scale testing is prohibitively time consuming and expensive. The low level — annealing procedures provide a powerful method of indicating instability with relatively small effort, time and expense.

The results of 30 to 40 annealing tests are summarized as follows:

- 1) Dehydriding from the δ phase to the two phase $\delta + \beta$ region produces copious gas release and should be associated with fuel element instability. Questions exist concerning the importance of the rate of dehydriding.
- 2) Samples that release little fission gas on extended heating at 950°C can be made to release fission gases on dehydriding at 860°C.
- 3) Dehydriding fuel material need not cause fission gas release unless a phase transformation results from the dehydriding operation. No fission gas release occurs until a phase boundary is crossed; namely, $\delta - \beta$.

- 4) Quenching dehydrided materials from the two-phase region, $\delta - \beta$, causes fission gas release. The rate of quenching (or cooling) is currently suspected to be important. More experiments are needed in this area to provide definitive conclusions.
- 5) Phase transitions, per se, are not sources of fission gas release. Hydriding below the eutectoid temperature from $H/Zr = 0$ to fully δ -phase material, $H/Zr = 1.6$, does not cause gas release. Conversely, hydriding above the eutectoid temperature appears to promote gas release at the composition where the new phase appears. Delineation of phase boundaries is not sufficiently precise to associate gas release with the exact crossing of a phase region.
- 6) Unmodified materials (no carbon additive) generally appear to be inferior to carbon-modified materials in the present studies. The difference in behavior may be attributed to differences in grain size as contrasted to differences in additive level or the grain shape, but conclusive data has not been obtained.
- 7) Rate effects appear to be important but the significance is masked by other potential variables. Further studies are again needed. Slow cooling and quenching tests have provided markedly different test results. No interpretation of the differences is evident. Similarly, slow and rapid dehydriding may yield different release rates.
- 8) Depending upon the test sequence of heating, quenching, and venting of hydrogen, the release of nongaseous fission products has been observed, but no direct correlation with test parameters is yet evident.

b. Materials Irradiations

(1) NAA-68 AI-8763D Coating

The NAA-68-1 irradiation capsule was inserted in the ETR during midcycle shutdown on March 13, 1963. The experiment will determine the

irradiation stability of the AI-8763D hydrogen barrier enamel by comparison of re- and post-irradiation permeation measurements. The capsule contains two AI-8763D-coated Hastelloy-N cladding membranes. The upper membrane has been operating at a maximum cladding temperature of about 1200°F, while the lower membrane is just under 1000°F. The capsule design was based on a flux of 5 watts/gm gamma heating to provide most of the heat to the membranes. It appears that the capsule is receiving only 3 to 4 watts/gm of gamma heating. The electrical heaters are not able to compensate for this flux difference.

The calculated fast flux exposure for the first 1-1/2 cycles is 3.11×10^{20} nvt. The capsule is scheduled for discharge in September 1963 with a total neutron dosage of 1.4×10^{21} nvt.

(2) NAA-77 Coating

An irradiation capsule to study the inpile hydrogen permeation of barrier coatings is nearing completion. The first capsule in this series, NAA-77-1, will contain a membrane of SCB coating on Hastelloy-N. Hydrogen flow through the coating will be monitored continuously during inpile operation by a mass spectrometer. The second capsule in the series, NAA-77-2, will contain a fission plate to expose the coating to fission fragment bombardment, similar to that expected with fuel in the cladding. Subsequent capsules in the series will also contain a fission plate in conjunction with other coatings and barrier materials.

Figure 32 shows the approved capsule design for NAA-77. The hydrogen permeation measurement system and the temperature control system are in the advanced stages of completion. A total of eleven (11) SCB-coated membranes has been fabricated to obtain a sample for this capsule. In addition to providing a good SCB coating on the sample, the total indicated reading (TIR) had to be <0.015 in. over the 16-in. length. Table XXIII gives the permeation values and TIR for the samples. All coatings are in the noncrystallized condition.

Effort this quarter was also devoted to the design, procurement of components, and construction of the hydrogen permeation system. All components used in the system have been received and are in the assembly stage. Installation drawings have been prepared for the external piping and electrical wiring required to install the instrumentation at the ETR. These drawings have been forwarded to the reactor test site for installation.

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Figure 32. Inpile Permeation Test Capsule NAA-77

TABLE XXIII
PERMEATION RATES OF SCB MEMBRANES, NAA-77 EXPERIMENT

Sample Number	Permeation Rate, [cc(STP)/hr/cm ² at 1500°F and 1 atm H ₂]	Total Indicated Reading (in.)	Order of Selection of Membranes for Capsule
E 807	1.1 x 10 ⁻³	0.009	1
E 865	3.3 x 10 ⁻³	-	-
RD 27	1.7 x 10 ⁻³	0.010	3
RD 30	1.7 x 10 ⁻³	0.019	-
RD 29	2.00 x 10 ⁻³	0.004	-
RD 22	1.8 x 10 ⁻³	0.024	-
E 876	1.7 x 10 ⁻³	0.010	2
E 853	1.1 x 10 ⁻³	0.042	-
E 799	1.9 x 10 ⁻³	0.019	-
E 882	1.2 x 10 ⁻³	0.006	-
E 805	0.8 x 10 ⁻³	0.021	-

Work has been started on the fission plate for the next capsule in the series. Analytical calculations have been completed and studies are being made on methods of fabrication of the plate. It appears that a ceramic rod containing a coating of enriched UO₂ will be placed in the center of the cladding tube. SCB coating is being considered as a binder to hold the UO₂ on the ceramic rod.

(3) NAA-116 Low Temperature Materials

The low temperature materials irradiation experiment (NAA-116) is designed to investigate the effect of structural and compositional parameters on fission gas mobility, as determined in postirradiation annealing experiments. The design proceeded through the preliminary stage to the completion of final design during this quarter. As originally planned, 20 segmented specimens, of nine distinct compositions, will be irradiated in two capsules with different degrees of irradiation. The burnup rate will be below 1.5 met. at. %/yr and the temperature will be below 350°C (660°F) throughout the specimens. Uranium contents will range from 5 to 20 wt %; carbon contents will range between 0.02 and 0.4 wt %, and the H/Zr ratio will range between 1.3 and 1.85. Total burnup

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Figure 33. Specimen Assembly, HAPO
Low Temperature

will range between 0.1 and 0.4 met. at.%. The detailed list of specimen compositions and burnup was supplied in the last quarterly report.

Advance melting and extruding of the fuel specimens was started this quarter. Two billets containing 20 wt % U and 0.1 wt % carbon were completed.

The capsule design will permit the required water flow in the HAPO process channel to maintain minimum sample temperature. A shroud tube will be designed to adjust the flux on the individual specimens. Physics experiments are presently underway to determine reactor flux conditions and their influence on the specimens.

To ascertain that the temperature has not exceeded the desired limits, two dummy specimens will each contain three melt wires. Figure 33 shows the clad fuel specimens and the melt-wire arrangement. The melt wires are lead, zinc, and 2017 aluminum alloy, to detect temperatures of 327°C (620°F), 420°C (788°F), and 540°C (1004°F). It is not anticipated that any of the specimens will reach these temperatures. As further precaution, the specimens of 20 wt % uranium are to be fitted to closer tolerance in their stainless-steel cladding tubes.

Tests have been made of the melt wires to determine their appearance after melting and their compatibility with the fuel. Studies indicated that if flattened wires are used, they will assume a round shape on melting (see Figure 34). This will afford a simple visual indication in the hot cell. Since it was found that the wires would not slide out of holes in the sample easily after heating, the melt-wire specimens will be made in two halves for ease of the wire removal after irradiation. Heating the wires in contact with zirconium for two hours at 670°C (1238°F) showed no gross compatibility problems.

Postirradiation examinations will involve step annealing followed by density and dimensional measurements. Optical and electron microscope examinations, as well as tapping for fission gas, gamma-ray scanning, and burnup analysis are planned.

Figure 34. Dummy Fuel Specimen Melt Wires

(4) High Temperature Materials

A high temperature fuel materials irradiation is now going through conceptual design. This series of experiments will study the effects of temperature, burnup, burnup rate, and composition on fuel integrity. These studies will determine the ultimate capability of the fuel alloy, indicate the effects of deviations of these parameters on present fuels, and give additional scientific information on the mechanisms of swelling. The design of specimens and assemblies will resemble that of the NAA-115 experiment, as the problems of heat transfer and containment of hydrogen are very similar in the two programs.

(5) Annealing Studies

Postirradiation annealing experiments have been performed on fuel samples from NAA-53-1 and NAA-53-2. The irradiated uranium-zirconium hydrided fuel was annealed at temperatures from 1128 to 1650°F. Hydrogen pressure within the chamber containing the fuel was varied from 15 to 115 psia to correspond to the fuel hydrogen dissociation pressure at the annealing temperature. Figures 35 and 36 give weight, volume, and density measurements as a function of the individual heat treatments. While there are inconsistencies

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Figure 35. Weight, Volume, and Density Data, NAA-53-1B

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Figure 36. Weight, Volume, and Density Data, NAA-53-2A

in these data, gross swelling or catastrophic failure of this irradiated fuel did not take place. The data for NAA-53-1 (0.6 met. at.% burnup) shows an approximate 11% decrease in density. The fluctuation in weight measurements may be due to oxidation of the fuel plugs and spalling of same, and/or conceivable loss or gain of hydrogen. The data from NAA-53-2 (1.2 met. at. % burnup) shows a greater scatter of the data. The density decreased about 19%.

Modifications now underway are intended to remove some of the sources of doubt concerning these results. These include improved pumping capability to reduce oxidation at the surface, and provision for adjusting the hydrogen pressure while the specimen is at the annealing temperature to maintain constant hydrogen content. Postirradiation annealing of fuel from the NAA-53-3 irradiation will be undertaken when the fuel is available and these modifications have been checked out.

c: Engineering Irradiations

(1) NAA-53-3 Carbon-Modified Fuel Capsule

The NAA-53-3 capsule was irradiated in the A-30-SE facility of the MTR during cycles 187, 188, 189, and 190. After receiving an estimated burnup of 0.43 met. at. %, it was removed on May 6, 1963. AEC approval for shipment to the AI Component Development Hot Cell (CDHC) has been received and a plan of action for hot-cell examination has been prepared.

(2) NAA-115 and NAA-117 Fuel Elements

This area of effort is directed at irradiating SNAP 8 fuel elements which are prototype except for length and range of fuel compositions. The test elements are 4 in. long and will have a range of effective H/Zr ratios and wt. % carbon additive levels. The NAA-115 irradiations are to be performed at design burnup rate at central fuel temperatures of 1600, 1550, 1500, 1400, and 1300°F. The NAA-117 irradiations are to be performed at twice burnup rate at central fuel temperatures of 1300, 1400, and 1500°F.

The main effort this quarter was devoted to the following three areas: capsule design, physics experiments, and fuel element fabrication.

A description of the basic design of NAA-115-1 was presented last quarter. This quarter's work consisted of completing the detail prints of all components in the capsule, procuring materials, and starting the machining

of the components. The one major problem area encountered in the capsule was the design of a bellows to compensate for the different NaK volumes resulting from the range of temperatures covered. A unique system which has a built-in safety feature has been designed. Briefly, a 1.60-in.-diam 304 stainless-steel bellows compartment is located at one end of each of the six subcapsules in the NAA-115-1 capsule. A 347 stainless-steel bellows — 4.0 in. free length with 1.5-in. OD, and 16 convolutions/in. — is welded internally to the end of the compartment with a 0.015-in. clearance between the OD of the bellows and the ID of the compartment. Maximum calculated NaK volume compensation is 2.0 in.³ requiring a bellows travel of 2.1 in. A 1.7-in.-long filler plug is inserted in the bellows to take up the excess internal bellows volume resulting at compression of the bellows. Stress analysis of the bellows shows it to be marginally acceptable for prolonged use. For this reason a secondary compensation has been built into the design. A slotted diaphragm separates the bellows compartment from the shell housing the fuel elements. If the bellows were to fail, the NaK level in the bellows would drop as the NaK entered the bellows leaving a void at the top of the bellows compartment. The net effect is a void volume sufficient in itself to compensate for the NaK expansion. Thus, the only loss is the simulated systems pressure supplied by the bellows. A sketch of the bellows housing is shown in Figure 37.

Reduction and interpretation of physics experiment data is continuing. Integration of this data to establish parameters for the SN cell code (programmed at AI for use on the 7090 computer) was started this quarter.

Six of the eight physics experiment assemblies were irradiated in the Snout Facility of the Hanford K-West Reactor. Three of the 321 stainless-steel shroud tube assemblies — 0.50-in., 0.30-in., and 0.10-in. thickness — were included in the irradiations. These assemblies did not contain any fuel element mockups. The $\text{Cu}^{63}(\text{n},\gamma)\text{Cu}^{64}$ and $\text{Co}^{59}(\text{n},\gamma)\text{Co}^{60}$ reactions were used to establish flux profiles in and around the assemblies. Although the shroud tubes depressed the flux, it was found that peaking occurred inside the tube from moderation of the neutrons by the water. The 4.25-in. equivalent radius graphite cell was used as a model in the SN computer code. The results between predictions and measurements are shown in Figure 38. Agreement is generally found in the immediate vicinity of the shroud tube but not in the center of the tube. This is a result of the lower average neutron energy as the water

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Figure 37. Fuel Capsule Bellows Housing

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Figure 38. Flux Profiles For Several Flux Depressor Tubes

moderator thermalizes the neutrons. Effort is now underway to vary the effective cross section of the absorber in the SN code to compensate for the different neutron energies.

Twenty-one of the 31 fuel elements (three spares) have qualified for use in the NAA-115-1 and NAA-117-1 experiments. Two instrumented and three uninstrumented fuel elements failed to pass the permeation rate test and are being rebuilt at the present time in an effort to lower their high hydrogen loss rates. Selected data for the NAA-115-1 and NAA-117-1 specimens are shown in Tables XXIV and XXV, respectively.

C. FUEL ELEMENT FABRICATION

1. Production Status

a. Performance Test Elements

Forty-three of the required 55 performance test elements were delivered during the quarter. The average hydrogen leak rate of 0.43 cc(STP)/hr/element at 1400°F is well within the maximum specified value of 1.60 cc(STP)/hr/element.

These elements were of the same design as those delivered for the S8ER except for two modifications on a fraction of the elements. The radial and axial gaps were increased from 0.0015 to 0.003 in., and 0.20 to 0.50 in., respectively, on 10 elements. This change is required on all future SNAP 8 fuel elements to allow space for fuel swelling during irradiation. It was initiated on a group of 10 elements of this design to gain advance information on the fabricability and performance of elements containing fuel which is free to move around inside the cladding.

The second change was the adaption of liquid honing to remove the chrome oxide from the outside surface of half of the fuel elements. The chrome oxide layer is a result of the coating firing operations. It is desirable to remove this oxide to minimize the NaK coolant contamination during the system checkout. This process change was initiated this quarter to determine the effects on overall yield and subsequent performance of the elements.

TABLE XXIV

NAA-115-1 FUEL ELEMENT DATA

Fuel Element Number	Fuel Rod Data					Fuel Element Data				
	Chemical Analysis			Dimensions		Sm ₂ O ₃ (10 ⁻³ gm/in. tube)	Dimensions		Permeation Rate, ϕ *	
	U (wt %)	Hydrogen-2 Wt gain/chem. (wt %)	Carbon (wt %)							
				Diameter (in.)	Length (in.)		Diameter (in.)	Length (in.)	1200°F (cc/hr)	1400°F (cc/hr)
IS-1	9.88	1.64/-	0.08	0.5274	3.386	3.2	-	-	0.70	9.70 [†]
IS-1A	9.88	1.64/-	0.08	0.5275	3.386	3.3	-	-	0.24	5.20 [†]
IS-2	10.03	1.55/1.60	0.12	0.5275	3.385	-	0.563	3.998	0.03	0.33
IS-3	10.00	1.68/1.78	0.06	0.5273	3.385	3.5	0.564	3.997	0.01	0.05
IS-4	9.90	1.61/1.61	0.40	0.5274	3.386	3.8	0.572	3.998	0.02	0.24
IS-5	10.10	1.70/1.74	0.05	0.5272	3.385	3.2	0.564	4.000	0.04	0.36
IS-6	9.99	1.64/1.66	0.11	0.5336	3.388	3.2	0.563	3.997	0.04	0.45
IS-7	10.06	1.78/-	0.11	0.5273	3.386	-	0.565	3.997	0.05	0.50
IS-8	10.06	1.76/-	-	0.5338	3.386	3.2	0.564	3.998	0.17	2.14 [§]
IS-9	9.88	1.64/-	0.08	0.5274	3.386	3.0	0.571	4.000	0.13	1.35
IS-10	10.03	1.64/1.65	0.12	0.5275	3.471	3.6	0.563	3.997	0.00	0.06
IS-11	10.03	1.62/1.65	0.12	0.5273	3.470	3.3	0.563	3.997	0.02	0.12
IS-12	10.03	1.64/1.78	0.12	0.5276	3.469	3.8	0.563	3.998	0.01	0.09

*Acceptance limits: ϕ , 1200°F—0.19 cc/hr; 1400°F—1.60 cc/hr.

†These elements, containing thermocouples, are in the process of being rebuilt in an effort to lower these unacceptable permeation rates.

§Rejected and being rebuilt.

TABLE XXV

NAA 117-1 FUEL ELEMENT DATA

Fuel Element Number	Fuel Rod Data					Fuel Element Data				
	Chemical Analysis			Dimensions		Sm ₂ O ₃ (10 ⁻³ gms/in. Tube)	Dimensions		Permeation Rate, ϕ [*]	
	Uranium (wt %)	Hydrogen 2, Wt. gain/chem. (wt %)	Carbon (wt %)	Diameter (in.)	Length (in.)		Diameter (in.)	Length (in.)	1200 °F (cc/hr)	1400 °F (cc/hr)
M-1	9.99	1.64/1.66	0.11	0.5336	3.388	3.1	0.563	3.998	0.01	0.05
M-2	9.99	1.64/1.70	0.11	0.5336	3.387	3.6	0.563	3.997	0.01	0.07
M-3	10.15	1.66/-	0.06	0.5336	3.337	3.4	0.562	3.996	0.02	0.13
M-5	10.10	1.51/-	0.05	0.5339	3.387	3.4	0.564	3.996	0.00	0.03
M-6	10.03	1.55/1.60	0.12	0.5339	3.384	3.0	0.563	3.997	0.01	0.08
M-7	9.90	1.56/1.55	0.40	0.5337	3.385	3.5	0.563	3.997	0.00	0.03
M-9	10.10	1.64/-	0.05	0.5338	3.472	3.7	0.563	3.999	0.00	0.03
M-11	9.90	1.61/1.63	0.40	0.5339	3.472	-	0.563	3.998	0.03	0.25
M-12	9.99	1.63/1.66	0.11	0.5336	3.472	3.3	0.562	3.997	0.03	0.32
M-13	10.03	1.56/1.62	0.12	0.5334	3.386	3.1	0.562	3.997	0.01	0.09
M-14	9.90	1.61/1.65	0.40	0.5336	3.387	3.4	0.563	3.997	0.01	0.11
M-15	10.06	1.77/-	0.11	0.5337	3.471	3.3	0.562	3.997	0.22	3.68†
M-16	9.90	1.75/1.79	0.40	0.5336	3.473	3.1	0.561	3.997	0.12	3.22†

*Acceptance Limits: ϕ , 1200 °F - 0.19 cc/hr; 1400 °F - 1.60 cc/hr

†Rejected and being rebuilt

The production data from 43 delivered elements are shown in Table XXVI. As can be seen, the liquid honing caused a significant increase in the hydrogen loss rate from many elements.

The necessary components for the assembly of 25 S8DRM-1 fuel elements were fabricated this quarter. Assembly of these elements has been delayed until the previously mentioned performance test elements have been delivered.

b. Evaluation of Delivered Fuel Elements

The elements delivered this quarter were of as high quality as those used in the S8ER. These elements had similar dimensional discrepancies of bow and pin diameter size that were present in the S8ER elements, but to a lesser degree.

The hydrogen leak rates of the elements, as expected, can be grouped according to the carbon content of the fuel alloy which determines the hydrogen dissociation pressure. Table XXVII illustrates this grouping.

The difference of a factor of two between the as-fabricated average leak rate of elements with low carbon and high carbon is the same as that predicted from dissociation pressure data for the fuel. Although the statistics are poor, the liquid honing appears to have a greater effect on elements with a very low initial leak rate.

The addition of the liquid honing process to clean the fuel elements resulted in failure to complete the 55-element delivery this quarter. Twenty-nine of the 96 elements fabricated were liquid honed. The leak rate of 26 of these increased. In nine elements the increase was large enough to reject the elements.

Seven of the highest leak rate elements were given end-zone hydrogen permeation tests to determine which had the greatest leak rate. This test consisted of heating one end to 1400°F with the opposite end kept below 250°F, while the permeation rate was taken at the hot end. The ends were then reversed and the rate was measured at the other end. It appears from data shown in Table XXVIII that the major increase in leak rate is at the seal end of the elements.

TABLE XXVI
PERMEATION RATE OF DELIVERED PERFORMANCE
TEST ELEMENTS

Element Number	Carbon (wt %)	N _H	Permeation Rate, ϕ (cc/hr)	
			Assembly (STP)	After Liquid Honing (STP)
E-802-N	0.35	5.98	0.46	-
881	0.38	5.97	0.31	-
811	0.34	6.11	0.58	-
886	0.36	6.08	0.25	-
798	0.37	5.96	0.27	0.34
810	0.18	6.13	0.27	-
880	0.18	6.15	0.07	-
887	0.18	6.14	0.10	-
760	0.18	6.01	0.05	-
804	0.18	6.20	0.18	-
897	0.18	6.15	0.30	-
841	0.17	6.10	0.09	-
814	0.16	6.11	0.08	0.13
821	0.18	6.20	0.11	-
939	0.18	6.18	0.17	0.33
960	0.18	6.10	0.14	0.15
963*	0.18	6.11	0.13	0.23
972*	0.20	6.23	0.14	0.32
861*	0.21	6.20	0.17	0.42
884*	0.21	6.23	0.34	1.13
858	0.18	6.18	0.17	-
870	0.34	6.11	0.25	-
918	0.36	6.08	0.89	-
951	0.36	6.04	1.42	-
962	0.19	6.08	0.09	-
904	0.36	6.08	0.07	-
916	0.38	6.12	0.34	0.50
937	0.37	6.15	0.17	0.10
940	0.35	6.04	0.17	-
947	0.18	6.13	1.13	-
976	0.37	6.08	0.20	0.15
993	0.35	6.08	0.38	-
996	0.40	6.01	0.29	0.74
961	0.39	5.93	1.00	-
984	0.36	5.93	0.93	1.02
990	0.34	6.02	0.50	0.78
1011	0.18	6.17	0.05	0.51
1012*	0.15	6.18	0.24	0.79
992*	0.20	6.21	0.13	0.56
1000	0.37	6.04	0.37	-
1001	0.36	6.08	0.43	-
1017*	0.16	6.15	0.39	0.94
1009	0.15	6.13	0.43	-

*Increased axial and radial gap

TABLE XXVII

HYDROGEN LEAK RATES AS A FUNCTION OF CARBON
CONTENT IN THE FUEL ALLOY

Fuel Carbon Content	As-Fabricated Leak Rate	Leak Rate Before Liquid Honing	Leak Rate After Liquid Honing
	[cc(STP)/hr at 1400°F]		
Low Carbon (0.15 to 0.21 wt %)	0.22 (Average of 23 elements)	0.16 (Average of 10 elements)	0.46 (Average of 10 elements)
Medium Carbon (0.34 to 0.40 wt %)	0.47 (Average of 19 elements)	0.40 (Average of 7 elements)	0.58 (Average of 7 elements)

TABLE XXVIII

END ZONE HYDROGEN PERMEATION TEST DATA

Element Number	Leak For Complete Element (cc/hr)	Seal End at 1400°F (cc/hr)	Top End at 1400°F (cc/hr)
E-875-N	3.37	1.00	0.05
862	3.52	1.26	0.02
864	3.09	0.85	0.06
909	2.88	0.64	0.29
910	1.59	0.76	0.02
915	3.88	1.06	0.08
930	3.50	0.96	0.02

Note: Two end leak rates are not additive to obtain leak rate for complete element.

The data for all 96 elements fabricated this quarter are shown in Table XXIX.

c. Production Process Evaluation

A cost reduction through recycling fuel material was proven feasible. The material used consisted of outgassed cracked hydrided fuel rods, and cleaned machine turnings. The recovered scrap material was blended 25 to 75%

TABLE XXIX

HYDROGEN LEAK RATE DATA FOR ALL ELEMENTS FABRICATED DURING THIS QUARTER

Element Number	N _H	Carbon (wt %)	Hydrogen Leak Rate		Element Number	N _H	Carbon (wt %)	Hydrogen Leak Rate	
			Before Liquid Honing [cc(STP)/hr at 1400°F]	After Liquid Honing [cc(STP)/hr at 1400°F]				Before Liquid Honing [cc(STP)/hr at 1400°F]	After Liquid Honing [cc(STP)/hr at 1400°F]
E-760-N	6.01	0.18	0.05	-	E-916-N	6.12	0.28	0.34	0.50
775	6.02	0.41	1.94	-	918	6.08	0.36	0.89	-
798	5.96	0.37	0.27	0.34	920	6.22	0.21	1.22	1.81
802	5.98	0.35	0.46	-	922	6.11	0.36	1.86	-
804	6.20	0.18	0.18	-	925	6.08	0.36	2.78	-
810	6.13	0.18	0.27	-	926	6.13	0.18	1.86	-
811	6.11	0.35	0.58	-	930	6.07	0.35	3.50	-
814	6.11	0.16	0.08	0.13	932	5.93	0.36	1.80	-
821	6.20	0.18	0.11	-	933†	6.15	0.16	0.13	0.63
826	6.08	0.34	2.78	-	937	6.15	0.37	0.17	0.10
828	6.15	0.34	3.46	-	939	6.18	0.18	0.17	0.33
841	6.10	0.17	0.09	-	940	6.04	0.35	0.17	-
849	6.22	0.22	2.23	-	941	6.04	0.37	3.29	-
851	6.04	0.36	1.87	-	944	6.18	0.15	2.01	-
852	6.15	0.16	1.71	-	947	6.13	0.18	1.12	-
855	6.08	0.34	2.88	-	949	6.15	0.16	1.45	2.00
856	6.08	0.36	1.73	-	951	6.04	0.36	1.42	-
858	6.18	0.18	0.17	-	952	6.13	0.38	0.22	0.15
859*†	6.15	0.14	0.12	0.35	960	6.10	0.18	0.14	0.15
860	6.04	0.36	2.76	-	961	5.93	0.39	1.00	-
861*	6.20	0.21	0.17	0.42	962	6.08	0.19	0.09	-
862	6.08	0.36	3.52	-	963*	6.11	0.18	0.13	0.23
863†	6.03	0.34	0.62	-	967	6.15	0.34	0.44	-
864	5.98	0.35	3.09	-	972*	6.23	0.20	0.14	0.32
866	6.23	0.20	1.86	-	975	6.08	0.33	1.81	-
868	6.11	0.34	2.33	-	976	6.08	0.37	0.20	0.15
869*	6.11	0.18	0.28	2.33	980	6.08	0.34	0.42	-
870	6.11	0.34	0.25	-	984	5.93	0.36	0.93	1.02
871	6.08	0.36	2.54	-	987	6.22	0.21	0.12	-
874	6.10	0.40	1.25	-	988	6.15	0.34	0.38	-
880	6.15	0.18	0.07	-	989	6.19	0.41	0.36	-
881	5.97	0.38	0.31	-	990	6.02	0.34	0.50	0.78
884*	6.23	0.21	0.34	1.13	992*	6.21	0.20	0.13	0.56
885*	6.15	0.16	1.06	1.65	993	6.08	0.35	0.38	-
886	6.08	0.36	0.25	-	995	6.08	0.33	0.56	-
887	6.13	0.18	0.10	-	996	6.01	0.40	0.29	0.74
888†	6.00	0.34	1.33	-	1000	6.04	0.37	0.37	-
889	6.11	0.32	2.51	-	1001	6.08	0.36	0.43	-
891	5.93	0.36	1.37	1.96	1004	6.02	0.41	0.43	-
894†	6.08	0.18	0.08	0.08	1009	6.13	0.15	0.43	-
895*	6.21	0.20	1.49	2.49	1011	6.17	0.18	0.05	-
896†	6.18	0.18	0.79	-	1012*	6.18	0.15	0.24	0.79
897	6.15	0.18	0.30	-	1014	6.15	0.16	2.17	-
899	6.08	0.36	1.65	-	1017*	6.15	0.16	0.39	0.94
901*	6.18	0.15	0.17	1.65	1021†	6.07	0.35	0.69	-
904	6.08	0.36	0.07	-	1024	6.15	0.14	0.07	-
907	6.11	0.32	0.13	2.88					
909	6.21	0.20	0.38	-					
910	6.01	0.40	1.59	-					
912	6.11	0.36	3.40	-					
915	6.15	0.14	0.22	3.88					

*Modified radial and axial gap element.

†Elements did not meet final dimensional requirements and were rejected.

with virgin material to obtain four billets. No deleterious effects on alloy composition and hydriding of fuel rods was noticed. Based on this experiment, recycled material will be used in all future fuel rods as a part of the standard melting process.

During this report period, a modified firing jig was developed to help reduce the bow in the cladding tubes caused by firing of the ceramic coat. The jig supports the tubes being fired in the center, and prevents them from sagging during firing. The bow was harmful because it contributed to assembly difficulties.

The variables in the liquid honing process were investigated in detail. The optimum nozzle speed, nozzle pressure, distance from element to nozzle, and element rotation were determined. These are: nozzle speed of 20 in./min, nozzle pressure of 20 psig, distance of 4 in. from the element surface, and element rotation of 9 rpm.

Other general improvements in the assembly area include use of ultrasonic cleaning equipment to remove all grease and foreign matter from the tubes prior to coating, and the purchase of 150 plastic-lined aluminum shipping tubes for use in protecting cladding tubes in transit.

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VI. COMPONENT DEVELOPMENT

A. REFLECTOR AND DRIVES

1. Introduction and Summary

The reflector and drive system is being designed to meet the startup and operational requirements of the nuclear-reactor, proof-of-principle tests on the components required in the drive system. Work has continued on the conceptual design. Special consideration has been given to the effects of storage and launch conditions which are to be evaluated by prelaunch environmental testing as well as testing under the operational environment. The developmental status of the various components is discussed below.

2. Control Drum Bearings

The flight control drum bearings are being developed for minimum self-welding and operational friction. In order to prevent deterioration by oxidation of these materials it may be necessary to limit the environments during the ground tests. The effects of a slightly oxidizing atmosphere on the reflector bearing materials are being evaluated.

The present most promising bearing materials, P-5-N mechanical graphite and Alpha Molykote (X-15) dry film, have been extensively tested in accelerated oxidation tests. Other materials have been included in these tests for comparison and screening.

The means for accelerating these tests has been to maintain predicted reactor operating temperature but increase environmental pressure to yield oxygen exposure time equivalent to reactor ground test requirements. In other words, a reactor requirement of 10,000 hr at 1μ or $10,000\mu$ -hr is simulated by a test at 100μ for 100 hr for equivalent exposure of $10,000\mu$ -hr. The validity of this accelerated testing approach, of course, varies with any material tested and its oxidation characteristics. One must expect that actual long-term operation will be somewhat more stringent than the accelerated testing. If the oxidation relationship to pressure is assumed to be proportional to the ratio of test pressure to operating reference pressure to an exponent of one or less,*

*B. E. Hopkins, M. S. C., F. I. M., Oxidation of Metals and Alloys, Second Edition, Butterworth, London, page 46.

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Figure 39. Sliding Friction Test Samples,
100- μ , Pretest

2-14-63

7568-4752

Figure 40. Sliding Friction Test Samples,
100- μ , Post-Test

the credibility of the testing can be examined by comparing tests at different pressures. Since the data for P-5-N thus far for tests at different pressures is rather linear, the accelerated testing of P-5-N appears to be a sound approach.

Radiation effects upon these materials are not included in this test series, but will be evaluated later.

Thirteen P-5-N samples have been run in nine tests to date at 1100°F and pressures from 10 to 500 μ of Hg. Figures 39 and 40 show a representative example of P-5-N oxidation at 1100°F for 100 hr at 100- μ pressure. The unit weight loss per time-air integral exposure is plotted in Figure 41. The dimensional change of exposed surface per time-air integral exposure is given in Figure 42. These figures show extrapolation to 100 μ -hr which is the same integrated air exposure as 10^{-5} Torr for 10,000 hr. Since this is almost a two-decade extrapolation, a test is being set up to run at $\sim 100\mu$ -hr.

It is hoped that with this latter test information a substantiated conclusion can be made for recommending operating pressure for reactor ground tests with adequate margin for radiation effects. A study is in progress to predict radiation effects.

A fixture has been designed and is being manufactured to use with equipment which will test the actual S8DRM-1 reference design drum thrust bearing at temperature in poor vacuum environments. This test will establish the performance of the bearing under S8DS ground prototype system conditions.

Recent material friction tests in simulated space conditions have demonstrated improved performance using "dry film" lubricants. However, the very close fits required in the reflector drum bearings will necessitate very precise control of lubricant film thicknesses.

A controlled test series has been completed in which the effects of humidity on the dimensional stability of molybdenum disulfide (Molykote X-15) were investigated. The test was initiated because of unstable clearance on X-15-coated bearings. Eight inspection "Jo" blocks were coated with X-15 in varying thicknesses from ~ 0.0018 - to 0.0064-in. Weight changes and thickness changes were recorded after various humidity and bakeout cycles with the time intervals at ambient conditions minimized.

Figure 41. Purebon P-5-N Weight Loss Per Time-Air Integral Exposure at 1100°F

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Figure 42. Purebon P-5-N Surface Dimension Loss Per Time-Air Integral Exposure at 1100°F

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The results of the weight measurements indicate that the sodium silicate binder in the X-15 absorbs moisture when subjected to humidity and re-emits this moisture during a bake cycle.

From measurements taken when the X-15-coated "Jo" blocks were subjected to controlled humidifying and subsequent drying, there were indications of dimensional instability. However, the measurement of thin compactible films is rather complex. Careful analysis of the results does not yield definite trends or quantitative dimensional changes which correlate with the environmental exposure. More precise methods for measurement of thin films are being investigated.

The SNAP 8 reference design drum bearings were subjected to the SNAP 8 shock and vibration environment with no apparent damage.

3. Control Drive Gears

An ultra-high-vacuum gear test fixture which is a modification of a SNAP 10A fixture has been designed. The fixture will test prototype gears and will utilize the SNAP 8 stepper motor for accurate drive inputs. The S8DRM-1 reference design gear materials are Inconel-X pinion and gear with molybdenum disulfide dry film lubricant (X-15) on both.

The design of the basic gear test fixture, which utilizes a rack and pinion concept and actually isolates the gear reactions (all bearings are outside the ultra-high-vacuum chamber), is ~90% complete. Basic developmental evaluation of various material couples will be possible with this equipment.

4. Springs

Several S8DRM-1 ejection springs are being subjected to an elevated-temperature, constant-height, relaxation test. The spring material is René 41. Figure 43 shows the pretest measurement rig and test fixture. The test is being performed with the springs compressed to the present design load, which gives an initial room temperature nominal stress of 50,300 psi, and then heated to 1100°F. This test has run ~600 hr, but not enough data is available yet to predict load relaxation rate.

Recent investigation has indicated possible superior creep strength of tantalum base alloys for spring material at 1100°F. Testing of these alloys will have to be done in a nonoxidizing atmosphere, and a test fixture is being fabricated for vacuum-load testing.

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Figure 43. Ejection Spring Test Fixture

NAA-SR-8692

5. Ground Test Reflector Drive and Scram Mechanism

The operational safety considerations of ground nuclear testing require a method of shutting down the reactor without losing its restart capability. To simulate the space system control as closely as possible it is desirable to provide this capability with a minimum of modification to the flight system.

The proof-of-principle model of the design described in a previous quarterly progress report* was fabricated and is ready for assembly and test except for the installation and checkout of the high-temperature magnet coil in the release clutch. The coil was developed at AI and wound according to the wire size and number of turns specified by a clutch vendor. The clutch passed design verification tests, when tested with a low-temperature coil installed by the vendor, and has been delivered to AI.

The test procedure has been completed and final preparations of the environmental chamber are being made.

6. Advanced Design

The extreme environmental requirements imposed on the moving parts of the reactor control system, e.g., bearings, gears, and actuator brake; and the complex problems of choice of materials, fabrication technique, and lubrication have pointed up the possible need for a backup approach to component design and development. A design concept has been generated and a preliminary conceptual layout is underway which will permit sealing all of the control drum and actuator bearings, gears, brakes, and actuator winding insulation in a hermetically sealed container without the use of a "dynamic" (i.e., sliding) seal for shaft movement. The sealed control drive chamber will be filled with argon or another inert gas and will permit use of dry film lubricants.

B. ELECTROMAGNETIC DEVICES

1. Indirect Drive

The long-term actuators (Figure 44) are used during startup and during the life of the reactor to provide torque for rotating the reflector control drums bi-directionally when it is necessary to maintain the reactor outlet coolant temperature within the preset dead band.

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, October - December 1962," NAA-SR-8092 (SRD), May 20, 1963.

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Figure 44. Long-Term Actuator

During this quarter, a high-temperature actuator using SAP II insulation system was assembled and bench-tested (Figure 45). A 200-hr checkout at 1100°F was performed to determine the quality of the insulation system. During the SAP II curing process, hairline cracks were observed in the outer surface (Figure 46). Application of a thermica base material coating, SAP III, has eliminated these cracks (Figure 47).

Further testing using the SAP III coating over the SAP II insulation and checkout at 200 hr at 1100°F is planned.

2. Direct Drive Actuator

a. Linear Actuator

One long-term control drum actuator, the linear actuator, is designed to be connected directly to the reflector control drums through a coupling, thereby eliminating the gearing required with the present indirect drive design.

Designed as a solenoid-driven, reversible-stepping actuator, it will produce 14.5 in.-lb and a 0.521° angular step upon command signal, and will also hold 14.5 in.-lb fixed under zero-power conditions.

Fabrication of components for a high-temperature developmental model was initiated. A major part of the components for the linear actuator have been received. A new design, using pyrolytic graphite as a drive-shoe and hold-shoe material, has been detailed. This design incorporates recent results from the materials self-welding program. Fabrication of components using the graphite shoe material is being scheduled.

b. One-Half Degree Stepper

Detail design drawings were made on the high temperature 1/2° stepper, Model 3, and a majority of the parts for four test units have been received. Assembly and initial testing are planned.

Since insufficient data is available on brazing the two rotor materials (titanium and Hiperco), rotor brazing will be postponed until further tests on development parts have been evaluated. The brazing alloys currently being investigated are: (1) silver with 0.2% lithium, (2) silver with 5% aluminum, and (3) silver-copper eutectic with 0.5% lithium.

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Figure 45. High-Temperature Actuator Test Assembly

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Figure 46. First Actuator Insulation After Curing Process

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7573-18281

Figure 47. Improved Actuator Insulation After Curing Process

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3. Startup Actuator

The electromechanical startup actuators have been replaced by a spring-driven mechanical startup mechanism. The spring-driven device, which is more simple in design and more reliable, can now be used because the startup drum insertion rate has been increased.

4. Coil Winding Technique

The coil winding technique is essential to the design and fabrication of all types of actuators. The previous technique was the vacuum impregnation method. An improved technique is wet winding in which the insulation encapsulant and the wire are wound onto a bobbin simultaneously. The wet-winding method provides a more reliable insulation between the coil windings.*†

The wet-winding technique has been adapted for the fabrication of indirect-drive actuator stator coils (Figure 48).

Mykroy 1100 was used as a bobbin material. This material grows physically while being fabricated at elevated temperatures in the plane in which it is compressed. Tests are being made to determine the amount of growth which is very difficult to predict analytically. An effort is underway to find a better bobbin material and to determine whether a free-standing coil can be wound reliably.

The wire currently being used on the indirect drive actuator is chrome-plated copper with double-glass-served insulation. It is planned to use this wire on the direct-drive linear actuator and 1/2° stepper. An experimental nickel-clad silver wire with ceramic insulation is to be used on another model of the 1/2° stepper.

5. Cable Harness

Test results of the cable insulation material utilized on the S8DRM-1 reactor cable harness indicate that the filler material carbonizes at a temperature of 1000°F and vacuum. Therefore an extensive testing program is scheduled to evaluate all available insulation materials at temperature and in vacuum.

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, January - March 1963," NAA-SR-8392 (SRD), August 20, 1963.

†See also Section VI-D-4, this report.

A terminal adapter has been designed which facilitates attachment of the high-temperature cables to the electrical connectors. Previously, two parts were required: one to connect the cable leads to the terminal pins, and the other to terminate the insulation on the cable end. The new adapter combines both of these functions.

A preliminary test was performed on a cable harness assembly at 100% humidity and 70° F for 80 hr (Figure 49). The data indicate that the insulation resistance ranged between 5 and 20 megohms for the first 50 hours of the test. In the remaining 30 hours, the insulation resistance dropped to between 60 and 200K ohms. While testing thus far is inconclusive, the indication is that the cable harness will perform satisfactorily in the above environment for ground test.

C. CONTROL AND INSTRUMENTATION

1. Introduction and Summary

The reactor control and instrumentation system controls the reactor throughout its operating life, provides for electrically actuated shutdown, and monitors reactor operating parameters for transmission to the telemetry system. Primary emphasis during the reporting period continued to be placed on the control system.

2. Controller

Work on the reactor temperature controller continued to be devoted largely to the development of the basic components of the controller. Several logic designs were worked on for various types of controllers including a controller with adjustable timing periods for S8DS-1, one capable of controlling four linear actuators, and one for S8DS-1 scram kit testing.

The irradiation of the thermionic integrated micromodule components (TIMMS) was completed; however, complete results are not yet available. Development of circuit (TIMMS) was concentrated on the design and testing of various flip-flop modules.

Work on the all-magnetic or core-wire logic system was initiated. Development and fabrication has been initiated on the scaler-sequencer portion of the controller which will utilize only multi-aperture ferrite cores and copper wire. Development of radiation-hardened driver stages and output stages was also initiated.

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Figure 48. Indirect Drive Actuator Stator With Wet-Wound Coils

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Figure 49. Cable Harness Specimen
After Preliminary Humidity Test

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3. Programmer

Breadboard circuits utilizing magnetic core-transistor oscillators, incremental magnetic-core pulse dividers, and relay driver stagers were designed built, and evaluated for performance. From the most promising circuits, the S8DRM-1 programmer was designed and a model has been assembled.

4. Temperature Sensor-Switch

A study was initiated to explore the possibility of using a magnetic material as a temperature sensor utilizing the rapid change of saturation flux density (B_s) at the curie point (T_c). The first phase was to determine whether the curie point for a particular alloy remains stable and does not shift with time and temperature cycling. Three materials have been investigated; 50% nickel-iron (Orthonol), 3% silicon-iron (Magnesil), and 50% cobalt-iron (Supemendur). These materials have curie points of 930, 1380, and 1720°F, respectively. Tests on the nickel-iron indicated no measurable change in curie point over ~200 hr and 10 thermal cycles from ambient temperature to 50°F above the curie point. The results of the initial tests on the silicon-iron were not conclusive because the higher temperatures caused oxidation of the core and the winding. Tests are now being setup in a vacuum chamber to minimize these effects.

5. Diagnostic Instrumentation

A new regulator circuit was designed and tested for use in the position sensor demodulator. This regulator will hold constant current which will, in turn, hold constant excitation on the sensor even as its winding resistance changes with the increase from ambient to operating temperature. This will improve accuracy over the SNAP 10A constant-voltage regulator design.

D. MATERIALS

1. Introduction and Summary

Materials evaluation tests are being performed on low friction materials for use in bearings and gears, high friction materials for use in brakes, electrical insulation systems for use in actuators, and emissivity control coatings for use on reflectors. The performance of these materials is being evaluated in the combined environment of hard vacuum, high temperature, and radiation. Tests on the extent of self-welding of various pairs of materials are

leading to the selection of materials for the prevention of self-welding. Reflector coatings are being evaluated for their performance in emissivity control.

The best materials for the prevention of self-welding evaluated to date are the graphitic inserts of WCG cloth and "grafoil" sheet. Dry-film coatings using compounds such as MoS_2 , WS_2 , and WSe_2 appear to be promising anti-self-weld materials and are currently being evaluated.

The development of a process for the application of electrical insulation onto wire by the wet-winding technique continues. The objective is to produce a self-supporting electromagnetic coil which does not require a bobbin. Preliminary results on aging characteristics of coils produced by this technique indicate excellent performance. Work continues on the emittance coatings for beryllium reflectors. The apparent reliability of the anodized coating will be investigated further. The development of oxidation-resistant coatings for beryllium was carried out by evaluating three coatings at 1350 to 1550°F under two conditions of atmospheric moisture content. Preliminary results show that at least one of the coatings tested will be satisfactory in this application.

2. Friction Studies*

a. Dynamic Friction Tests

The first phase of ultra-high-vacuum bearing materials screening tests has been completed. These materials have been grouped into seven categories as follows:

Group A: Metal vs metal	15 combinations
Group B: Metal vs metal, with dry lubricant	8 combinations
Group C: Metal vs carbon-graphite	4 combinations
Group D: Ceramic vs ceramic	4 combinations
Group E: Ceramic vs ceramic, with dry lubricant	3 combinations
Group F: Metal vs ceramic	5 combinations
Group G: Ceramic vs carbon	8 combinations

*Jointly funded by SNAP 10A/2 and SNAP 8. This section also reported by R. J. Gimera in "SNAP 10A/2 Quarterly Progress Report, April - June 1963," NAA-SR-8694 (SRD), to be published.

Figure 50 illustrates the average results obtained from the screening tests. The groups, as listed, are shown graphically to provide comparative frictions for the total number of tests run in each group.

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Figure 50. Average Results for Materials Screening Tests

All tests were conducted in vacuum below 10^{-7} Torr and in many tests the pressure dropped below the lower reading limit of the gage (2×10^{-9} Torr). Only the values for starting (static) friction are used in determining the averages shown in the graph. It should be pointed out that running (dynamic) friction is, in most cases, much lower.

The second phase of materials screening is currently underway. These studies are divided into two parts: (1) the application and use of dry film lubricants, and (2) self-lubrication from a special impregnant that is contained in a high-temperature absorbent material such as carbon-graphite.

Sliding block friction tests are being conducted with carbon-graphites in combination with metals and ceramics. Data have been collected on P-5-N, P-03-N, P-03-NHT, and P-5. Figure 51 shows the effect of temperature on the coefficient of friction for these materials. Figure 52 is a plot of percentage of graphite vs coefficient of friction. Note that friction is minimized at between 30 and 60%. One of the reference design materials for bearings, Purebon P-5-N, is ~60% carbon and 40% graphite which further substantiates its use as a bearing material for this application.

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Figure 51. Friction Coefficient vs Temperature for Purebon Carbon Graphites Sliding Against Al_2O_3

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Figure 52. Friction Coefficient vs % Graphite for Purebon Materials Sliding Against Al_2O_3

Dry film lubricants having vapor pressures below 10^{-9} Torr at 1300°F will be necessary to meet the environmental requirements. Compounds of the refractory metals with sulfur, tellurium, or selenium appear to form a high-temperature, low-shear solid film when mixed with a binding agent. These compounds have been obtained and are being evaluated for their ability to reduce friction and prevent self-welding as a function of both temperature and load. Figures 53 and 54 show the results obtained for MoSe_2 as a function of temperature and time, respectively. Figures 55 and 56 show the effect of system pressure and temperature on Molykote X-15 in a controlled vacuum.

Both the carbon-graphites and the dry films have at least two modes of failure as a lubricant: oxidation and evaporation. Tests are being conducted on these materials in controlled vacuum to establish oxidation rates as a function of pressure at a fixed temperature. A new facility has been fabricated and is being assembled (Figure 57) to measure the weight loss of materials due to evaporation in ultra-high vacuum.

The use of a residual gas analyzer has been initiated in studying the behavior of carbon-graphite materials at high-temperatures in ultra-high vacuum. The rate of gas evolution and the composition of the gases evolved will be related to the life and lubricating properties of these materials. This analyzer will also aid in the study of the evaporation rates of various materials to be used for bearings, gears, and antiself-weld surfaces.

b. Development of Friction Coatings

The current state-of-the-art for dry-film lubricants indicates the use of the sulfides, selenides, and tellurides of either molybdenum or tungsten as low-shear materials. These materials are bound to one or both surfaces with the oxides of silicon and sodium. The latter compounds are usually introduced in the form of sodium silicate, which when dissolved in water is known as "water glass." Unfortunately it softens at a relatively low temperature. When the glass melts, it no longer fulfills its function as a dry film thus permitting the two surfaces to adhere. In an attempt to develop higher softening point glasses to bind calcium fluoride to mating surfaces it was found that these glasses might be good dry film lubricants in themselves. In friction tests AI glass 208 had a coefficient of friction, μ , of 0.26 at 1150°F in a 10^{-3} Torr atmosphere. The test was continued for 20,000 in. of travel showing the long wearing ability of these glass surfaces.

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Figure 53. Friction Coefficient vs Temperature for Drilube 867N (MoSe₂) Sliding Against Inconel-X

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Figure 54. Friction Coefficient vs Time for Drilube 867N (MoSe₂) Sliding Against Inconel-X

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Figure 55. Friction Coefficient vs Pressure for Molykote X-15 at a Fixed Temperature

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Figure 56. Friction Coefficient vs Temperature for Molykote X-15 in a Controlled Vacuum

- 1 AUTOMATIC RECORDER
- 2 IONIZATION GAGE POWER SUPPLY
- 3 ION PUMP POWER SUPPLY
- 4 HEATER CONTROL AND POWER SUPPLY
- 5 ROUGH PUMP VALVE
- 6 ROUGH PUMP
- 7 FLEXIBLE LINE
- 8 OIL DIFFUSION PUMP
- 9 COLD TRAP

- 10 FLEXIBLE LINE
- 11 FORE PUMP VALVE
- 12 BELL JAR AND AINSWORTH BALANCE
- 13 TEST CHAMBER
- 14 100 l/ SEC ION PUMP
- 15 ADJUSTABLE TABLE
- 16 LIQUID NITROGEN SHROUD
- 17 TEST SPECIMEN
- 18 EXTERNAL HEATER TUBULATION

8-20-63

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Figure 57. Evaporation Rate Facility Schematic

TABLE XXX
SELF-WELD DATA SUMMARY

Material Combinations	Temperature, Approximate (°F)	Time (hr)	Load* (psi)	Force of Adhesion* (psi)
316 stainless steel vs Molybdenum	1200	504	65.0	14.2
Beryllium vs Micaceram (Figures 58 and 59)	1300	1050	80.0	7.2
316 stainless steel vs Molybdenum + WCG graphite cloth (Figure 60)	900	2256	64.0	0.4
316 stainless steel vs Molybdenum and Grafoil	1100	2178	60.0	0.4
Al ₂ O ₃ vs Al ₂ O ₃	1280 [†]	550	64.8	0.4
		1130	80.0	0.4
316 stainless steel vs Al ₂ O ₃ + Grafoil (Figure 61)	1000	2330	65.0	0.4
	1000	2280	65.0	0.4
Beryllium vs Mykroy (Figure 62)	1000	1200	65.0	4.0
316 stainless steel vs Micaceram	1100	1440	65.0	24.0
K162B vs K162B	1200	1200	65.0	1.6
Cr ₂ O ₃ vs Cr ₂ O ₃ [§]	950	312	62.8	0.4

*The apparent area of contact is 0.25 in.²

[†]The initial temperature was 450°F and the final temperature was 1000°F. A maximum temperature of 1200°F was obtained after 940 hr of operation. The mating surfaces showed the presence of a foreign material of unknown composition.

[§]The Cr₂O₃ coating was severely cracked after the test.

A modification in the chemical composition of this AI glass (208A) resulted in an increase in the softening point from 1150 to 1525°F. Friction tests have been run at 1100°F and 10^{-3} Torr to determine the effect of change in asperity in these glasses. The glass-208A surfaces of three coated specimens were abraded with abrasives of different coarseness. Use of the least coarse abrasive resulted in a μ of 0.26; medium coarse abrasive, μ of 0.46, and the most coarse abrasive 0.66. These tests are being replicated to verify the results. Further studies will be made on the effects of chemical change in the glasses and the resultant coefficient of friction at 1100°F and 10^{-3} Torr.

3. Self-Welding Studies*

Twelve combinations were evaluated during this quarter. The results of these tests are summarized in Table XXX and illustrated in Figures 58 through 62. In every test to date, Micaceram and Mykroy have self-welded to their respective mating samples. The graphitic inserts of WCG cloth and Grafoil sheet proved to be the best materials for preventing self-welding. Dry film coatings using compounds such as MoS_2 , WS_2 , and WSe_2 appear to be very promising antiseal materials. These compounds are presently being evaluated.

4. Electrical Insulation

Wet-winding process development conducted over the last year has resulted in the fabrication process shown in the diagram in Figure 63. The objective of this process is to produce a self-supporting electromagnetic coil which does not require a bobbin. The viscosity of the paste (insulation plus water) is controlled to provide a continuous coating of ceramic (vitreous) encapsulant to the magnet wire. The paste contains <33% moisture and the subsequent drying enroute to the coil winder results in uniform high solid-content standoff electrical insulation.

The aging characteristics in 10^{-5} Torr of a vitreous encapsulated coil prepared by the above process were compared to those of a SAP II coil prepared by the technique of dripping the encapsulant on the coil just prior to forming on the winder.

*Jointly funded by SNAP 10A/2 and SNAP 8. This section also reported by R. J. Gimera in "SNAP 10A/2 Quarterly Progress Report, April-June 1963," NAA-SR-8694 (SRD), to be published.

BERYLLIUM

8X

MICACERAM

8X

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Figure 58. Self-Welding of Beryllium
Against Micaceram (8X)

BERYLLIUM

50X

MICACERAM

50X

4-4-63

7573-47111

Figure 59. Self-Welding of Beryllium
Against Micaceram (50X)

MOLYBDENUM

250X

316 STAINLESS STEEL

100X

4-12-63

7573-47107

Figure 60. Self-Welding of Molybdenum
Against 316 SS, Insert Material:
WCG Graphite Cloth

316 STAINLESS STEEL

10X

316 STAINLESS STEEL, Al_2O_3

10X

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Figure 61. Self-Welding of 316 SS Against
316 SS, Al_2O_3 After 2,330 hr

MYKROY

10X

BERYLLIUM

10X

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Figure 62. Self-Welding of Mykroy
Against Beryllium

8-20-63

Figure 63. Schematic of Process for Fabricating Self-Supporting Electromagnetic Devices

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Figure 64. Adjustment in Viscosity with Change in Shear for PBX and Vitreous Encapsulants

Virtually no aging occurred for the case of the vitreous encapsulated coil; SAP II exhibited its usual drastic aging under these conditions (temperature 800 to 1000°F, 10^{-5} Torr). Additional aging studies on similar coils are presently being conducted.

Figure 64 compares the viscosity of PBX and a vitreous paste encapsulant. The thixotropic nature of the vitreous paste, a requisite for self-supporting coil fabrication, is apparent.

5. Emittance Coatings for Beryllium Reflectors

The objective of this study is to determine whether plasma-sprayed chrome-oxide coatings would be reduced to chromium metal in a 10^{-9} Torr atmosphere at up to 1100°F and whether under these conditions anodized coatings would evaporate. The emittance would decline radically if either of the above processes occurred. The results shown in Table XXXI indicate that this has not been occurring under the conditions of these tests.

The apparent reliability of the anodized coating versus the plasma-sprayed chrome oxide will be investigated further by running 30 samples of the anodized beryllium against 15 samples of the reference coating and 15 samples provided by a different vendor.

6. Protective Coatings for Beryllium

The development of oxidation-resistant coatings for beryllium was continued at Armour Research Foundation (ARF). The objective of the work during this reporting period was to obtain data at 1550°F which would permit extrapolations to the reference design temperature of 1350°F. Three coatings were evaluated: Brush Beryllium Company anodized beryllium, ARF anodized beryllium, and ARF beryllium-silicon alloy.

The catastrophic oxidation of beryllium is dependent on moisture content, as well as temperature. Tests run at 1550°F were exposed to two moisture contents: +6°F dewpoint, and +32°F dewpoint (Table XXXII). These results show that anodized coatings will last 6 to 12 times longer by utilization of the drier atmosphere. These data also show that there would be at least a 12-fold gain in coating life by a reduction in the test temperature from 1550 to 1350°F.

On the basis of these results, it was determined that these coatings will protect beryllium against oxidation for 10,000 hr at 1350°F in 2% oxygen, 98% nitrogen at 6°F dewpoint. Replication of these results is being conducted to verify these conclusions.

TABLE XXXI
EFFECT OF AGING ON THE EMITTANCE OF COATINGS FOR BERYLLIUM

	Material	Vendor	Time of Test (hr)	Emissivity		
				Pretest (E ₀)	E ₁	E ₂
Coating Technique	Chrome-Oxide*	Plasmadyne				
	1st Lot		1200	0.885	0.90	0.94
	2nd Lot		400	0.875	0.87	Still Running
Plasma-sprayed	Chrome Metal-Air Cover	Plasmadyne	400	0.60	0.73	Still Running
	Chrome Oxide	Advanced Processes Company	800	0.90	0.905	0.86
	Reference Coating†	Plasmakote	800 400	0.87	0.865 Spalled	Spalled
Anodized	Proprietary	Brush Beryllium Company	800 400	0.77	0.75 0.94	0.90 Still Running

*An argon cover gas was used followed by an air cover.

†The 10A reference coating is applied in three successive coats, as follows:
1st Coat - beryllium metal; 2nd Coat - 1/2 beryllium metal, 1/2 chrome oxide; 3rd Coat - chrome oxide.

TABLE XXXII
ACCELERATED AGING TESTS FOR BERYLLIUM*

Test Temperature (°F)	Dew Point (°F)	Test Duration (hr)			
		Bare Beryllium	Brush Beryllium Company Anodized Beryllium	Armour Research Anodized Beryllium	Armour Research Beryllium-Silicon Alloy
1350	+32	50 to 100 Failed	1300 No Failure	1204 No Failure	1300 No Failure
1550	+ 6	50 to 100 Failed	670 Failed	1200 Failed	1300 No Failure
1550	+32	16 Failed	110 Failed	100 Failed	340 Failed

*The atmosphere in all these tests was 2% oxygen plus 98% nitrogen

7. Beryllium Oxidation Studies

a. Oxidation of Beryllium 1 Wt % Calcium in Dry Oxygen

Since a preliminary investigation showed that beryllium 1 wt % calcium alloy had excellent oxidation resistance properties in both moist and dry oxygen at 750°C, the experiments were extended to cover the temperature range, 500 to 1050°C, in dry oxygen. Table XXXIII shows total exposure time and initial parabolic rate constant.

TABLE XXXIII
BERYLLIUM 1 Wt % CALCIUM OXIDATION DATA

Temperature (°C)	Total Exposure Time (hr)	Initial Parabolic Rate Constant [(mg/cm ²) ² /min]
516	230	3.6×10^{-8}
662	150	1.9×10^{-6}
750	200	3.2×10^{-6}
862	150	4.8×10^{-6}
976	100	2.4×10^{-5}
1050	70	2.3×10^{-4}

The parabolic constants are higher than those of commercial beryllium by a factor of about 10 with the temperature dependence being about the same as for commercial beryllium; i. e., an activation energy of about 30 kcal. The significant point with regard to oxidation resistance is that at 750°C and below, after a few hours the rate decreased below the parabolic curve and after 25 to 40 hr no further oxidation occurred. Above 750°C, although oxidation did not cease neither did it accelerate, indicating a high degree of stability towards catastrophic failure. The oxide film in the case of the 1050°C run was 6μ thick.

b. Comparison of Several Beryllia in Moist Oxygen at 750°C

The following specimens were oxidized in oxygen saturated with water vapor at room temperature: (1) brush anodized beryllium; (2) 1 wt % calcium alloy; (3) commercial Be-S200 preoxidized at 900°C; and (4) uncoated commercial Be-S200. The objective was to evaluate the effectiveness of the various techniques for improving resistance to catastrophic oxidation. Table XXXIV shows time to acceleration along with total exposure time.

TABLE XXXIV
BERYLLIUM OXIDATION TEST DATA

Material	Time to Acceleration (hr)	Total Exposure Time (hr)
Brush anodized	0	115
1 wt % calcium	350	350
Be-S200, preoxidized at 900°C	10	15
Be-S200, uncoated	1/2	5

Both the anodized beryllium and 1 wt % calcium alloy showed excellent catastrophic oxidation resistance to moisture at 750°C.

E. SAFETY COMPONENTS

1. Introduction and Summary

The objective of the safety component development program is to provide the conceptual design and development of the various devices to fulfill reactor system safety requirements during reactor launch, prestartup, and during and after the operational life. Also, parametric design information must be generated to permit the design of reliable components for any revised requirements or advanced systems. An explosive reactor destruction system, which is ejected prior to reactor startup, is being provided to reduce the core and the fuel to noncritical masses in the event launch failure occurs. The operational reactor shutdown concept is based upon separating the reflector retaining band to initiate reflector ejection. Several devices which operate in series are being developed.

2. Electrically Operated Fusible Link Actuator

This component provides the reactor shutdown capability either by ground telemetry command or on-board initiated command.

An improvement was made in the basic actuator design by modifications in the heater design and the fabrication technique. The heater design modification, which consisted of using a smaller diameter wire and making changes in the configuration, was directed toward improving the efficiency and thereby reducing the electrical input. The smaller diameter wire improves the heater efficiency but also reduces the safety margin before heater burnout. Of the 10 development models built with the smaller diameter wire, 5 were successfully operated.

The other modification which was completed should improve heater reliability as well as simplify fabrication. This design utilizes a small ceramic mandrel which serves as the core for winding the heater. Heater burnout is reduced because a fixed separation of the heater coils can be maintained. The parts for this mandrel design are being fabricated.

The evaluation of various heater encapsulating materials at temperature and vacuum was made. The tests were conducted by checking the heater to ease resistance of the actuator assembled with the various insulating materials. The results of one of the tests up to 220 hr are shown in Figure 65 (attached graph). As noted, three of the materials show very favorable performance.

Another modification is use of stainless-steel sheathed thermocouple type wire as the heater elements. This design has the advantage of good heater reliability since the heater coil insulation is controlled during manufacture (see Figure 66). However, the currently available sheathed wire requires considerably greater electrical input and a longer actuation time. Efforts are currently underway to obtain samples with other heater wire sizes and insulating materials.

3. Thermomechanical End-of-Life Actuator

The thermomechanical end-of-life actuator is a device which senses the decay in the temperature of the coolant at the end of useful reactor life. When a predetermined low temperature drop is reached, the actuator will fracture the flexural rupture member in the reflector retaining band resulting in band separation. This actuator will be required if the command shutdown actuator fails to operate.

The first prototype of the SNAP 8 actuator has been designed and fabricated except for the molybdenum parts, which had to be remade due to cracking of threaded portions during machining. This problem is being corrected by more carefully controlled machining procedures.

Because of the small deflection produced by temperature changes in the actuator, the tensile strength and elongation of the rupture member must be predicted accurately. The rupture member must have adequate tensile strength to withstand the launch loads with a safety margin, yet rupture at a predictable stress with minimum elongation. A test fixture was designed and is being fabricated for use in a standard tensile test machine for accurate determination of the mechanical properties of the several ceramic materials under study for the rupture element.

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Figure 65. Electrical Insulation Resistance vs Temperature

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Figure 66. Heaters Wound with Stainless-Steel Sheathed Thermocouple Wire

Figure 67. Strength of Band and Braze Joint as a Function of Temperature

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8-20-63

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Figure 68. Creep of Reentry Joint Specimen at 800°F

4. Reentry Band

The reflector retaining band is the final reactor shutdown backup to the electrically operated fusible link and the thermomechanical end-of-life shutdown actuators. The brazed joints in the reentry band must be designed to fail when they become heated during the early aerodynamic heating phase. However, the band design must also provide adequate prevention of excess tensile creep during the reactor operational lifetime.

Short-time tensile yield — 0.2% — and strength at various temperatures were determined for brazed joint samples of the current design. These results are shown on Figure 67. The stresses are based upon the original cross-section of the band and on the original shear area of the joint. The long-term creep characteristic of the current design was also determined at 850°F and 500 psi. The results are shown in Figure 68.

F. SHIELD

1. Introduction and Summary

The objectives of the shield development effort are to provide the SNAP 8 program with the required fabrication processes and engineering properties of high efficiency, minimum weight, gamma and neutron shielding. The current program centers mainly around lithium hydride casting and associated supporting development.

During the past quarter, the major accomplishments were: (a) casting and delivery of the SNAP 8 DRM-1 two-region shadow shield; (b) completion of the lithium hydride corrosion tests; (c) continuation of the hydrogen migration study; (d) continuation of the proof-testing phase of the shield emissivity coating study; and (e) continuation of the heat cycling tests on a thermocouple instrumented shield to develop heat transfer data.

2. S8DRM-1 Shield

The SNAP 8 DRM-1 two-region shield vessel was completed and delivered to AI for buildup into a shield casting assembly. Figure 69 shows the shield casting assembly consisting of the shield vessel, filler tube-bellows and reservoir, etc., ready for installation in the safety container. The shield vessel and casting assembly were fabricated under rigid welding and leak-checking

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Figure 69. S8DRM-1 Shield Vessel Ready for Installation
in the Safety Container

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specifications to reduce or eliminate defects or leaks in the shield vessel which might appear during the casting cycle.

Upon receipt of the assembly at the Santa Susana Casting Facility, granulated lithium hydride was loaded into the two regions of the shield. Natural lithium hydride was used in both regions of the shield, since the specimen will not be used for nuclear testing. Difficulty was encountered in loading both of the regions, and it was possible to add only a minimum amount of lithium hydride to the shield assembly. It was decided to melt the lithium hydride in the reservoir, in order to provide additional volume to permit the addition of a definite excess of material. The melting operation was conducted successfully by heating only the reservoir while maintaining the bulk of the assembly well below the melting point of lithium hydride (1270°F). This procedure permitted the addition of a total of 777 lb of lithium hydride (542 lb large region - 235 lb small region) to the shield assembly.

The shield assembly was loaded onto the retort base and the retort installed, evacuated and backfilled with purified hydrogen to 850 mm Hg pressure. About 30 hr of heating with the bell-type furnace were required to melt the charge. The furnace was raised partially to begin the directional cooling in the same manner as used for the previous castings. Periodically the furnace was raised in 2-in. increments during the 97-hr cooling period. At this time (total 127 hr), freezing was observed to have progressed into the filler tube-bellows, and the furnace was removed completely from the retort.

All systems and components operated satisfactorily during the operation, except one of the two filler tube heaters, and one thermocouple which monitored the lithium hydride temperature near the middle of the shield. Neither loss interfered with the operation.

It was necessary to allow the assembly to cool for about three days before the temperature was low enough for the retort to be opened safely. Figure 70 shows the shield assembly removed from the safety container after casting. A circular stain, characteristic of a small leak, was observed at that time on the conical shield wall at the junction of the vertical and upper circumferential welds. The center of the stained area (Figure 71) was located at the top tack weld which anchors a cable harness holddown nut.

Figure 70. Shield Assembly, Removed from
Safety Container After Casting, Showing
Spot Indicative of a Leak

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Figure 71. Closeup of Stained
Shield Vessel

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A series of x-rays were taken to establish the soundness of the casting, but the x-rays failed to define the interior of the casting clearly. It was indicated that no major voids or "shrink-pipes" were present.

The filler tube was cut and the void volumes of the two regions determined to establish the bulk density of the casting. The residual amount of lithium hydride in each reservoir also was determined to obtain a comparison bulk density value. Table XXXV shows the void volumes and percentage of full-can density as determined by both methods for both regions of the shield. The two percentages agree very well, and the average percent full-can density (93.5) is equivalent

TABLE XXXV
VOID VOLUME AND PERCENT DENSITY OF S8DRM-1
TWO-REGION SHIELD

	Large Region	Small Region	Total
Wt LiH added, lb	542	235	777
Excess in reservoir, lb	49	45.5	94.5
Excess in neck, lb	3	2	5
Net wt LiH, lb	490	187.5	677.5
Void volume (V_v), cu ft	0.688	2.235	0.923
Volume, total, cu ft	10.75	4.18	14.93
% Full can density (V_v)	93.6	94.4	93.8
% Full can density (ρ)	93.6	92.2	93.2

to a lithium hydride bulk density of 0.729 g/cc. Correspondingly, the net weight of lithium hydride in the shield is 680 lb as compared to the precasting calculated weight of 681 lb.

After the void volume measurements were made and the closure cap was welded onto the filler tube stub (Figure 72), an extensive inspection was made to identify the suspected leak at the shield vessel scram weld. Neither a dye-penetrant check nor a helium leak detector established the presence of a leak. Eventually, a very careful grinding and polishing operation plus Zyglo inspection revealed a pinhole. The hole was traced through about 0.030-in. of metal by repetitively grinding, polishing, and inspecting until it was concluded that the hole penetrated the steel completely. The ground-out area was heliarc-weld filled with stainless-steel filler rod, inspected and found to be satisfactory.

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Figure 72. Completed Shield Vessel with Final Closure

3. Supporting Development

a. Thermal Test Shield

Work continued on the thermal conductivity experiment to determine an equivalent value for the conductivity of a cast lithium hydride shield. Initially, the experimental procedure consisted of cycling the shield from room temperature to about 1150°F and using the temperature history of the surface as input for the TAP-2 code. By varying conductivity values in the code, the calculated centerline temperature history is made to correspond to the experimental history, thus giving an overall equivalent conductivity value.

Partial analysis of the early runs clearly showed that a constant value could not be determined over this temperature range by this method and also that the equivalent value was not a straight-line function.

The cycling procedure, therefore, has been changed, so that the temperature range has been split into 200-degree increments. A constant value for the conductivity is determined through use of the TAP code and a plot of equivalent conductivity vs temperature is determined for the temperature range. The same procedure is being repeated using 200°F cool-down steps, with the expectation that an effective conductivity value closer to the actual performance characteristics of a SNAP shield will result.

An identical set of test runs are also being made using argon as a backfill gas instead of helium as in the first runs. These runs are being made to study the effect of the backfill gas on the conductivity value.

Figure 73 shows a typical centerline heatup curve with the results of the computer calculations simulating the test run. The conductivities used are as shown, and in this particular case, it appears that the conductivity is about 2.6 (in the Btu-ft system of units).

b. Emissivity Coating Study

A shield-case coating with high emissivity is required to reduce the temperature of the shield material.

Porcelain enamel emissive coatings with titanium-oxide and chrome-oxide opacifiers were tested under conditions similar to those to which an actual SNAP 8 shield would be exposed during the casting operation. The temperature-atmosphere test environments are presented in Table XXXVI. Figure 74 shows

Figure 73. Typical
Centerline Heatup
Curve

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TABLE XXXVI
EMISSION COATING TEST ENVIRONMENT

	Porcelain Enamel	
	Chrome Oxide	Titanium Oxide
Thickness, mils	4	4
Average LiH temp, °F	1280	1300
Time exposed to H ₂ , hr	105	66.5
Time exposed to Ar, hr	304	217.5

the chrome-oxide — porcelain-enamel test container prior to filling with lithium hydride. The filled test container was loaded into a stainless-steel retort which was sealed with an upper flange. The retort was placed in a furnace heated to the test temperature. The test capsule was exposed to a hydrogen atmosphere, but for safety reasons the hydrogen was replaced with argon when the test was left unattended.

The temperature inside the capsule was monitored by a stainless-steel sheathed, magnesia-insulated, Chromel-Alumel thermocouple welded into

the bottom of the capsule with the thermocouple junction near the center of the capsule. A second thermocouple was spot-welded to the outside of the reservoir.

Figure 75 shows the chrome-oxide — porcelain-enamel-coated container after testing in the heated atmosphere of hydrogen (or argon) gas and lithium hydride vapor (or its decomposition products). Definite corrosion of the coating was noted, especially in spots where solid or liquid lithium hydride appeared to have come in contact with the coating. The coating was believed to be unfit for SNAP 8 shield application.

The porcelain-enamel (titanium-oxide opacifier) coated test container is shown in Figure 76 before testing. Coated stainless-steel discs also were exposed at the same time. Results of the test exposure are shown in Figure 77. The coating was observed to be darker, but retained a smooth finish and showed no signs of corrosion.

One-inch-wide circumferential strips were cut from the container and 6 one-inch-sq specimens were cut from the strips to be tested for total normal emittance. It was noted that the coating was still intact and showed no signs of coming off even after flexing the flattened strip. The results of the emittance tests will be available next quarter.

c. Graphite Braze Study

A study was initiated to develop a technique for brazing graphite to a 300-series stainless steel. Three braze alloys were considered: (1) copper — silver; (2) gold — nickel — tantalum; and (3) zirconium — titanium. Of these alloys, the Cu-Ag is the easiest to work with and will wet the materials to be brazed. Test braze samples using the Cu-Ag alloy bonded well, but the graphite cracked due to the difference in thermal expansion between the stainless steel and the graphite.

The use of a tantalum or molybdenum intermediate layer was investigated to relieve the stress causing the graphite to crack. A suitable braze alloy has not been found to bond the tantalum and graphite. The molybdenum-graphite bond can be made with Cu-Ag alloy, and possibly with Cu-Ag-Ti alloy.

The graphite-molybdenum-stainless steel-braze concept seems to be the best method of obtaining a good bond, and during the next quarter a set of test samples will be prepared for thermal cycling and long-time static heating tests.

Figure 74. Test Container Coated with
Porcelain Enamel and Chrome-Oxide
Opacifier, Shown Prior to Loading
with Lithium Hydride

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Figure 75. Test Container Coated
with Porcelain Enamel and
Chrome-Oxide Opacifier,
Shown After Exposure

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Figure 76. Test Container Coated with
Porcelain Enamel and Titanium-
Oxide Opacifier, Shown
Before Exposure

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Figure 77. Test Container Coated with
Porcelain Enamel and Titanium-
Oxide Opacifier, Shown
After Exposure

d. Hydrogen Migration Tests

Four of the six hydrogen migration capsules were tested and sectioned into wafers (slices through the stainless-steel tube and the cast lithium hydride) for chemical analysis (H, Li, OH⁻ and CO₃²⁻ content). Only partial results have been obtained to date on three of the four capsules.

Table XXXVII presents the values of wt % hydrogen along the length of the capsule on samples removed from the inner core, where impurities in the

TABLE XXXVII
HYDROGEN CONTENT ALONG LENGTH OF HYDROGEN
MIGRATION CAPSULES

Capsule and Wafer Number	Temperature at Wafer (°F)	Hydrogen (wt %)		Average
		Inner	Outer	
1 (250 hr)	A	1100	11.9 and 12.4*	12.4
	B	600	12.2	11.9
	C	500	11.8	11.6
	D	500	9.4 and 10.6*	10.1
	E	500	11.9 and 12.1*	12.3
3 (1000 hr)	A	1100	12.2	12.4
	AA	1100	—	—
	B	1100	12.2	12.4
	C	1100	12.9	12.4
	D	1100	12.3	12.3
	E	1100	12.4	12.5
6 (1000 hr)	A	1100	11.0	11.15
	AA	900	12.0	11.85
	B	600	11.1 and 11.4*	11.63
	C	500	10.0	10.4
	D	500	9.8	9.95
	E	500	11.7	12.0

*Two values reported

lithium hydride might tend to concentrate during casting of the capsule and the outer edge. The wt % H in the lithium hydride before testing varied from 12.1 to 12.4. The control capsule (No. 3), which was heated at 1100°F for 1000 hr, was found to be essentially unchanged in hydrogen content along the length of the capsule.

The test results to date are shown in Figure 78 for capsules 1 and 6 which were heated for 250 and 1000 hr, respectively, with a temperature gradient along the length of the capsule as indicated. The wt % H data appear to be very similar for the two capsules; however, the results are unexplainably low, and show a minimum in the 500° F region about 15 in. from the hottest end of the capsule.

Loss of H₂ from the capsule appears probable, although the calculated loss by diffusion is <0.2 wt % H during 1000 hr. In addition, no apparent loss was observed in the control capsule. If a leak caused the loss of hydrogen, both capsules apparently had leaks in the same region which is possible, but unlikely.

Duplicate sets of samples will be chemically analyzed to verify the data and, in addition, samples from other portions of the capsule will be analyzed to obtain a more complete hydrogen profile. Additional capsules will be tested, as needed, to clarify the anomalous results obtained to date.

e. Lithium Hydride Corrosion Studies

Corrosion testing of stainless steel exposed to lithium hydride at 900, 1200, and 1400° F for 1000, 2000, and 4000 hr was completed.

Weight loss tabs of three types of stainless steels 316, 347, and 19-9DL, were exposed to lithium hydride, then cleaned and weighed. The weight changes are listed in Table XXXVIII.

TABLE XXXVIII
WEIGHT CHANGES IN STAINLESS STEELS AFTER EXPOSURE
TO LITHIUM HYDRIDE

Temperature (°F)	Weight Change in 347 SS (mg/sq in.)			Weight Change in 316 SS (mg/sq in.)			Weight Change in 19-9DLSS (mg/sq in.)		
	1000 Hr	2000 Hr	4000 Hr	1000 Hr	2000 Hr	4000 Hr	1000 Hr	2000 Hr	4000 Hr
900	-0.56	-0.51	+0.29	-0.86	-0.61	+0.78	-0.26	+2.01	-5.88
1200	+1.46	+2.6	+2.76	+1.39	+2.05	+1.37	-0.35	+1.69	-7.94
1400	-4.19	-6.55	-8.61	-1.91	-1.63	-1.13	-0.18	+2.86	-9.65

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Figure 78. Hydrogen Migration Capsule Hydrogen Content

All three materials exhibited the same order of magnitude changes, although 19-9DL showed the largest change and 347 SS showed the least. The weight change in 316 SS does not appear to be time-dependent. However, the weight changes in both 347 SS and 19-9DL do appear to be time-dependent. When comparing the 4000-hr exposure data with shorter term data, it is obvious that different mechanisms of attack are operating at different temperatures. The nature of these mechanisms has not been determined.

The mechanical properties of 347 SS exposed to lithium hydride for 2000 and 4000 hr are shown in Figure 79 through 84. For purposes of comparison, the previously reported 1000-hr data have been included in Figures 85 through 87. Comparison of the control and exposed properties after 4000 hr indicates that the same general temperature effect occurs in both control and exposed specimens. However, the effect of exposure to lithium hydride seems to be less than the experimental error and, for all practical purposes, may be disregarded.

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Figure 79. Ultimate Tensile Strength of 347 SS Exposed to LiH for 2000 hr

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Figure 80. 0.2% Yield Tensile Strength of 347 SS Exposed to LiH for 2000 hr

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Figure 81. Percent Elongation of 347 SS Exposed to LiH for 2000 hr

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Figure 82. Ultimate Tensile Strength of 347 SS Exposed to LiH for 4000 hr

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Figure 83. 0.2% Yield Tensile Strength of 347 SS Exposed to LiH for 4000 hr

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Figure 84. Percent Elongation of 347 SS Exposed to LiH for 4000 hr

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Figure 85. Ultimate Tensile Strength of 347 SS Exposed to LiH for 1000 hr

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Figure 86. 0.2% Yield Tensile Strength of 347 SS Exposed to LiH for 1000 hr

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Figure 87. Percent Elongation of 347 SS Exposed to LiH for 1000 hr

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VII. SYSTEM TESTS

A. SNAP 8 EXPERIMENTAL REACTOR (S8ER)

1. Summary

The objectives of the S8ER test program are to demonstrate reactor operation and determine reactor performance characteristics at both 450 and 600 kw. The tests are required to verify the reactor design and to provide experimental data upon which to base the design of the flight reactor, especially with regard to reactor stability, temperature coefficients, power coefficients, and hydrogen loss rate.

A unitized core loading procedure was developed to replace the original remote-handling, one-at-a-time, core loading technique. The advantage of loading the fuel-moderator elements as a unit-package is the greatly decreased probability of fuel element damage. Tools were developed to perform unitized loading of the S8ER core. Core loading was accomplished in the Critical Experiment Facility at Santa Susana (Building 012), and the core was then placed in the core vessel in the S8ER Test Facility (Building 010).

A report* was submitted to the AEC on containment vessel nil ductility transition temperature, and proposed methods which will assure ductility of the vessel throughout S8ER operations. An amplification of certain safeguards matters requested by the AEC was included in this report.

AEC approval for nuclear operations was received on May 17, 1963. Initial criticality of the S8ER (without coolant in the core) was achieved at 2153 on May 19, 1963. The "dry critical" experiments included determination of a source-independent critical position and calibration of two control drums.

The reactor was shut down for the final closure weld and completion of heat transfer and vent piping within the containment vessel. At this time, a welded rupture disc assembly was installed on the primary coolant system. During the reactor outage, a system was installed to heat the containment vessel wall in order to eliminate risk of brittle failure at any time in the life of S8ER. The primary and secondary heat transfer systems were filled with NaK and heat transfer tests were conducted for the systems and the reactor core.

*J. C. McKinley, "SNAP 8 Experimental Reactor (S8ER) Final Safeguards Summary Report, Addendum III, "NAA-SR-MEMO-8353, March 28, 1963

During this report period, S8ER operation has been demonstrated with a dry core and the reactor has been readied for operation with the coolant. Experimental data has been obtained from reactor performance with a dry core and from performance of the heat transfer systems.

2. Experimental Results

The dry critical phase of the S8ER test program was completed. The dry tests consisted of the unitized core loading, approach to critical control element configuration, and a limited control element calibration.

Although accurate parameter analysis was not possible due to the limited dry critical tests performed, some preliminary parameter values can be estimated. These values are presented below.

Control element worth with adjacent elements in	\$4.10
Control element worth with one adjacent element in, and one adjacent element out	\$3.30
Total cold, dry excess worth	\$9.80

A detailed investigation of control element worths and interaction effects as well as a determination of the isothermal temperature coefficient and the zero power hydrogen loss rate will be performed during the wet critical tests.

Upon the completion of these tests, a more accurate and detailed analysis of reactor parameters will be performed.

3. Unitized Core Installation

a. Introduction

A unitized core loading procedure was developed to replace the original, remote-handling, one-at-a-time, core loading technique. The advantage of loading the fuel-moderator elements as a unit-package is the greatly decreased probability of fuel element damage. In the unitized loading procedure, the 211 fuel elements did not have to be handled individually by remote-handling tools.

The original loading technique involved the placing of each fuel element in the core individually, using a 10-ft-long handling device to reach down into the core vessel through the center hole in the shutdown shield. In the event that a fuel element or a handling tool would be accidentally dropped down on the

partially or fully loaded core, extensive fuel element damage could result, especially to the shock-sensitive, ceramic coating inside the cladding. Such an occurrence would require the removal of all elements from the core. Since ceramic damage cannot be detected visually, all elements would be permeation tested to locate those with damaged cladding. Such a mishap would have resulted in a serious delay in the operation of S8ER.

Another difficulty encountered in remote loading is the lack of operator "feel" through the long, remote-handling tools. S8ER elements fit very closely in the core vessel and, in addition, some fuel elements have slightly irregular dimensions. Each element should be inserted in such a way as to minimize the force of contact so that damage is not caused by binding of one fuel element against its neighbors. Excessive contact between elements is also undesirable during power operation because of the increased possibility of hot spots.

The unitized loading method eliminated both of the problems discussed above. Since the individual fuel elements were actually loaded into the unit-package by hand, they could be positioned more precisely than could be done with remote-handling tools. The assembled unit-package offered excellent protection against damage to the fuel elements as the package was lowered into place in the reactor vessel.

The unitized core loading procedure consisted of the simultaneous assembly of 211 fuel elements and 150 poison rods into a safely subcritical unit-package, which was then installed into the reactor vessel. When the installation of the poisoned core was completed, the poison rods were removed, making the reactor capable of attaining criticality.

The poison rods used in this procedure were fabricated specially for use with the S8ER core. A mixture of samarium oxide and gadolinium oxide in a 45 to 45% ratio, the remaining 10% being other rare earths in the form of 48/200 mesh powder, was poured into 0.083-in.-OD stainless-steel tubing. The tubes were seal-welded at both ends, and a collar was silver-soldered at the upper end of each tube for support when the tube is mounted in the core.

b. Safeguards Considerations

The prevention of accidental criticality of the S8ER core package was the foremost consideration in the design of this core loading procedure. The

hazard of accidental criticality during the unitized loading operation was eliminated by installing 150 poison rods in a uniform geometrical arrangement throughout the unit-package of fuel elements to ensure that the package would be subcritical under all credible conditions.

The reactivity worth of the poison rods in the dry core (in air) was determined during the S8ER critical experiment. Results of several experiments showed no significant interaction (shadowing or flux depression) between adjacent poison rods or groups of poison rods. The total worth of 150 poison rods in the dry core was conservatively extrapolated from these experimental data and found to be $-\$11.5$.

The configuration in which the S8ER core is completely immersed in and surrounded by water was investigated experimentally in the water immersion test facility. It was found that 75 poison rods were required to suppress the excess reactivity, and maintain exact criticality ($k_{\text{eff}} = 1$) under conditions of complete core submersion. A second group of 75 poison rods was tested, and the results were nearly identical to the results obtained using the first group of poison rods. During the unitized loading operation both groups of poison rods (150 rods) were used. With this poison loading, the water-immersed core was estimated to be $\sim \$4.5$ subcritical.

It should be noted that complete submersion of the bare S8ER core ("bare" meaning without external beryllium reflectors) in a moderating liquid results in the most reactive configuration possible for the bare core, and water is the only such moderator which might conceivably be available for accidental submersion. Subcriticality of the poisoned core package was proven under conditions of total water immersion; thus, subcriticality was assured for all other possible situations which might credibly arise during this loading operation. A summary of calculated effective multiplication factors of the core package for various configurations which were attained during the unitized core loading operation is given in Table XXXIX.

A number of nuclear configurations which might be attained by the S8ER unitized core package, as a result of possible accidents during the loading operation, were examined. A situation was presumed in which the assembled unit was accidentally turned over, so that the poison rods would fall out of the core and the fuel elements remain in the package. The case of the bare,

TABLE XXXIX
REACTIVITY SUMMARY FOR CORE LOADING OPERATION

Configuration	k_{eff}	ρ (\$)
1. Initial (no fuel, no poison)	0	0
2. Bare, dry, poisoned core (150 poison rods)	0.74*	-33.5*
3. Dry, poisoned core passing through shutdown shield	0.88*	-15.5*
4. Dry, poisoned core in core vessel (all drums out)	0.81†	-24.6†
5. Dry, poisoned core in core vessel (two drums in)	0.87†	-17.5†
6. Dry, unpoisoned core in core vessel (two drums in)	0.955†	-5.9†
<u>Experimental Reactivity Data</u>		
Average worth of one poison rod (dry core)	= -7.7¢	
Average worth of one poison rod (water-immersed core)	= -6.0¢	
Totally water-immersed core, poisoned with 75 rods	Reactor just critical	

*Analytical

†Experimental, extrapolated

unpoisoned core falling on the floor was considered, as well as the case where the core package disassembled itself after falling on the floor. The configurations described above were considered credible; incredible accidents were also examined. It was experimentally demonstrated in the water immersion experiment that if the loaded poison core package was dropped in a pool of water, it would be subcritical. The results of the water immersion experiment were applied to the so-called "fat man" type of accident, caused by human proximity. Another incredible accident involved disassembly of the core package and removal of the poison rods, with the fuel elements becoming wedged during passage through the concrete shutdown field. The result of conservative analyses of all of these accidents indicated that the core package would remain subcritical by a safe margin.

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Figure 88. Core Assembly Fixtures on Buildup Table

c. Description

During the assembly of the core in a critical experimental facility, poison rods were loaded simultaneously with fuel elements to ensure sub-criticality of the unit-package. The upper and lower grid plates of the reactor were held in place by fixtures on a special mobile buildup table (Figure 88). The table was designed to support the core during buildup and transportation (Figure 89*). Individually suspended wheels and pneumatic tires on the table provided good resistance to shock. Foam rubber bumpers installed on the extremities protected the table for movement through doorways.

The fuel elements were individually hand-loaded through the space in the retaining ring into position between the two grid plates. Simultaneously, poison rods were inserted vertically downward through flow-holes in the upper grid plate. A collar on the upper end of each poison rod (Figure 90) enabled it to hang from the upper grid plate and extend down through the core (Figure 91).

Several channels of nuclear instrumentation were employed to monitor the core during the assembly operation. Two fission chambers and four compensated ion chambers were mounted on the table behind the buildup fixture, as shown in Figures 88 and 92. The signals from these detectors supplied scalars in the critical facility control room. A small Po-Be source (2×10^5 n/sec) was located below the lower grid plate, and remained until the core was completely assembled for continuous monitoring. At several points during the loading sequence, inverse multiplication measurements were taken to ensure subcriticality during succeeding loading steps. During these measurements, a stronger neutron source (Po-Be, $\sim 1 \times 10^7$ n/sec) was temporarily positioned beneath the core for the duration of the count.

The total core-loading (211 fuel-moderator elements, 18 internal reflector elements, and 150 poison rods) required a two-day assembly period. Two individual inventories were maintained during assembly, one for fuel-moderator elements, the other for poison rods. These inventories were checked by two persons after each loading. After the upper grid plate was lowered into position, a poison retaining plate (Figure 93) was installed over the top of the

*Figures 89, 90, 91, 93, and 94 were taken during core-loading demonstrations in which mock fuel elements were used.

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Figure 89. Assembled Core (Mock Elements) on Mobile Buildup Table

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Figure 90. Poison Rods and Mock Fuel Element

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Figure 91. Core During Assembly of Mock Elements

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Figure 92. Nuclear Instrumentation on Buildup Table

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Figure 93. Poison Rod Retaining Plate

poison rods, and this plate was bolted to the upper grid plate. The retaining plate securely clamped the poison rods in place, preventing inadvertent poison rod removal from the core. Assembly dimensions were taken and found to be within tolerance. Six lifting cables (1/16-in. stainless steel — any one cable could support the entire core package) were threaded through coolant flow passages and holes in the grid plate, and secured by modified tubing fittings (Figure 93) attached to the poison rod retaining plate. These clamp fittings and the cables secured the entire core package together (Figure 94). The cables were later used to lower the package into the core. The assembly was then inspected and covered with polyethylene in preparation for transport to the S8ER facility.

Prior to transfer of the core, the S8ER facility was thoroughly cleaned and a clean room installed over the containment vessel. The clean room was equipped with a filtered air supply, entry air lock, communications to the control room, and an audible signal repeater ("poppy") connected to a nuclear flux monitoring channel. A work platform was installed in the containment vessel approximately three feet below the top flange. Three safety flood lights suspended from the work platform provided ample lighting for core installation.

Early in the day on which the core was installed in the reactor vessel, the route between the critical experiment facility and the S8ER facility was cleaned and the area roped off and barricaded by security police. With a man stationed at each corner of the mobile table containing the assembled core package, the table was rolled to the S8ER facility as shown in Figure 95.

After the core was transported to S8ER, dimensions were rechecked (Figure 96), and the core was lifted from the buildup table by the one-ton chainfall (with dynamometer attached), and placed on a work stand in the containment vessel. The polyethylene cover was then removed and the lifting cables extended full-length and connected to the chainfall. During these and subsequent operations, a qualified operating crew was on duty and the nuclear instrumentation was under constant observation. The core then was placed on the reactor tie-rod extensions, which served as guides during installation, and lowered into the reactor vessel. These extensions were screwed into the S8ER tie-rods and extended well above the shutdown shield. Seating of the core was confirmed

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Figure 94. Core Package With Mock Elements

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Figure 95. Transporting Core Assembly to Test Facility

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Figure 96. Dimension Check of Actual Core

by dimensions obtained using a depth micrometer. The tie-rod extensions were then removed and the grid-plate holddown bolts installed. The lock plates were then crimped to the bolts, firmly securing the core in place. This completed the installation phase.

Upon receipt of approval for nuclear operations, the cable clamp fittings were disassembled, the six lifting cables removed, and the poison-rod retaining plate was withdrawn from the reactor. Prior to poison removal, all reactor instrumentation was checked for readiness, and then two control elements were rotated to their most reactive position. This gave a scram capability of approximately seven dollars during withdrawal of poison. The poison rods were removed individually in previously established increments. Neutron multiplication was observed after each increment. The safety of removing the next increment was then verified by evaluating a plot of inverse count rate vs number of poison rods. Removal of poison required four hours.

d. Conclusion

The successful loading of S8ER demonstrated that the unitized technique prevented mechanical shocking of the fuel elements, was operationally convenient, could be performed expeditiously, and was a safe method. In addition to its usefulness at this laboratory, the technique may find application elsewhere in the storage and transport of nuclear reactors.

4. System Assembly and Checkout

Test S8ER-006, Final Seal Welds and Heat Transfer Checkout was started upon completion of dry critical testing. The following portions of the test were performed: seal welding of the shutdown shield center plug to the shutdown shield, installation and seal welding of the primary heat transfer piping from the reactor to the piping penetrating the containment vessel walls from the primary equipment vault, installation and seal welding of all shutdown shield air cooling and vent piping, and installation of the primary system rupture disc and associated NaK overflow thimble (see Figure 97).

The welded design of the system rupture disc is an innovation. The normal state-of-the-art rupture disc is sealed by bolting two flanges together forming a pressure seal. In S8ER, the possibility of this pressure seal leaking NaK had to be eliminated due to the critical location of the rupture disc on the shutdown shield surface (Figure 97). Consequently, a rupture disc assembly was

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Figure 97. Equipment Above Shutdown Shield

designed (Figures 98 and 99). This assembly consists of two Inconel rupture discs, each disc mated with an Inconel vacuum support, designed to burst at 100 psig $\pm$ 10% up to temperatures of 1000°F. Operating NaK temperature at the rupture-assembly tee in the reactor inlet line is 1100°F. A special piping arrangement was used to minimize convective flow at the first rupture disc and increase the length of the piping connection. In addition, a portion of the connecting piping was not insulated to allow for dissipation of heat. By these techniques maximum temperature at the disc is limited to 1000°F. The assembly as installed in the system consists of one rupture disc welded into the assembly (this forms a positive NaK seal) and a second rupture disc which is held in place by the bolted flange. A NaK leak detector has been installed in the annulus between the two discs. The detector has been so wired as to initiate a warning signal in the control room in the event of failure of the welded rupture disc.

The NaK overflow line (Figure 97) is directly attached to the discharge side of the second rupture disc. In the event of failure of both rupture discs NaK will be conveyed through the rupture and down to the bottom of the containment vessel. This arrangement of piping precludes damage to any of the vital scram and reflector drive assemblies located on the top surface of the shutdown shield.

Permanent equipment for controlling the temperature of the containment vessel wall was installed. The system provides a means of being able to heat or cool the wall to maintain the temperature sufficiently above the nil ductility transition temperature during the predicted life of the vessel. This operational characteristic assures ductile behavior of the vessel.

The system (Figure 100) consists of a separate water loop which contains a 100,000-Btu/hr, gas-fired heater in parallel with a shell-and-tube heat exchanger, through which heat can be transferred to the plant cooling tower. It has been designed to permit preheating the containment vessel to 250°F, and to reject heat during reactor operations.

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Figure 98. Rupture Disc Components

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Figure 99. Welded Rupture Disc Assembly

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Figure 100. Containment Vessel Temperature Control System Progress Diagram

B. SNAP 8 GROUND PROTOTYPE (S8DS-1)

1. Introduction and Summary

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 a 35-kwe flight power conversion system. The S8DS-1 reactor will be essentially a flight reactor and will be modified only to the extent required for safety and feasibility of ground testing.

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 purpose of the S8DS-1 project is to determine the steady-state and transient operating characteristics of a SNAP 8 flight reactor; to determine reactor control requirements for automatic reactor startup and for system stability during a PCS startup; to demonstrate satisfactory reactor operation at approximately 450 kwt, 20 automatic startups, 90 days endurance, and a 10,000-hr operational capability of a complete 35-kwe EGS; and to determine the nuclear and thermal operating characteristics of SNAP 8 shielding materials in a SNAP 8 irradiation environment.

Since definitive requirements for a SNAP 8 shield are not expected to be determined in time to be used in the S8DS test program, the shielding development has been directed toward determining the nuclear operating characteristics of SNAP 8 shielding materials. S8DRM-1's 4-in.-diameter reflector drums were found to furnish insufficient reactivity for control of the S8DS reactor. The drum diameter was increased to 4-1/2 in. to assure ample drum reactivity worth. Direct-drive actuators were selected as the reference design drives.

The vacuum requirement for the S8DS test environment was increased to 10^{-5} Torr. This increase is necessary to prevent oxidation of the bearing materials.

An evaluation of the effects of irradiation on the nil ductility transition temperature indicated a requirement for pit vessel temperatures as high as 390°F. This high temperature plus the need of other modifications resulted in a requirement for replacing the pit vessel.

New vault equipment layouts were prepared to accommodate revised plans for installation of the reactor in the north reactor pit.

2. System Analysis

a. Drum Worth Studies

Four-and-a-half-in.-radius control drums were adopted for the S8DS reactor design after extensive calculations involving transport diffusion theory and scaling techniques. This decision was based largely on the hand-scaling techniques as these seemed most credible and the larger control drums would assure sufficient control.

The diffusion theory calculations on the 4-in.-radius control drums were performed using the two-dimensional, three-group CURE program in cylindrical geometry. The mockups and input constants were calculated on the same basis as the constants used on previous calculations of the S8ER control drum worths which had given close agreement with experimental results. The calculated S8ER drum worth according to CURE was 2.7% $\Delta k/k$ per control drum. When correlated with S8ER experimental values, the result would be about 2.87% $\Delta k/k$.

A similar set of calculations was run using the 2DXY two-dimensional transport theory program in cartesian geometry. Four neutron groups were used. The S8DS control drum was calculated at 3.1% $\Delta k/k$. When corrected for comparisons with similar 2DXY calculations of the S8ER, the expected drum worth would be 2.95% $\Delta k/k$. However, the scaling technique which scaled the S8ER experimental drum worths to the S8DS 4-in. drums gave drum worths of 2.06% $\Delta k/k$. This method utilized experimental curves of reflector worth as a function of radius. The incremental beryllium volumes removed by rotation of the drums were calculated and multiplied by their reactivity worth to find the loss of reactivity due to rotation of the drums. Additional calculations were performed with drums of 4.25-, 4.40-, and 4.5-in.-radius. The 4.5-in.-radius drum was estimated to give a drum worth of about 2.4% $\Delta k/k$ which would be

adequate for safe control. Calculations were also made to estimate the range of errors involved with this technique. A curve of drum worth as a function of drum radius was then plotted and is presented as Figure 101.

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Figure 101. Effect of Drum Radius on S8DS Drum Worth

The decision to base the S8DS drum design largely on the basis of the scaled results as contrasted to the computing machine calculations was due to two reasons. The first was that the smaller volume of beryllium rotated with the four-in.-radius drums would be expected to give lower drum worths than the larger S8ER drums, especially since considerably more beryllium was left between the moveable portion of the drums and the core. This consideration made the scaling technique results seem more credible. Second, it was also found that changing the drum radius from 4.0 to 4.5 in. would not significantly reduce the structural strength of

the stationary portion of the reflector. Since insufficient control worth would present a need for extensively redesigning the control procedure, it was decided to use 4.5-in.-radius control drums for the S8DS.

b. Thermal Studies of S8DS Containment Vessel

Prolonged exposure to fast neutrons (neutrons of >1 Mev energy) hardens and strengthens steel but also decreases ductility. It has long been known that operating a normally ductile structure at a temperature near or below its nil ductility transition temperature can lead to brittle failure of the material. Therefore, the service life of a material subjected to irradiation (decreasing ductility) is a function of its operating temperature which can compensate for the loss in ductility, thereby preventing brittle failure of the material.

The operation of the containment vessel surrounding SNAP 8 reactors at a safe temperature to prevent such a brittle failure then becomes an important standard procedure for the safety of personnel and equipment. The use of

different structural materials in the S8DS containment vessel dictates the qualification of a safe temperature range for each material. The lower limit must be above the nil ductility transition temperature of the steel liner due to thermal stresses which would be produced if hot NaK were spilled in the containment vessel. The upper limit should be below that temperature which will produce damaging thermal stress in the liner due to differences in thermal coefficients of expansion between steel and the surrounding concrete. Another limitation to the upper limit is the designed concrete temperature.

To look into the various aspects of the S8DS containment vessel embrittlement problem, calculations of the expected temperatures of the containment vessel were made using a modified version of the 30-node transient heat generation code (THGC). Results of this investigation are mapped out in the following figures for the various operating conditions. Boundary conditions for the thermal model chosen were taken from studies on the S8DS reflector. A linear temperature ramp in 10 hr is the closest approximation of the actual "startup" heating of the beryllium reflectors before steady-state conditions. Cooling water and nitrogen temperatures are the designed values from their respective heat exchangers. Earth temperature was assumed a constant 60°F.

A symmetrical section of the reactor pit wall located at the core midplane was selected (Figure 102). Starting with 60°F ambient temperatures around the 600°F core, the transient temperatures were predicted for a zero-kilowatt reactor power with the effect of cooling coils. Temperatures appear to approach steady-state values after 15 hr, and these are indicated in Figure 103. The reflector temperature increases from an initial value of 100°F to a maximum of 215°F in 10 hr, and beyond this time it remains constant at 215°F. The cooling water follows a linear ramp from 60 to 95°F in one hour.

Another study for zero-kilowatt reactor power at 1200°F NaK in the core was simulated. Figure 104 indicates the expected temperatures after 15-hr operation. The reflector follows a linear ramp from 100 to 800°F in one hour.

Figure 105 indicates the expected containment temperatures at the core midplane region after 15-hr operation of the reactor at 450 kw and 1200°F NaK core temperature. Heat generation (as ramp function dependent on time) in the steel, therman, concrete, and sand are included in this case. The cooling

Figure 102. S8DS Thermal Model at Core Plane Region of Pit Wall

Figure 104. S8DS Reactor Pit, Core Midplane Region, Zero kw, 1200° F NaK

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Figure 105. S8DS Reactor Pit, Core Midplane Region, 450 kw, 1200° F NaK (with cooling water)

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Figure 106. S8DS Reactor Pit, Core Midplane Region, 600 kw, 1200°F NaK (with cooling water)

ALL TEMPERATURES IN °F

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Figure 107. S8DS Reactor Pit, Core Midplane Region, 600 kw, 1200°F NaK (no-cooling case)

ALL TEMPERATURES IN °F

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water is at a constant temperature of 95°F, after attaining that temperature from an initial value of 60°F in one hour. The maximum steel liner temperature to be expected is $\approx 115^\circ\text{F}$.

With the same thermal model, the expected containment vessel temperatures were investigated for 600-kw reactor power and 1200°F NaK core temperature, with and without the effect of cooling coils. In both cases the reflector temperature increases from an initial value of 100°F to a maximum of 1160°F in 10 hr and beyond this time it remains at 1160°F. The cooling water temperature ramp is 60 to 95°F in one hour for the cooling case. For the no-cooling case, the space occupied by the water coil was filled with thermon, and the reflector temperature was optimistically assumed to be the same as the case with cooling described above. Figures 106 and 107 are the calculated steady-state temperatures for the cooling and no-cooling cases, respectively.

The containment vessel, as a whole, was investigated and the thermal model selected for solution in the 30-node THGC is shown in Figure 108. The temperature ramp of the cooling water is the same as in previous cases.

Figure 109 indicates the steady-state temperatures at 450-kw reactor power and 1200°F NaK. The reflector and shield temperature ramp from 100 to 1070°F and 60 to 550°F in one hour, respectively. Maximum liner temperature expected is $\approx 105^\circ\text{F}$. The pit cover plate temperature is $\approx 500^\circ\text{F}$.

Figure 110 indicates the steady-state temperatures at 600-kw reactor power and 1200°F NaK. It can be noted that the 4-in. cover plate is running much hotter than the upper portion of the steel liner; therefore, a more detailed calculation of the temperature map at the upper pit region was initiated. Figure 111 indicates the result of such a study.

To reduce the temperature of the 4-in. cover plate of the reactor pit, the effect of the presence of cooling fins parallel to the flow of nitrogen was investigated. Figure 112 indicates the result.

Again using the thermal model of Figure 108, a case was simulated where the shield was not installed. This resulted in a higher temperature of the pit floor as shown in Figure 113.

So far, the containment vessel has been considered during reactor operation without any incident. From a safety standpoint, one must consider the transient

Figure 108. S8DS Thermal Model of Reactor Pit

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Figure 109. S8DS Total Reactor Pit, Steady-State Temperatures, 450 kw, 1200°F NaK (with cooling water)

ALL TEMPERATURES IN $^{\circ}\text{F}$

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Figure 110. S8DS Total Reactor Pit, Steady-State Temperatures, 600 kw, 1200° F NaK (with cooling water)

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Figure 111. Temperature Profile of S8DS Pit Upper Region, 600 kw, 1200° F NaK (with cooling water)

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ALL TEMPERATURES IN °F
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Figure 112. S8DS Reactor Pit Cover With Fins, 600 kw, 1200° F NaK (with cooling water)

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ALL TEMPERATURES IN °F

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Figure 113. S8DS Total Reactor Pit (without shield), 600 kw, 1200° F NaK

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EARTH 60°F

ALL TEMPERATURES IN °F

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temperatures throughout as a result of a maximum credible accident (MCA). An MCA was assumed to be the accident spill of hot NaK into the containment vessel following a failure or break in the NaK pipe or core vessel. For this reason, the thermal model of Figure 108 was analyzed for such a simulated condition.

Following one month's operation at steady-state conditions (450- and 600-kwt power levels), an MCA as described above was assumed to occur resulting in the formation of a NaK pool (536 lb NaK inventory at 1460°F) at the bottom of the pit. Figures 114 and 115 indicate the containment temperatures following an MCA at a 450-kw or 600-kw power level. In either case, the maximum steel floor temperature expected is ~750°F.

c. Radiation Aspects of S8DS Vacuum System Redesign

Tentative redesign plans for the vacuum system in the S8DS Test Facility (Building 059) call for a relocation of the vacuum equipment room, presently in the high bay area, to an area immediately adjacent to the test vault. A brief study was made of the anticipated radiation levels in the proposed vacuum equipment room relocation. Two sources of radiation were considered: (1) gamma rays due to the decay of activated NaK in the test vault, and (2) fission product gamma rays from the shutdown reactor after its removal from the reactor pit to the floor of the test vault. Neutrons and gamma rays which escape from the operating reactor and travel through the vacuum line must make a minimum of three scatters to enter the vacuum equipment room. This radiation was judged to be negligible with respect to the other two sources.

The gamma dose rates at the surface of the vacuum equipment room wall are presented in Figure 116 as a function of concrete thickness between the test vault and this room. The data correspond to a 600-kwt reactor power level and the following set of assumptions:

- 1) Shutdown reactor located on test vault floor on a line joining reactor pit centers.
- 2) NaK concentrated at a point 10 ft from wall separating test vault and vacuum equipment room.
- 3) No self-attenuation of NaK in PCS boiler.

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Figure 114. S8DS Reactor Pit, Shutdown (due to maximum credible accident) After One Month Operation at 450 kw, 1200°F NaK

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Figure 115. S8DS Reactor Pit, Shutdown (due to maximum credible accident) After One Month Operation @ 600 kw, 1200°F NaK

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Figure 116. Radiation Levels in Vacuum Equipment Room

d. Scattered Neutron Contribution in S8DS Reactor Pit

It is presently planned to install an S8DRM-1 type shield in the S8DS facility to study the effects of long-term irradiation on the shield. One item of major concern with such an installation is the possibility that the neutrons scattered from the walls of the reactor pit will result in excessive heating of the shield. This problem was investigated and the heating due to scattered neutrons appears to be negligible when compared with the heating rates due to neutrons impinging on the shield directly from the reactor core.

For the purposes of this study, the reactor, shield, and pit configuration shown in Figure 117 was assumed. The shield vertex is assumed to be positioned 1 ft above the bottom of the reactor pit. In addition to neutrons and gamma rays entering the shield in a direct path from the core, there is a scattered contribution from the steel pit liner and the shield safety vessel. Another, though minor, contribution arrives from the ceiling and floor of the pit. Scattered gamma flux levels were estimated to be negligible compared to the direct component.

The results of the calculations are shown in graphic form. The nuclear heating in the shield due to direct radiation from the reactor is shown in Figure 118 while Figures 119 and 120 show the effect of scattered neutrons.

The decision to use a one-region Li^7H shield placed about 1 ft away from the reactor has resulted in relatively low heating rates which will not endanger the shield but which will provide significant data on the physical properties of the shield material.

3. Reactor Vessel and Support Structure

The conceptual design of the S8DS reactor vessel is based on the S8DRM design. A few modifications to the S8DRM design which were considered, qualitatively, to improve the fabricability of the vessel are shown in Figure 121. The core and lower flow baffle plate are shown supported on a ring ledge in the lower plenum. The upper grid plate is supported by the internal reflectors and the whole core is held down by a lip on the inside of the vessel upper plenum head.

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Figure 117. S8DS-1 Reactor Pit Geometry

Figure 118. Nuclear Heating in S8DS-1 Shield Due to Direct Radiation From Reactor, 620 kw

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Figure 119. Nuclear Heating in S8DS-1 Shield Due to Fast Neutron Scattering From Pit Liner, 620 kw

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Figure 120. Nuclear Heating in S8DS-1 Shield Due to Fast Neutron Scattering From Shield Safety Vessel, 620 kw

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Figure 121. S8DS Reactor Vessel

4. Neutron Shield

The shield design will vary with the geometry of the spacecraft and the requirements of the particular mission. Since the definitive requirements for a SNAP 8 shield are not expected to be determined in time to be used in the S8DS test program, the shielding development has been directed toward determining the nuclear and operating characteristics of SNAP 8 shielding material.

The successful completion of the design and fabrication phases of a shield for the S8DRM-1 system serves as a firm base for extending the SNAP 8 shielding program to an irradiation test in the S8DS-1 facility. It is anticipated that this test will provide extremely valuable information on the behavior of Li^7H under SNAP 8 conditions.

Since the operation of a LiH shield under SNAP 8 conditions represents a significant extrapolation of existing technology, it is planned to contain the shield within a separate safety vessel. This will provide a very high degree of assurance that the shield can in no way adversely affect reactor operation. Thermocouples, pneumatic connections, and neutron detectors will be employed to monitor shield performance.

A thermal analysis was performed to determine the operating temperatures of an S8DRM-1 type shield contained in a safety vessel under S8DS-1 conditions of 450-kwt reactor power level. The S8DRM-1 configuration was assumed in order to utilize the spare shield casing which is available from the S8DRM-1 program. In the analysis, helium gas was assumed in the annulus between the shield and safety vessel. The contribution of the gas to the transfer of heat across the annulus is shown in Figure 122. The safety vessel used in the analysis was a cylindrical type leading to a peak temperature of 968°F. Analysis is proceeding on the assumption of a close fitting conical safety vessel to take advantage of the better heat transfer.

5. Reflectors and Drives

Nuclear experiments on the S8ER nuclear test system control drum calibration, as well as more refined analysis of the effective control drum reactivity worth of the S8DRM-1 size drum, have necessitated an increase in the control drum size from a 4- to a 4-1/2-in.-radius (see Section VII-B-2, System Analysis). The scram drive mechanism has been "scaled-up" to accommodate the larger drum and a scram mechanism has been designed for the

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Figure 122. Comparison of Heat Transfer Modes Across Helium-Filled Annulus

direct-drive type actuators. The direct-drive scram mechanism and the geared stepper-motor scram mechanism design are shown in Figure 123.

The direct-drive actuator is considered the reference design for the SNAP 8 Flight System. Acceleration of the development program for these actuators makes them "contenders" for the S8DS system.

6. Heat Transfer System and Instrumentation

Changes in the PCS and its test support equipment (principally the replacement of the flight type radiator with a workhorse radiator) necessitated the preparation of new in-vault equipment layouts. The reactor was moved to the north reactor pit and the reactor test support equipment was moved from the south to the north half of the test vault. A preliminary layout of the new arrangement is shown in Figure 124. The primary NaK heat transfer equipment is a compact package which is installed on a pallet to allow its being handled as a unit. It is sized to permit its passage through the large circular vault access port. The test support equipment design was nearly complete at the time that the PCS changes were made and much of the original design will be used with the new layout.

The heat transfer package was lowered from 8 ft to 18 in. above the vault floor to reduce the NaK pressure head on the reactor vessel. Figure 125 is an isometric drawing of the new heat transfer loop arrangement.

7. Facility

Construction was completed and acceptance testing was initiated by the architecture and engineering contractor. In general, these tests are to be conducted to establish that the system meets the design criteria and to determine the maximum system capabilities.

The reactor pit liners were evaluated with respect to the effects of irradiation on the nil ductility transition temperature. It was determined that the initial nil ductility transition temperature for these reactor pit liners could be as high as 160° F in the heat-affected zones and could increase 170° F during a 10,000-hr test at 600° F. Since the vessel should be operated at nil ductility transition temperature +60° F, the required operating temperature after 10,000 hr could be as high as 390° F.

PLUG IN SCRAM MECHANISM FOR
LINEAR ACTUATOR

PLUG IN SCRAM MECHANISM FOR
 $\frac{1}{2}^\circ$ STEPPER ACTUATOR

PLUG IN SCRAM MECHANISM FOR
GEARED STEPPER MOTOR

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Figure 123. S8DS Scram Mechanisms

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Figure 124. S8DS Heat Transfer System Layout

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Figure 125. S8DS Heat Transfer Loop

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This vessel was originally installed as a reactor pit liner in the test vault and was later converted to a vacuum vessel. Principal contributors to the assumption of this high initial nil ductility transition temperature were the lack of information about the material and procedures used in fabricating the vessel plus the fact that it was never stress relieved. The high operating temperature requirement plus the fact that further modifications would be required before the pit could be used resulted in a requirement for replacing the pit vessel.

Development tests of bearing materials for the S8DS reactor indicate that vacuums of 10^{-5} Torr or better are required as the S8DS environment. As the existing system is only capable of achieving $\sim 10^{-3}$ Torr, it is now required that additional vacuum equipment be added to the facility.

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

1. Introduction and Summary

The SNAP Developmental Reactor Mockup-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.

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

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.

During the present quarter, the assembly of the S8DRM-1 continued. Sufficient components were assembled to start fabrication of the wiring harness. Figure 126 shows the status of the assembly at the end of this quarter.

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Figure 126. S8DRM-1 Reactor and Shield

During the past quarters the first main objective has been completed. The second objective is nearly complete as is the third major objective of the program.

2. 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 support lugs attached to the bottom head.

During this reporting period, fabrication of the following reactor core vessel components was completed: one reactor vessel, 4 upper plenum head assemblies, 215 mockup fuel elements, 18 mockup internal reflectors, and one upper grid plate. Fabrication of the second reactor vessel was started and is ~30% complete.

The 25 natural uranium - zirconium hydride fuel elements will be manufactured in a manner similar 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 number of S8DRM-1 natural uranium fuel elements was increased from 6 to 25. This increase was due to a reevaluation of the fuel element qualification testing program and the S8DRM-1 project. This evaluation incorporated the S8DRM-1 fuel elements into both programs. The fuel elements will undergo some qualification testing; they then will be installed in the S8DRM-1 core for vibration, shock, and startup reliability testing. After this the elements will undergo further qualification testing culminating in a 10,000-hr endurance test.

Thirteen of the fuel elements will have holddown devices while 12 will be assembled without holddown devices.

3. Reflector and Drive

The reflector and drive system consists of a beryllium reflector 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 which provides that when the drums are in their full-in position, the core is surrounded by a nominal 3-in.-thick beryllium cylinder.

The S8DRM-1 reflector and drive system is the first flight design for the SNAP 8 reactor. The system must provide accurate positioning of the control drums to increase or decrease effective reflectivity consistent with the long-term control and automatic startup requirements. The system also incorporates irreversible shutdown by reflector-control system ejection. The fabrication and assembly of the system* was continued with one change in concept. The geared electric motor drive originally planned for the three coarse control drums was replaced by a clock type torsional spring drive due to a reevaluation of the safety criteria. This redesign was completed and all detail parts released for fabrication. This drive mechanism will be installed on the control system before shock and vibration testing begins.

All S8DRM-1 reflector and drive system detail parts were completed during this period except: (a) the reentry band, (b) the automatic coolant-temperature-drop reflector-ejection actuator, and (c) the destruct charge.

The reentry band fabrication is 90% complete.

There were machining problems encountered in fabricating the molybdenum parts of the coolant-temperature-drop reflector-ejection actuator. However these have been worked out and fabrication is proceeding satisfactorily.

The complete reflector and drive system was assembled during this period with the exception of the above three items and the wiring harness is now being fitted to the assembly (Figure 126).

4. Instrumentation

The S8DRM-1 instrumentation consists of a startup programmer, a long-term controller, drum position sensors, and relay box.

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, October - December 1962," NAA-SR-8092 (SRD), May 20, 1963, pp 166, 174-176.

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 drum position sensors are small analog-type rotary position transmitters coupled to a demodulator which is physically separated from the position sensor. The relay box contains the power switching relays receiving input signals from the startup programmer and other command circuits and then closing the power relays to various squibs and other actuating devices.

The programmer design and fabrication was completed and is now in final stages of quality assurance testing. Design of the position sensor and relay box was started. Final checkout and testing of the controller was completed.

5. Actuators

a. Long-Term Actuator

The long-term actuators used on the S8DRM-1 are indirect-drive actuators using a slotted, magnetless rotor to produce bidirectional discrete steps. The rotary motion is reduced by step-down gears to produce the required incremental reflector control drum step size, in order to maintain the desired reactor temperature. During the present quarter, three long-term control drum actuators have been assembled, inspected, and delivered for acceptance testing.

The long-term actuators used for S8DRM-1 are insulated with SAP II, an insulation system developed at Atomics International. Prior to final assembly, each stator assembly and brake solenoid assembly was checked out for 200 hr at 1100°F at a pressure of 1×10^{-5} mm Hg (Figure 127). Data on the coils was automatically recorded during this testing in order to evaluate the insulation resistance and coil resistance (Figure 128).

b. Startup Actuators

In view of the reevaluation of the safety criteria requirements, the requirement for the startup actuators has been replaced by spring-type drive similar to the SNAP 10A/2 system.

c. Cable Harness

All components necessary for the fabrication of the S8DRM-1 reflector and shield cable harnesses, the S8DRM-1 reflector cable harness bracketry, and the S8DRM-1 shield cable harness have been delivered. The fabrication of the shield cable harness has been started.

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Figure 127. Stator and Brake
Solenoid Test Assembly

Figure 128. Stator and Brake
Solenoid Test Instrumentation

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6. 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 kwt. In order 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 neutron shield to a mating ring attached to the bottom of the neutron shield vessel.

The casting of LiH into the shield was successfully completed.* Upon removal of the shield from the casting hardware, a dark deposit was noted on the large end of the conical section. This condition is evident in Figure 70 and is characteristic of a minute leak of LiH during the casting process. Barely evident in the photograph is the vertical weld seam in the cone that meets the circumferential seam at the location of the deposit. The cable harness attachment nut was removed and the suspect area was polished, etched, and examined by means of fluorescent dye. A minute pinhole leak in the vertical weld was finally detected. The weld repair was routine.

The shield was returned to the vendor for installation of the Min-K thermal insulation and upper structure assembly. Following finish machining of various mounting surfaces and AI inspection at the vendor's plant, the shield was returned to AI for preliminary assembly work and fitup of the electrical harnesses. All S8DRM-1 shield fabrication work is now completed except for application of the chrome oxide emissivity coating which will be accomplished after the preliminary assembly work.

7. Testing

The objective of this project is to perform non-nuclear assembly, environmental and operational tests to qualify the nuclear system as an assembly capable of startup, proper operation, and endurance when subjected to the environmental conditions it will experience in a flight system. The purpose of testing the S8DRM-1 is to disclose weaknesses and malfunctions which are not

*Refer to Section VI-F-2 for fabrication details.

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Figure 129. High-Speed Photography Apparatus for
Observing Control Drum Motion

evident during testing of individual components and during acceptance testing of the nuclear system.

During the present quarter, the assembly has proceeded to fitting of the wiring harness. For this purpose the reflector halves have been completed with bearings, drums, and all miscellaneous hardware available, using a tooling band and tooling vessel. This assembly has been integrated with the shield and mockup or real devices installed so that all electrical connections can be made during the harness fitting.

Three actuators have been received and are undergoing test. The controller (not flight-type) has been received, tested, and is being used in the actuator test.

The checking of the bellows was completed except for the torsional deflection test. Equipment difficulties caused adoption of a different procedure, i. e., to use high speed photography (Figure 129) to observe the drum motion when driven by an actuator. This permitted evaluation of the total drive system backlash and torsional deflection, which appears to be small and well within tolerance. However, as S8DRM-1 actuators were not available, a SNAP 10A actuator with adapter was used. This will be repeated for all drums when assembly is complete.

The medium vacuum system was performance tested and was found to require some improvements. Minor modifications were made to the system and its instrumentation was returned to the supplier for rework.

Fabrication of heat simulators and test fixtures for testing the complete nuclear system in vacuum is 85% complete. Design of heat simulators and test fixtures for endurance testing of the nuclear system was started.

A study is being made of methods of upgrading the vacuum system for long-term endurance testing.

D. SNAP 8 FLIGHT SYSTEM (S8FS)

1. Introduction

The S8FS experiment is the proof test for the flight prototype 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

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Figure 130. S8FS Test Facility Floor Plan and Elevations

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will be integrated with the power conversion system (PCS) and assembled in the final spacecraft arrangement. Endurance tests will be performed at power levels up to the full-rated 35-kwe level.

The objectives of the S8FS test are: (a) to prove the adequacy of the SNAP 8 design and hardware to meet the overall program objectives by demonstrating rated performance, control stability, and system reliability of the complete EGS up to 10,000 hr at 35 kwe in a simulated space environment under changing loads; and (b) to demonstrate the reliability of the startup sequence by performing three consecutive successful startups.

2. System Description

The NS will be composed of flight components which have been modified to accommodate the scram and restart capabilities required for startup demonstrations and 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.

3. Test Facilities

The completed design of the Flight Prototype Test Facility (FPTF) was evaluated against revised EGS test requirements established last quarter. These new test requirements reflect the recent reorientation of the PCS development program. The EGS radiators or radiator condensers have been removed from the system; therefore, space heat sink simulation or large volume test vaults are no longer required in the FPTF. Equipment procurement and facility construction was halted on the existing facility design and a new overall FPTF concept has been evolved.

The facility will still be located at the North American Field Laboratory, Santa Susana, and will make maximum use of existing site grading and excavation. The new FPTF (Figure 130) will consist of below-grade test operation areas and above-grade facility service areas. The above-grade portion of the building will include a high-bay area equipped with a 30-ton bridge crane, a control room, office area, instrument repair room, and related utility and service areas.

Test operations will be conducted in shielded, below-grade spaces. The actual S8FS test will be located in a concrete vault, nominally 30 ft wide by

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Figure 131. S8FS Environmental Test Equipment Layout

60 ft long by 40 ft high. The vault is sized to accommodate the EGS in either vertical or horizontal test orientation. The test vault will be lined with aluminum plate for containment of radioactive gases in case of an operational incident and the walls will be constructed of special neutron absorbing materials to reduce activation. The FPTF will have the capability of permitting installation of a standby S8FS test within 30 days after a failure of the initial test system. This capability is achieved by providing a second test vault and sufficient related service areas. The two test vaults will be separated by a shield wall which will be designed to be removed or relocated without damage to the permanent structure, thereby permitting adjustment of final vault length in the event of future changes to the test system configuration. Remote viewing systems will be provided for each test vault.

The FPTF will include a disassembly area that is immediately above, and provides full coverage to, both test vaults (Figure 131). All weight handling and remote handling equipment will be located within the disassembly area for servicing this area and the test vaults. The disassembly area will be utilized during test installation and test dismantling. Final stages of test dismantling in the vault below will be accomplished remotely using a rectilinear manipulator, 20-ton bridge crane, and TV viewing equipment. The test vaults will be separated from the disassembly area by removable shield plugs which will be designed to provide a full width opening to the vaults when removed.

A transfer lock will be provided above the disassembly area at the high-bay floor level. This transfer lock will provide shielded space and equipment for remote transfer of tools or equipment to and from the disassembly area and for detailed disassembly, examination, and packaging of parts of the test prototype. The transfer lock will be equipped with a bridge crane, rectilinear manipulator, slave manipulators, and a viewing window.

Mechanical and electrical equipment for the test operations and test support systems will be located in below-grade service areas immediately adjacent to the test vault — disassembly area. These service areas will include a vacuum equipment room which will contain the primary environmental vacuum pumps; a pump room which will contain standby vacuum pumps and other auxiliary test support systems equipment; and a vault mechanical and electrical equipment area which will contain heating and ventilating equipment, controlled atmosphere equipment, and electrical distribution equipment.

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The prototype test system will be located in the test vault inside an aluminum vacuum chamber. This chamber will be connected to vacuum pumps located in the adjacent vacuum equipment room. The chamber will provide a controlled environment of 10^{-5} -mm Hg vacuum and 100°F wall temperature in order to simulate space conditions during the test program.

Figure 131 depicts the anticipated test environment and support equipment installed for operation in this facility. The entire facility will incorporate features that increase the flexibility of the facility to accommodate possible changes in test system configuration.

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