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SNAP 8 PROGRESS REPORT

FEBRUARY-APRIL 1965

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SNAP 8 PROGRESS REPORT

FEBRUARY-APRIL 1965

(Title Unclassified)

By

C. E. JOHNSON

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I. PROGRAM OBJECTIVES

The objectives of the SNAP 8 reactor program are to develop and test the 600-kwt, 1300°F SNAP 8 reactor, and to continue the development of technology to provide a reactor capability at thermal power levels up to approximately 1 Mwt for both manned and unmanned space electric power applications. The SNAP 8 reactor program is being performed for the Atomic Energy Commission (AEC). The performance objectives of the reference design SNAP 8 reactor are summarized in Table 1.

TABLE 1
SNAP 8 REACTOR PERFORMANCE OBJECTIVES

Thermal Power	600 kwt
NaK Outlet Temperature	1300°F
Reactor Operating Period	10,000 hours
Reliability	
Instrument-Rated Reactor	>96%
Manrated Reactor	>99%

II. PROGRESS SUMMARY

A. REFERENCE SYSTEM DESIGN

Conceptual design studies of a 4 π shielded reactor for manned applications continued. These studies included consideration of extending the core life to 20,000 hours.

Analysis of the 4 π shielded reactor indicated that either beryllium or beryllium oxide may be used as a reflector material. Use of beryllium oxide would permit higher design temperatures and would allow the use of poison-backed control drums to further reduce the reflector envelope. Analysis also indicated the feasibility of cooling the shield with the 1100°F primary NaK coolant.

The reactivity worths of reflectors made of Be, BeO, and Ni were calculated. Experiments verified that poison-backed control drums provide more reactivity control than the conventional control drums of the same diameter.

B. FUEL DEVELOPMENT

Assembly of enriched fuel elements for the S8DS core was started in February and stopped in March, 1965. Variability of the chromized layer applied to the cladding tubes resulted in a low production yield of acceptable fuel elements. A chromizing process assuring better control is being developed in-house.

NAA-77-2 and -3, inpile permeation tests of the S8DS-type hydrogen barrier coating, successfully completed irradiation and are presently cooling in the ETR Canal.

NAA-115-1 and NAA-117-1, irradiation tests of sublength SNAP 8 fuel elements, continued operation.

C. COMPONENT DEVELOPMENT

The platinum resistance temperature detector (RTD) being developed for the NaK temperature sensing system has drifted less than -1.3°F after 7500 hours of endurance testing at 1300°F. The design goal is +2°F, -6°F drift during 10,000 hours at 1300°F.

Two test actuators of the S8DS design are being endurance tested at operational temperature (1025°F) and vacuum. The performance of these actuators has been satisfactory, stable, and above design requirement through 1600 and 1250 hours of testing, respectively.

One developmental test assembly of the S8DS control-drum-drive and scram mechanism has been on thermal-vacuum test for over 400 hours. The tests included four thermal cycles plus 60 drive-in and scram cycles. All operations have been satisfactory and within design requirements.

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The first two S8DS design position sensors were assembled and are being tested at 800°F in vacuum. The sensors are meeting all design requirements.

D. SYSTEM TESTS

1. SNAP 8 Experimental Reactor (S8ER)

The S8ER reached 1 year of operation in the power range 400 to 600 kw with a NaK outlet temperature of 1300°F on March 26, 1965.

The S8ER test program was completed as scheduled on April 15, 1965. Operating statistics for S8ER include:

- 1) 500 days of nuclear operation,
- 2) 365 days operation in the power range 400 to 600 kw at 1300°F NaK outlet temperature,
- 3) Over 5000 hours continuous operation, and
- 4) 5.1×10^6 kw hours energy release.

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

The second startup reliability test, consisting of nine automatic startups, was completed. After the startup tests, a performance record test was conducted. All systems operated satisfactorily.

3. SNAP 8 Ground Prototype (S8DS)

The reactor core vessel was completed and assembly of the spare core vessel was started.

The shield was successfully cast and is now being readied for thermal cycle tests.

Fabrication of pipe spools for the shakedown system is approximately 80% complete. The mounting of equipment to the support pallet has continued. Pre-assembly and wiring of instrumentation racks is about 50% complete. The vacuum system installation is about 90% complete.

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III. REFERENCE DESIGN

A. DESIGN DESCRIPTION

1. Instrument-Rated System

The SNAP 8 nuclear system for unmanned applications (shown in Figure 1) 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, by their position, control the amount of leakage flux. Each major assembly of the instrument-rated SNAP 8 nuclear system is described in the following paragraphs. Table 2 lists the detailed characteristics of the instrument-rated SNAP 8 nuclear system. A weight summary of the instrument-rated nuclear

system is given in Table 3, and the environmental criteria are summarized in Table 4. The design reliability goal for the instrument-rated nuclear system has been tentatively established at 96.3%.

a. Fuel Elements

The SNAP 8 reactor core contains 211 fuel elements made of Zr - 10.5 wt % U alloy hydrided to a hydrogen content of 6×10^{22} hydrogen atoms/cc of $(U-Zr)H_x$. The uranium is 93% 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 16.825 in. Other dimensions and characteristics are listed in Table 2.

b. Core Vessel and Internals (Less Fuel Elements)

The core vessel is 0.105-in.-thick 316 stainless steel. The core vessel contains an upper grid plate of Hastelloy-C, a coolant baffle plate of 316 SS, internal BeO reflectors clad in 316 SS, and the 211 fuel elements previously described.

c. Instrumentation and Reflector Assemblies

The beryllium control reflector assembly is made in two halves so that it forms a hollow right-circular cylinder completely surrounding the active core length. The beryllium reflector thickness is 3 in. $\pm$ 1 in. Each reflector drum is driven by its own drive mechanism. Three reflector drums are used as startup drums and are driven by springs. The other three reflector drums are used as control drums and are driven by long-term bidirectional control-drum actuators. A startup programmer and controller are provided to achieve proper operational sequence of the drums. A platinum resistance temperature sensor is used to measure the coolant outlet temperature.

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Figure 1. SNAP 8 Reflector, Core, and Shield

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TABLE 2
SNAP 8 REFERENCE INSTRUMENT-RATED NUCLEAR SYSTEM CHARACTERISTICS

Reactor Design		Nuclear Parameters	
Fuel Elements		Average thermal neutron flux <1 ev	3 x 10 ¹²
Number	211	n/cm ² -sec	0 21
Fuel alloy, wt % U in Zr	10.5	Median fission energy, ev	8 4
Uranium enrichment, at % U ²³⁵	93.15	Mean prompt neutron lifetime, μ -sec	7 55
Degree of hydriding, NH ₄ , 10 ²² a/cm ³	6 05	Uranium loading, kg U ²³⁵	0 50
Fuel rod diameter, in	0 530	Hydrogen reactivity worth, % Δ k/k	0 50
Active fuel length, in	16 825	Volume fractions in core	
Fuel element length, in	17 145	Fuel	0 782
Cladding material	Hast-N	Cladding	0 061
Cladding ID, in	0 540	NaK	0 123
Cladding thickness, in	0 010	Ceramic	0 013
Ceramic barrier material	SCB	Void	0 021
Ceramic barrier thickness, in	0 002	Radial peak to average power	1 22
Diametral hydrogen gap, in	0 006	Axial peak to average power	1 44
Source		Average power density, watts/cc	36 6
Type	Po-Be	Reliability Goals %	
Strength, curies	0 1	Overall	96 30
Location	top head of vessel	Core assembly	99 45
Internal Reflectors		Reflector assembly	99 28
Composition	BeO	Shield assembly	99 84
Cladding material	316 SS	Safety assembly	99 11
Cladding thickness, in	0 030	Reactor controls	98 54
Inlet Plenum Grid Plate		Cumulative Reliability Goals	
Material	Hast-C	Through launch	99 25
Thickness, in	0 650	Through startup	98 58
Diameter, in	9 186	Through 10,000 hours	97 08
Number of coolant holes	372	Complete mission	96 30
Diameter of coolant holes, in	0 156	Control System Parameters	
Pitch of coolant holes, in	0 570	Single control drum worth, % Δ k/k	2 4
Outlet Plenum Grid Plate		Full-in drum position, degrees	0
Material	316 SS	Full-out drum position, degrees	154
Thickness, in	0 850	Drum movement per step, degrees	0 26
Number of coolant holes	372	Maximum drum rotation rates,	
Diameter of coolant holes, in	0 156	degrees/sec	
Pitch of coolant holes, in	0 570	Startup drums (simultaneous motion)	0 086
Coolant		Control drums	0 053
Material	NaK-78	Maximum differential drum worth,	
Melting point, °F	12	¢/degree (nominal shim)	5 2
Core Vessel		Kinetic Parameters	
Material	316 SS	Number of delayed neutron groups	6
Internal diameter, in	9 214	Decay constants, sec ⁻¹	
Thickness, in	0 105	λ_1	0 0124
Length, in	26 24	λ_2	0 0305
Distance between grid plates, in	17 224	λ_3	0 111
Reflector Control Elements		λ_4	0 301
Material	Be	λ_5	1 13
Number of startup drums	3	λ_6	3 0
Number of long-term control drums	3	Effective delayed neutron fraction	0 0080
Total number of drums	6	Group delayed fractions	
Radius of curvature of drums	4 5	α_1	0 0329
Nominal thickness of drums, in	3 125	α_2	0 2194
Shimming capability, in	+1 5, -1 0	α_3	0 1960
Envelope, drums full out, in	27 6	α_4	0 3947
Overall length, in	18 5	α_5	0 1150
Drum length, in	15 375	α_6	0 0419
Reactor Parameters		Transport delay times (% 35 kwe) sec	
Design Operating Conditions		Nuclear system to boiler	1 0
Reactor thermal power, kwt	600	Boiler	6 8
Net electrical power, kwe	35 (@ 470 kwt)	Boiler to nuclear system	4 5
Coolant inlet temperature, °F	1100	Nuclear system	1 7
Coolant outlet temperature, °F	1300	Typical Shield Parameters	
Coolant flow rate, lb/sec	13.6	Instrumented Payload Dimensions	
Operating pressure, psia	45	Diameter, ft	20
Core pressure drop, psi	4 8 (@ 600 kwt)	Axial distance from reactor core center, ft	40
Maximum hot spot fuel temperature, °F	1520	Shield Materials	
Maximum hot spot cladding temperature, °F	1430	Cladding	347 SS
Operating lifetime, hr	10,000	Gamma	W
Peak heat flux, Btu/hr-ft ²	~100,000	Neutron	L ₁ H
NS pressure drop, psi	<8 (@ 600 kwt)	Shield Dimensions	
Reactivity Requirements, % Δ k		Overall shield length, in	34 3
Xenon	-0 9	Neutron shield thickness, in	33 3
Samarium	-0 9	Gamma shield thickness, in	1 0
Burnable poison (Sm ₂ O ₃)	+2 5	Maximum diameter of conical shield, in	51 5
Fission product poisons	-0 7	Minimum diameter of conical shield, in	39 2
U ²³⁵ burnup	-0 5	Radiation Levels Due to Reactor Sources	
Temperature and power	-1 7	Fast neutron dose at payload, nvt	1 x 10 ¹¹
Hydrogen redistribution	-0 4	Gamma dose at payload, rads (C)	1 x 10 ⁶
Hydrogen loss	-1 6	Shield Assembly Weight	
Contingency	-2 4	Neutron shield, lb	1365
Cold clean excess reactivity	+6 6	Gamma shield, lb	115
Total control drum worth	13 1	Shield vessel and structure, lb	450
Shutdown margin	6.5	Total shield assembly weight without armor, lb	1930
Temperature Coefficients, ¢/°F (at design power level)			
Spectrum	-0 01		
Axial expansion	-0 048		
Upper grid plate	-0 026		
Lower grid plate	-0 019		
Reflector (prompt, lower plenum)	-0 04		
Reflector (prompt, upper plenum)	-0 04		
Total isothermal (@ 1200°F)	-0 20		
Total isothermal (@ room temperature)	-0 15		

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TABLE 3
SNAP 8 NUCLEAR SYSTEM WEIGHT SUMMARY

System Component	Quantity	Weight Per Item [†] (lb)	Total Weight (lb)	Center of Gravity [†] (in)		
				X	Y	Z
Core Assembly	1	-	310.3	26.0	0 0	0.0
Core vessel	1	45.2	45.7	24.5	0.0	↑
Outlet plenum grid plate	1	5.9	5.9	16 1	0.0	
Fuel elements	211	0.8	193.9	25.1	0.0	
Internal reflectors	42	-	12.0	25.0	0.0	
Inlet plenum grid plate	1	6.5	6.5	33 4	0 0	
Flow baffle plate	1	7.0	7.0	34 6	0 0	
Inlet NaK pipe (16 in long)	1	1.8	1.8	40 0	13 5	
Outlet NaK pipe (54 in. long)	1	8.4	8.4	25.0	10 0	
NaK in vessel	-	18 7	18.7	26 9	0.0	
NaK in pipe	-	10.4	10.4	27 9	11.2	
Reflector Assembly	1	-	284.6	22 8	+ 0.1	↓
Control drum actuators	3	20.0	60 0	11.5	0 0	
Startup drum drives	3	0.8	2 4	14.0	0 0	
Reentry actuator assembly	1	6.0	6.0	16 5	-11 2	
Upper drum support assemblies	6	2.0	12.0	16.5	0 0	
Fixed external reflectors	2	22.5	45.0	25.0	0 0	
End-of-life shutdown actuator	1	1.8	1.8	16.5	11 2	
Control and startup drums	6	19.9	119.5	25.0	0 0	
Shear and load lugs	8	1 0	8.0	31.2	0 0	
Drum release actuators	6	0.6	3.6	33.5	0 0	
Lower drum support assemblies	6	2.0	12.0	33.8	0.0	
Reflector ejection springs	4	1.0	4.0	36 0	0.0	
Drum position indicators	3	0.5	1.5	36.4	0.0	
Drum key lock brackets	4	0.9	3.6	15.3	0.0	
Miscellaneous brackets and fittings	-	-	5.2	34.9	0.0	
Shield Assembly	1	1930	1930	58.1	+ 0.3	
Miscellaneous						
Supplemental reflector drum shims	6	6.2	37.1	25.0	0.0	↓
Environmental cover	1	~65	~65	~44	0.0	
Payload instrumentation	-	~17	~17	-	-	
Diagnostic instrumentation	-	~50	~50	-	-	
Interconnecting cables	-	~20	~20	~44	-	-

*Weight expressed to the nearest 0.1 lb

†X axis — is the centerline of the nuclear system. X = 0 is a point 25.000 in. forward of the midpoint between the upper and lower grid plates. The positive direction is toward the nuclear system.

Y axis — is perpendicular to the X axis with the positive direction being toward the side of the reactor where the coolant outlet pipe crosses over the side of the reactor.

Z axis — is orthogonal to the X and Y axes with the positive direction being 90° clockwise from the positive Y axis when viewing the nuclear system from the reactor end.

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TABLE 4
SNAP 8 NUCLEAR SYSTEM ENVIRONMENTAL CRITERIA
(Sheet 1 of 2)

Environment	Ground Handling and Prelaunch [§]	Launch [†]	Orbit [§]															
1. Pressure	External pressures: sea level to 50,000 ft with maximum change ±1.25 psi/min.	External static pressure envelope as a function of time. (To be determined)	External pressure: 10 ⁻¹² mm Hg															
2. Temperature	+50 to +160 °F	-	-															
3. Humidity	a. Protected during shipment and storage (RH < 70%) ^{§§} b. During vehicle integration: 6 hr at 65% RH and 106 °F; 6 hr of decreasing temperature to 91 °F and increasing RH to 100%; 8 hr of decreasing temperature to 81 °F with release of 4.4 grains H ₂ O/ft ³ ; 4 hr of increasing temperature to 106 °F and decreasing RH to 50%	-	-															
4. Rain	a. Protected during shipment and storage b. During vehicle integration: Anticipated precipitation <table><tr><td>Rate (in./hr)</td><td>Amount (in.)</td><td>Time</td></tr><tr><td>0.6</td><td>7.1</td><td>12 hr</td></tr><tr><td>6</td><td>0.5</td><td>5 min</td></tr><tr><td>12</td><td>0.2</td><td>1 min</td></tr><tr><td>0.34</td><td>4.0</td><td>12 hr</td></tr></table>	Rate (in./hr)	Amount (in.)	Time	0.6	7.1	12 hr	6	0.5	5 min	12	0.2	1 min	0.34	4.0	12 hr	-	-
Rate (in./hr)	Amount (in.)	Time																
0.6	7.1	12 hr																
6	0.5	5 min																
12	0.2	1 min																
0.34	4.0	12 hr																
5. Sand and dust	a. Protected during shipment and storage b. Applicable components tested with 140-mesh silica flour for 3 hr at 77 °F and RH < 30%; sand and dust density, 0.1 to 0.25 grams/ft ³ , with air velocity = 2500 ± 500 ft/min	-	-															
6. Salt spray	a. Protected during shipment and storage b. Applicable components tested to demonstrate ability to withstand exposure to salt spray for 6 wk as experienced at coastal regions of continental U.S. A.	-	-															
7. Fungus	Applicable components tested per MIL-E-52720(ASG), Fungus Resistance Tests, Procedure 1.	-	-															
8. Shock	Drop of 18 in. measured at either a corner or a surface, onto a concrete surface.	System Three shocks of 18 g's in each of 3 mutually perpendicular axes (18 total) of one of the following pulses: a. Triangular pulse 10 msec b. Half-sine pulse 8 msec c. Rectangular pulse 5 msec Component: Three shocks of 35 g's in each of 3 mutually perpendicular axes (18 total) of one of the following pulses: a. Triangular pulse 10 msec b. Half-sine pulse 8 msec c. Rectangular pulse 5 msec	Nonoperating ^{††} : capable of withstanding 3-g shocks on each of 3 mutually perpendicular axes at one of the following wave forms: a. Triangular pulse 10 msec b. Half-sine wave pulse 8 msec c. Rectangular pulse 5 msec Operating: withstand 3-g peak, half-sine wave pulse of one msec at mounting interface															

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TABLE 4
SNAP 8 NUCLEAR SYSTEM ENVIRONMENTAL CRITERIA
(Sheet 2 of 2)

Environment	Ground Handling and Prelaunch*	Launch†	Orbit§
9. Vibration	Sweep applied to transportation container on 3(30-min log) mutually perpendicular axes: 2 to 5 cps at 1 in. double amplitude; 5 to 27.5 cps at 2-g peak; 27.5 to 43.5 cps at 0.0517 in. double amplitude; 43.5 to 500 cps at 5-g peak; then a 5-min dwell at each resonant frequency noted at above amplitude	<p>System</p> <p>Sweep from 5 to 16 cps at 0.368 in. double amplitude and 16 to 2000 cps at 5-g peak on each of 3 mutually perpendicular axes; then 5-min dwell at each major resonant frequency at the above amplitude</p> <p>Component</p> <p>16 to 100 cps at 6-g peak; 100 to 180 cps at 0.0118 in. double amplitude; 180 to 2000 cps at 19-g peak; then dwell at 1/2 above amplitude for 5 min at each major resonant frequency</p> <p>Random Vibration - Components</p> <p>A random vibration sweep test over a frequency interval of 20 to 2000 cps for 5 min in each of 3 mutually perpendicular axes according to the following schedule:</p> <p>20 to 200 cps at +2 db/oct 200 to 700 cps at 0.64 g²/cps 700 to 900 cps at -17.5 db/oct 900 to 2000 cps at 0.15 g²/cps</p>	Capable of withstanding vibrations of 0.25-g peak over a frequency of 5 to 2000 cps for 5 min at the NS mounting interface
10. Wind	Withstand steady-state wind force of 54 mph with wedge-shaped gusts of 75 mph	-	-
11. Storage life	Two years (except source)	-	-
12. Radio frequency generation and interference	-	Designed and tested to MIL-STD-826 (USAF) dated 20 January 1964	Designed and tested to MIL-STD-826 (USAF) dated 20 January 1964
13. Acoustic noise	-	Total integrated sound pressure level of 148 db, Re 0.0002 microbar over a specific frequency spectrum	-
14. Acceleration	-	Longitudinal axis: +7 g for 5 min, -3 g for 5 min; lateral and normal axes: ±4.5 g for 5 min	Operating: +1 g along axis of symmetry and ±1/4 g radially including rotational effects Nonoperating:†† +3-1/2 g along longitudinal axis in combination with ±1 g on any lateral axis for a period of time less than 5 min
15. Magnetic fields	-	-	To be determined
16. Gravitational fields	-	-	To be determined
17. External radiation	-	-	To be determined

*Environment to which the nuclear system may be subjected without any detrimental effects during shipment from Atomics International up to the time of encapsulation in the launch vehicle nose cone

†Environment to which the nuclear system may be subjected without any detrimental effects after exposure to the ground handling and prelaunch environment up to the time of separation from the launch vehicle

§Environment to which the nuclear system may be subjected without any detrimental effects after exposure to the launch environment

**RH = Relative humidity

††Nonoperating: Defined as prohibiting operation of the reflector-control system; i.e., control drums are held stationary

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d. Shield

A typical neutron shadow shield designed to limit the dose at the payload is in the shape of a truncated cone whose cone angle completely hides the payload from the reactor assembly. The neutron shield is made of lithium hydride (LiH) cast in a stainless-steel container. Between the neutron shield and the reactor assembly is a circular plate of tungsten designed to limit the gamma dose at the payload. The design parameters for a typical 600-kwt shield are given in Table 2.

e. Safety System

The safety system includes a reentry actuator, reflector ejection springs, a destruct charge, an end-of-life scram actuator, a telemetry scram actuator, a reflector unlocking 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 actuator is designed to hold the reflector assembly in place at the top of the core vessel. The actuator is designed and located so that it will be melted by reentry heat. Heating of this actuator allows the reflector ejection springs to remove the reflector assembly from the vicinity of the core vessel.

The destruct charge is a shaped explosive charge surrounding the core vessel. It can be fired prior to reactor startup to preclude reactor criticality in the event of an unsuccessful launch.

The reflector unlocking actuator is designed to hold the reflector assembly to the core vessel during launch. Thus, it is not necessary for the reentry actuator assembly to hold the reflector assembly under launch conditions.

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

An environmental cover is supplied to protect the nuclear system from the natural environments such as rain, sand, and dust. The cover, using a side ejection concept, is supported from the base ring of the shield. View ports are provided for access to the pinpullers and the key locks. The cover is made of 5-mil aluminum honeycomb. The destruct charge is mounted on the cover and is ejected with the cover once startup is initiated.

2. Manrated System Concepts

The results of studies of the conceptual shadow-shield manrated SNAP 8 design have been previously reported.* This is the first time preliminary results of studies of conceptual 4π -shield designs are reported. Upon completion of these two studies, another study will be conducted to combine the results into one reference SNAP 8 nuclear system which will be adaptable to either shadow shielding or 4π shielding.

a. Shadow-Shielded Concept

The SNAP 8 manrated shadow-shield nuclear system concept is shown in Figure 2. Each major assembly of this concept is described in the following paragraphs.

*C. E. Johnson, "SNAP 8 Progress Report, November 1964 - January 1965," NAA-SR-10792 (CRD), March 15, 1965

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(1) Core Assembly

The core assembly is composed of a 316 stainless-steel core vessel which contains 211 fuel elements and internal reflectors in a hexagonal array as in the reference instrument-rated design. A stainless-steel flow-baffle plate is supplied at the inlet to the core to provide proper flow distribution. Attached to the outside of the core vessel are brackets to support and position the reflector assembly.

(2) Reflector Assembly

The core vessel is surrounded by an annulus of beryllium approximately 17-1/2 in. long which tapers from a thickness of about 2-1/2 in. at the control-drum-drive end of the core to about 4 in. at the shield end.

Eight control drums made of beryllium are located around the core vessel. The drum shape approximates a truncated half-cone with the axis of rotation inclined at approximately 6° to the axis of the core vessel. The flat sides of the control drums are 1/4-in. from the parallel to the axes of the drums.

Each control drum is driven by a stepping actuator. These actuators are mounted above the core vessel to place them in the most favorable position for radiation of their heat to space and to eliminate any interference with the piping at the lower end. The diameter of the actuators makes it necessary that they be set inward from the drum axes and that a gear train be used to connect the actuator and drum shafts.

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Figure 2. SNAP 8 Manrated Shadow-Shielded Nuclear System Concept

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The control-drum motor used is a bidirectional stepper motor, which upon command, steps the control drum a nominal $1/4^\circ$. The drive motor is a "fail-as-is" device which maximizes nuclear system life and minimizes inadvertent shutdown. No "scram" circuits are provided and shutdown of the nuclear system is accomplished by driving the control drums to their least reactive (out) position. Shutdown can be accomplished even with four control drums in the full-in (most reactive) position.

(3) Control and Instrumentation Assembly

The control and instrumentation assembly consists of a temperature sensor, a controller, a programmer, and a relay box.

The reference instrument-rated control scheme is used in the manrated concept because of its simplicity and inherent reliability. It senses the reactor coolant outlet temperature and maintains this temperature between preset limits (1280 and 1330°F) by means of an on-off digital control system. The electronics for the manrated control system, however, will be simplified by omitting some of the memory functions required for instrument-rated applications. Also, manual overrides will be provided in the manrated controller.

(4) Shield Assembly

An important part of the manrated nuclear system is the radiation shield. The shield must be light-weight and long-lasting, and must provide adequate protection to the crew from both neutron and gamma radiation emitting from the nuclear system. Since the shield weight is a large fraction of the total system weight, the most efficient shield materials and the optimum configuration are of primary importance.

The manrated SNAP 8 shadow shield concept consists of two sections, each having a two-region shield, separated by a primary coolant loop gallery. This gallery will contain the primary loop pumps and the primary-to-secondary heat exchanger, i. e., NaK-to-NaK or NaK-Hg heat exchanger, depending upon the type and size of the power conversion system being used.

The overall shield assembly must provide sufficient reduction in dose to the spacecraft to protect the astronauts, while the first section, i. e., that section close to the reactor, must provide sufficient protection to prevent activation of the secondary loop, whether it contains mercury or NaK.

Each section of the reference shield concept contains two regions, a depleted-uranium gamma shield and a lithium-hydride neutron shield. As the gamma shield is particularly heavy, it appears advantageous to place as much of this shield as possible in the first section of the shield. As the gamma shield itself produces gammas, not all of the gamma shielding can be located in the first section and the ratio of core-generated gammas to shield-generated gammas should be optimized.

In order to eliminate all neutron scatter sources, the primary and secondary piping and all electrical cabling are routed through the shield rather than around the shield.

A typical manrated-concept shield assembly weighs about 5135 lb. It shields a 60 -ft-diameter payload located 150 ft from the core center to a dose rate of 1.5 mrem/hr when the reactor is operating at its rated power of 600 kw.

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b. 4π Shield Concept

The SNAP 8 manrated 4π shield nuclear system concept will use as many of the instrumented and manrated shadow shield system components as practical. The fuel elements are identical. To increase the kw-hr/lb, however, consideration is being given to adding approximately 30 fuel elements to increase either the system lifetime or rated system power or both. Changing the reflector material from Be to BeO is under study as the temperature of the reflector assembly is increased due to the close proximity of the 4π shield.

Preliminary results of critical experiments indicated that a suspected problem of decreased control-drum worth did not materialize.

B. ANALYSIS

1. Nuclear Analysis

a. SNAP 8 Control-Drum Worths

A thorough experimental study of reflector worth was conducted on the SNAP Critical Assembly - 4A (SCA-4A), illustrated in Figures 3, 4, and 5. The incremental reactivity

worth of the beryllium reflector was measured radially and axially. Description of the assembly and its components and detailed description of some of the results have been reported by Clifford.* The axial and radial reflector worth functions were found to be generally separable. These functions may be used to predict reactivity changes due to reflector modifications within 10%.

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Figure 4. SCA-4A Core Diagram

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Figure 3. SNAP Critical Assembly No. 4A

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Figure 5. Top View of a Quadrant Without its Cover Plate

*Clifford, D. W., "Interim Report on the SNAP Critical Assembly 4A (SCA-4A) Experimental Results," NAA-SR-10245 (CRD), March 20, 1965

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A parametric study of drum worth evaluated four drum thicknesses, each at a number of different shimmed thicknesses. In some configurations, lucite was used to adjust the nonrotating portions (Figure 5) of the reflector quadrants. Lucite has proven to be very suitable for mock-ups of beryllium.

The observed control worths ranged from \$1.04 for a 2.0-inch-radius, 2.75-inch-thick drum to \$4.42 for a 4.65-inch-radius, 4.50-inch-thick drum. The drum geometry was optimized and a basis was found for prediction of control drum worth. The optimum drum thickness, with respect to reactivity worth, was determined to be approximately equal to the radius of the drum (Figure 6).

Figure 7 shows the variations of SCA-4A semicylinder drum worths with drum radius. The drums-out position affording the greatest control was 135 degrees rotation from the drums-in position (Figure 8): The SCA-4A data was scaled to S8ER control-drum calibration data available at one thickness and radius. The ratio of worths of identical drums on the S8ER and SCA-4A assemblies was assumed to be independent of drum geometry.

The differences between the S8ER and S8DS were taken into account both analytically and experimentally. The differences between the composition, height, and radius of the two cores were estimated by the reflector worths calculated by the DTF-II transport theory code. The drum worths were assumed to be proportional to the calculated reflector worth. The other difference considered was that the S8DS drum is shorter than the reflector (and the active fuel length). This effect was also investigated on the SCA-4A and the data was normalized to the SCA-4A reflector worth and height. The normalized data was applied directly to the S8DS to get the final value of the predicted drum worth, \$4.0 + \$0.4 - \$0.6.

1-8-65 7568-01944
Figure 6. Variation of SCA-4A Drum Worth with Drum Thickness

4-27-65 7568-01945
Figure 7. SCA-4A Semicylinder Drum Worth

10-26-64 7647-0098
Figure 8. SCA-4A Maximum Control Position of a 3-1/4-in.-thick Control Drum

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Reactivity worth of an eight-drum, 135-degree reflector system was estimated from an S8ER six-drum calibration. This calibration was performed with an adjacent drum open. The ratio of two S8ER calibrations gave the interaction between nearest neighbor drums. This interaction effect results in a total drum worth about 10% less than the sum of individual drum worths. Figures 9 and 10 show the worth of an eight-drum semicylindrical reflector system. The drum worths for other thicknesses (thickness $\neq$ radius) are seen in Figure 11.

Analysis of the nominal shim S8DS drums by the one-dimensional DTF-II transport theory code indicated a worth of \$3.4 to \$3.7 for drums

3-30-65 7568-01946
Figure 9. Single-Drum Reactivity Worth of an Eight-Drum, 135° Semicylinder Reflector System

3-30-65 7568-01947
Figure 10. Total Reactivity Worth of an Eight-Drum, 135° Semicylinder Reflector System

3-30-65 7568-01948
Figure 11. Effect of Drum Thickness and Radius on the Total Reactivity Worth of an Eight-Drum, 135° Reflector System

capable of 105 degrees rotation. This is in good agreement with the results above. This approach determines the total reactivity difference between the drums-in and the drums-out configuration.

b. Hydrogen Worth in SNAP 8 Reactors

Figure 12 shows the reactivity worth of the hydrogen concentration in a SNAP 8 reactor. The worth of each hydrogen atom decreases as the hydrogen concentration increases. The design points are shown for the S8ER, S8DS, and the manrated concept.

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4-5-65 7568-01949
Figure 12. Hydrogen Worth

TABLE 5
PROJECTED REACTIVITY REQUIREMENTS
FOR 20,000-HOUR OPERATION AT
600 kwt, 1300°F OUTLET
TEMPERATURE

Parameter	Reactivity (\$)
Rise to temperature and power	2.2
Hydrogen redistribution	0.6
Xenon fission product	0.9
Samarium fission product	1.2
Other fission products and uranium depletion	2.8
Hydrogen loss	4.3
Helium-3 buildup (shut-down device)	2.0
Contingency	3.0
Total	17.0

4-27-65 7568-01950
Figure 13. Poison Penalty for SNAP 8
Operation at 600 kwt as a Function of
Mission Time

TABLE 6
REACTIVITY MODIFICATIONS TO THE
S8DS FOR THE MANRATED
SNAP 8 REACTOR

Modification	Reactivity (\$)
Addition of 30 fuel elements	3.0
Increase element separation to 30 mils	-2.5
Add one inch to core length	1.0
Increase hydrogen concentration from 6.05×10^{22} to 6.25×10^{22} atoms/cc	2.0
Increase equivalent reflector thickness	2.5
	+6.0
S8DS nominal excess	11.0
Manrated SNAP 8 nominal excess	17.0

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c. End-of-Life Shutdown Device

The purpose of the shutdown device is to render the reactor subcritical in water after 50 years in orbit. The shutdown device is based on the decay of tritium atoms (12.26 years half-life) to the highly absorbing helium-3 atom ($\sigma_{n,p} = 5400$ barns). Figure 13 shows the fraction of tritium converted to helium-3 for a 600-kwt SNAP 8 reactor for various "shelf times." Preliminary analysis suggests that the incorporation of this device in the reactor (hydrogen concentration = 6.25×10^{22} atoms per cc) would require a one-inch increase in core length to compensate for the reactivity loss.

d. Manrated SNAP 8 Reactor Concept Reactivity Requirements

Table 5 provides the projected reactivity requirements for 20,000 hours of operation at conditions of 600 kwt power and 1300°F outlet temperature. The modifications to the S8DS design which would furnish the necessary reactivity are presented in Table 6. A comparison of the significant design parameters of the S8ER, S8DS, and the manrated reactor concept is given in Table 7.

2. Thermal Analysis

The operating capabilities of the SNAP 8 reference design were determined for various levels of system reliability. The high reliability of the SNAP 8 nuclear system is attained by redundant use of components. The magnitude of the subsystems' redundancy is proportional to

the amount of surplus reactivity provided in the reactor. The existing SNAP 8 reference system, operating at the design point of 600 kwt for 10,000 hours with a 1300°F outlet coolant temperature, has approximately \$12 of reactivity of which \$3 is redundant. The \$3 reactivity surplus is equivalent to the failure of 18 elements, or 1 control drum and 9 fuel elements. The nuclear systems analyzed were assumed to have \$15 of excess reactivity which corresponds to increasing the beryllium reflector thickness of the reference reactor by approximately one inch.

A conservative analysis was obtained by reserving a constant rate of surplus reactivity (\$3 per 10,000 hours). This assumes that component failure per unit time is constant. However, the total component reliability for the first 10,000 hours is dependent upon the reliability achieved during all three of the following phases of operation: launch, startup, and 10,000 hours of endurance. Continued system operation is only dependent upon the reliability associated with endurance. It is expected that the component failure rate associated with long-term endurance will remain lower than the total failure rate for the first 10,000 hours. Therefore, the assumption that the system reliability remains constant if \$3 of reactivity is reserved per 10,000 hours of operation should provide conservative results.

TABLE 7
COMPARISON OF THE S8ER, S8DS, AND MANRATED SNAP 8 REACTORS

Reactor	Active Fuel Element Length (in.)	Fuel Rod Diameter (in.)	Fuel Element Cladding Outside Diameter (in.)	Uranium Concentration (wt %)	Hydrogen Concentration (10^{22} atoms/cc)	Number of Fuel Elements	U-235 Loading (kg)	Core Vessel Inside Diameter (in.)	Fuel Element (Triangular) Spacing (in.)	Active Core Volume (in. ³)
S8ER	14.00	0.5324	0.560	10.01	5.96	211	6.02	9.214	0.57*	832
S8DS	16.825	0.530	0.560	10.5	6.05	211	7.65	9.214	0.57*	1000
MRS8	17.825	0.530	0.560	10.5	6.25	241	9.25	10.06	0.59†	1295

*Fuel element spacing 10 mils

†Fuel element spacing 30 mils

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4-20-65 7568-01951
Figure 14. Probable Operating Limit of SNAP 8
Reference Design Reactor, \$3.00 Surplus
Reactivity per 10,000 Hours

4-20-65 7568-01953
Figure 16. Operating Potential of SNAP 8
Reference Design Reactor as a Function
of Outlet Coolant Temperature

4-20-65 7568-01952
Figure 15. Probable Operating Limit of
SNAP 8 Reference Design Reactor,
\$3.00 Surplus Reactivity per
Mission

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For the SNAP 8 reference system, the only operating limitation is loss of excess reactivity; however, in order to maintain a high system reliability, other criteria limits are imposed in attempting to minimize reactivity losses. The other limits are fuel swelling, phase change, and thermal stress; cladding and core vessel creep; and beryllium swelling. In the outlet coolant temperature range analyzed, only the fuel swelling criteria and the excess reactivity limit provide bounds to reactor operation. All analysis was based on the entire reactor operating as the "hot channel" fuel element. Figure 14 indicates the wide range of operating power levels, outlet coolant temperatures, and lifetimes of SNAP 8 reactors. The conservative assumptions used in the calculations, particularly the \$3 surplus reactivity per 10,000 hours, should provide a high probability of achieving the calculated lifetimes.

A more realistic, yet still conservative, assumption was to provide \$3 of surplus reactivity per reactor mission. The \$3 redundant reactivity is about the minimum redundancy which will assure adequate system operating reliability. Figure 15 shows the capability of the system using these assumptions.

An approximate upper bound to the performance of the SNAP 8 reference design reactor is shown in Figure 16. This figure is based on the assumptions that no surplus reactivity is required and that the reactor will achieve the maximum operating potential. The conservatism of the hot channel factors, however, is retained in this analysis.

3. Control Analysis

a. Cladding Simulation for Core Transient Heat Transfer

The present analog computer model for core transient heat transfer is a modified "backward

difference" model. It consists of five axial nodes plus an entrance fuel node, each divided into two radial nodes (fuel and coolant). In using this model, the heat capacity of the fuel cladding has been either neglected or lumped with that of the fuel for convenience and conservatism. The rate of change of cladding temperature has been assumed to be equal to the fuel or coolant rate, whichever is greater.

In order to verify or modify this assumption, a study has been made on the analog computer comparing the response of the present two-node model with that of a three-radial-node model including a cladding node. The rate of change in the cladding temperature as computed by the three-node model was taken as the base case with which the fuel and coolant rates computed by the two-node model were compared.

Input to each model consisted of step and ramp changes in coolant flow, core power generation, and coolant inlet temperature. Flow changes were step decreases from 6.59 to 1.32 lb/sec and ramp decreases over this same interval with ramp rates varying from 0.105 to 1.05 lb/sec². Power changes were step increases from 76 to 176 kwt and ramp increases over the same interval at rates varying from 0.5 to 4.0 kwt/sec. Inlet coolant temperature changes were step increases of 20°F and ramp increases of 20°F at rates ranging from 50 to 600°F/min.

The combined flow, inlet temperature, and heat load changes corresponding to a reference power conversion system startup* were also used as input for the model comparison. This input involves a combination of inlet coolant temperature change of about -150°F/min, flow change of about 0.25 lb/sec², and load change of about 2 kwt/sec.

*Code, C. J., "Analysis of the Response of the SNAP 8 Nuclear System During Startup of the Power Conversion System," NAA-SR-9626 (CRD) June 8, 1964

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For the step changes and for the most rapid ramp changes, the heat capacities of the coolant and fuel in the two-node model were varied to simulate inclusion of the cladding heat capacity. Analog runs were made lumping 0, 25, 50, 75, and 100% of the cladding heat capacity with that of the coolant. The remaining cladding heat capacity was lumped with that of the fuel.

The equations for the three-radial-node model (fuel, cladding, and coolant) can be written as follows for the j^{th} axial node, in Δ form.

$$(1) \quad \frac{d}{dt}(\Delta T_{fj}) = \frac{G_{jr}}{M_{fj} C_f n_r} - \frac{U_{fcd} A_j}{M_{fj} C_f} (\Delta T_{fj} - \Delta T_{cdj})$$

$$(2) \quad \frac{d}{dt}(\Delta T_{cdj}) = \frac{U_{fcd} A_j}{M_{cdj} C_{cd}} (\Delta T_{fj} - \Delta T_{cdj}) - \frac{U_{cdc} A_j}{M_{cdj} C_{cd}} (\Delta T_{cdj} - \Delta T_{cj})$$

$$(3) \quad \frac{d}{dt}(\Delta T_{cj}) = \frac{U_{cdc} A_j}{M_{cj} C_c} (\Delta T_{cdj} - \Delta T_{cj}) - \frac{W}{M_{cj}} (\Delta T_{cj} - \Delta T_{c,j-1}) - \frac{\Delta W}{M_{cj}} (T_{cjo} - T_{c,j-1,o})$$

where

T = temperature ($^{\circ}\text{F}$),

t = time (seconds),

G = heat generation rate (Btu/sec),

M = mass (lb),

C = specific heat (Btu/lb- $^{\circ}\text{F}$),

n = nuclear power level

U_{fcd} = overall heat transfer coefficient from fuel to cladding (Btu/sec-ft 2 - $^{\circ}\text{F}$),

A_j = heat transfer area (ft 2),

U_{cdc} = overall heat transfer coefficient from cladding to coolant (Btu/sec-ft 2 - $^{\circ}\text{F}$), and

W = coolant flow rate (lb/sec).

Subscripts

r = reference level,

o = initial value,

j = j^{th} axial node,

f = fuel,

cd = cladding, and

c = coolant.

For a transient of the order of the power conversion system startup transient, the maximum rate of change of coolant temperature as computed with the two-radial-node model (fuel and coolant) is about 15% higher than the maximum rate of change of cladding temperature in the three-radial-node model. Since the two-node model does not yield cladding temperature directly, this leads to about 15% extra conservatism in the calculated cladding rates.

This excess conservatism can be as high as 250 to 300% for step changes in flow or inlet temperature which, of course, could be approached but never realized. The error for step changes can be reduced to about 40 or 50% by lumping the cladding heat capacity with that of the coolant.

For rapid ramp changes in flow (~ 1 lb/sec 2) or inlet coolant temperature ($\sim 600^{\circ}\text{F}/\text{min}$), the estimated cladding rate is from 20 to 40% too high. This error can be reduced to 10 or 20% by lumping the clad with the coolant. For rapid ramp changes in power (4 kwt/sec), the estimated cladding rate from the two-node model is about 30% high regardless of the distribution of the cladding heat capacity in the fuel or coolant nodes.

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TABLE 8
RESPONSE OF CLADDING SIMULATIONS TO CHANGES IN FLOW, POWER, AND
INLET COOLANT TEMPERATURES

Driving Force of Transients	Maximum Coolant or Fuel Rate* (°F/sec)	Maximum Cladding Rate† (°F/sec)	Error (%)
<u>Flow Changes</u>			
5.27 lb/sec step (all clad with coolant)	23.5	19.5	20
5.27 lb/sec step (no clad with coolant)	82	19.5	320
1.05 lb/sec ² ramp (all clad with coolant)	5.0	4.5	11
1.05 lb/sec ² ramp (no clad with coolant)	6.2	4.5	38
0.105 lb/sec ² ramp (no clad with coolant)	2.0	1.8	11
<u>Power Changes</u>			
100 kwt step (all clad with coolant)	4.5	2.9	55
100 kwt step (no clad with coolant)	4.2	2.9	45
4 kwt/sec ramp (all clad with coolant)	2.5	1.95	28
4 kwt/sec ramp (no clad with coolant)	2.5	1.95	28
0.5 kwt/sec ramp (no clad with coolant)	0.42	0.30	40
<u>Inlet Temperature Changes</u>			
20°F step (all clad with coolant)	32	23	39
20°F step (no clad with coolant)	80	23	250
600°F/min ramp (all clad with coolant)	6.3	5.4	17
600°F/min ramp (no clad with coolant)	6.6	5.4	22
50°F/min ramp (no clad with coolant)	0.8	0.8	0
Reference PCS startup	2.2	1.9	16

*2 radial node model (fuel and coolant)

†3 radial node model (fuel, cladding, and coolant)

Table 8 summarizes the comparison of the cladding temperature rate of change from the three-node model with the greater of the fuel or coolant rates from the two-node model.

The error involved in estimating cladding temperature rate of change from the two-radial-node model is dependent on both the severity of the transient and the nature of the driving force (e. g., power, flow, or inlet temperature). The error for power changes is practically unaffected by lumping the cladding heat capacity with either

the fuel or the coolant. The error for flow and inlet temperature changes can be reduced, however, by lumping the cladding heat capacity with that of the coolant, and the results will still be conservative.

In future calculations, the three-node model will be used instead of the two-node model unless time or the availability of the computer equipment are severely limited. In particular, the three-node model will be used if:

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1) The transient to be studied is expected to be more severe than the reference PCS startup,

2) The transient is driven mainly by power changes, or

3) The transient is expected to come close to any limitations on the rate of change of cladding temperature.

If the two-node model is to be used, the cladding heat capacity will be lumped with the coolant heat capacity.

b. Minimum Control Deadband

An analog computer study was performed to determine the minimum allowable control deadband necessary to prevent limit cycling in SNAP 8 systems. The limit cycling considered was an oscillation in the core outlet coolant temperature between the control deadband limit temperatures. The oscillation would be caused by the control drum stepping action. It was assumed that the long-term control stepping interval was greater than 150 sec so that only one positive and one negative control step would be present during one limit cycle (Figure 17).

Several parameters were varied during this study. The total temperature coefficient was

varied between -0.15 and -0.21 $^{\circ}\text{F}^{-1}$ and the control step worth was varied between 1 and 4 $^{\circ}\text{F}$ per step. The reactor system power level was considered to be constant at 450 kw. The flow was varied so that the heat exchanger ΔT changed between 140 and 200 $^{\circ}\text{F}$.

It was found that to avoid limit cycling, the minimum control deadband must be greater than the maximum increase of outlet coolant temperature resulting from a single control step.

The results of this study indicate that the minimum deadband required increases with an increase of control step reactivity worth and with an increase of core temperature difference. The required deadband decreases as the total prompt temperature coefficient of reactivity becomes more negative. Figures 18, 19, and 20 show these effects in detail. A minimum deadband width of 29 $^{\circ}\text{F}$ is required if (1) total temperature coefficient is -0.15 $^{\circ}\text{F}^{-1}$ (minimum value) and (2) the control drum step worth is 2.96 $^{\circ}\text{F}$ (0.38 degree/step at a differential drum worth of 7.8 $^{\circ}\text{F}$). A minimum deadband of 20.5 $^{\circ}\text{F}$ is required if the total temperature coefficient is close to the expected value (-0.21 $^{\circ}\text{F}^{-1}$) with the same control drum step worth of 2.96 $^{\circ}\text{F}$.

4-27-65

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Figure 17. Typical Limit Cycle

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Figure 18. Minimum Control Deadband Width, 200 $^{\circ}\text{F}$ Heat Exchanger ΔT

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Figure 19. Minimum Control Deadband Width, 170°F Heat Exchanger ΔT

Figure 20. Minimum Control Deadband Width, 140°F Heat Exchanger ΔT

It was concluded that the minimum deadband width should be expanded from 26°F* to 29°F unless the minimum total temperature coefficient can be expected to be $\sim -0.17\text{¢}/^\circ\text{F}$ or the maximum control step worth decreased to $\sim 2.7\text{¢}/\text{step}$. These control deadband widths assume 170°F ΔT at the heat exchanger and ~ 420 kwf reactor power. The control deadband width requirements should be similar at other reactor power levels.

4. Shielding Analysis

a. Neutron Scattering

The SNAP 8 reference lithium hydride shadow shield is designed to limit the neutron dose at the reference dose plane (40 feet from the center of the reactor) to 10^{11} nvt in 10,000 hours of

reactor operation at 600 kwf. The dimensions of the lithium hydride shield are designed to afford protection for the 20-foot-diameter payload from neutrons which scatter from the control drums and other reactor components as well as from neutrons which come directly from the core.

Portions of the NaK pipes and electrical cable assemblies which pass around the shadow shield provide an additional source of radiation at the payload due to neutron scattering from these exposed materials. An analysis was made of the neutron dose levels at the payload due to scattering from the pipes, electrical cables, and an assumed 100-mil 304 SS meteoroid shield covering the cables. The results, shown on Figure 21, indicate a peak fast neutron dose of 10^{13} nvt (10,000 hours at 600 kwf) at the radial edge of the 20-foot diameter payload decreasing to 10^{11} nvt at a distance of 5 feet from the horizontal centerline axis of the payload. The NaK pipes are the predominant source of scattering. Preliminary calculations indicate that gamma scattering will be only a small percentage of the gamma tolerance dose at the payload.

4-27-65 7568-02012
Figure 21. Scattered Dose at Payload from NaK Pipes, Electrical Wire Cables, and Cable Meteoroid Shield (nvt/10,000 hours)

*Johnson, C. E., "SNAP 8 Quarterly Progress Report, May Through July 1964," NAA-SR-9992 (SRD), September 22, 1964

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b. SNAP 8 Shutdown Radiation Levels

Calculations were performed to determine the shutdown radiation levels from an unshielded SNAP 8 reactor. The dose rate in rem/hr/kw from the SNAP 8 fission product inventory at a 10-foot separation distance as a function of reactor operating time and time after shutdown is shown on Figure 22. The source data was taken from Scoles' compilation.*

3-25-65

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Figure 22. Fission Product Gamma Dose Rate 10 Feet From Unshielded SNAP 8 Reactor

c. Manrated System Concept with a Shadow Shield

A proposed manrated SNAP 8 shadow shield design contains 16 canned (stainless steel) lithium hydride compartments located around the periphery of the top shield to facilitate the placement of NaK pipes and electrical cables within the top shadow shield. This was done

primarily to eliminate neutron scatter sources of the shadow shield envelope. Due to the possibility of hydrogen loss from the canned LiH sections as a result of meteoroid penetration (with a corresponding decrease in shielding effectiveness), an analysis of shield performance under these conditions was required. The relative importance of the increase in neutron flux was evaluated by considering the total number of neutrons penetrating the upper shield section after a hydrogen loss occurs and comparing this to the number of neutrons penetrating the upper section normally. This increase in neutron penetration, expressed as a ratio, is illustrated on Figure 23 as a function of the fraction of cans punctured ($n/16$) and the fraction of hydrogen (F_H) lost from the LiH in each can. These curves assume uniform hydrogen loss for each can punctured. For the limiting case of 100% hydrogen loss in all of the cans, neutron penetration is increased by a factor of 36; i. e., $S_2/S_1 = 36$ from Figure 23.

d. Manrated System Concept with a 4π Rendezvous Shield

Calculations have been performed to determine the dose rate and nuclear heating in a preliminary 4π shield design. The 4π portion of the shield was designed with a 6.5-cm-thick tungsten layer adjacent to the core. A 30-cm-thick layer of lithium hydride was placed external to the tungsten. At 600-kwt power level, this configuration would result in a dose of 93 rem/hr at 100 feet from the reactor in the reactor midplane, and a total of 32 kilowatts nuclear heat generation. The SNAP 8 reactor core was assumed to have a 3-1/2 inch beryllium oxide reflector. Dose and heating rate breakdowns are shown in Tables 9 and 10, respectively.

*J. F. Scoles, "Calculated Gamma-Ray Spectra from U^{235} Fission Products," NARF-58-37T, August 29, 1958

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4-20-65 7568-02013
 Figure 23. Effect of Meteoroid Puncture of SNAP 8 Manrated Shadow Shield Concept

TABLE 9
 DOSE RATES — 4π SHIELD DESIGN:
 6.5 cm W, 30 cm LiH

Radiation Source	Dose Rate (rem/hr)
Neutrons	19
Primary gammas	2
Secondary gammas	
Radial tungsten	56
Axial tungsten	16
Total	93

TABLE 10
 HEATING RATES — 4π SHIELD DESIGN:
 6.5 cm W, 30 cm LiH

Shield Region	Heating Rate (kw)
Radial region	
Tungsten	21
Lithium hydride	4.3
Axial region (top only)	
Tungsten	4.9
Lithium hydride	1.4
Total	32

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An alternate preliminary design using 10 cm of 316 SS and 30 cm of LiH in a similar configuration resulted in a dose rate of 81 rem/hr at 100 feet from the reactor midplane broken down as shown in Table 11. Heating rate calculations have not been made, but they should not differ appreciably from the data in Table 10.

(1) Nuclear Heating in a 4π -Shield Design Reflector

An analysis was made of the heating rates in a 4-inch BeO reflector around a SNAP 8 core surrounded by the 4π tungsten-LiH shield design previously described. The axially averaged heating rates are shown in Figure 24.

(2) NaK Activation

Estimates have been made of the induced activation of the NaK coolant in the manrated SNAP 8 concept. Due to core redesign (primarily the effect of increasing the fuel element spacing from 10 to 30 mils), the Na-24 activity will increase to ~440 curies (from 346) and the K-42 activity will increase to ~155 curies (from 120) at 600 kw. The total gamma radiation from the activated NaK was determined to be ~1 rem/hr at 100 feet from the reactor. This contribution does not affect the design of the 4π rendezvous shield.

TABLE 11
DOSE RATES - 4π SHIELD DESIGN:
10 cm 316 SS, 30 cm LiH

Radiation Source	Dose Rate (rem/hr)
Neutrons	10
Primary gammas	17
Secondary gammas	54
Total	81

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Figure 24. Nuclear Heating in Reflector,
 4π -Shield Design

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IV. FUEL ELEMENT DEVELOPMENT AND FABRICATION

A. INTRODUCTION AND SUMMARY

It is the objective of the fuel element engineering program to develop reliable fuel elements for the 600-kwt SNAP 8 reactor. The performance requirements established for the fuel elements are the following:

1) The peak power density of the fuel is approximately 5.0 kwt/ft of fuel-rod length at a peak central temperature of 1550°F for 10,000 hours. The peak fuel burnup is about 0.3 metal atom percent.

2) The hydrogen loss from the fuel elements must be less than 5% of that in the fuel alloy during the 10,000 hours the reactor is in operation. The hydrogen barrier coating and closure seal must withstand: all the pre-launch thermal and mechanical checkout; the acceleration, vibration, and shock due to launch; the thermal shock of approximately 150°F/min during startup; and a 1450°F peak cladding operating temperature for 10,000 hours in a total flux of 1.4×10^{21} nvt fast neutrons, where E_n is greater than 0.9 Mev. 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 99% reliability at the 50% confidence level of meeting these performance goals. Some redundant fuel elements are provided in the core so that loss of the hydrogen barrier in a few elements will not cause shutdown of the reactor.

Significant progress has been made in fuel element development and fabrication during this report period. Short-term mechanical property tests on the fuel material were completed.

Fuel-coating interaction studies continued with the S8DS-type fuel elements, which contain nickel-plated fuel rods, showing improved performance. Other fuel element performance tests continued to indicate the superiority of the SCB-1-coated S8DS fuel elements which incorporate the S8DRM-1 closure seal over the S8ER fuel elements.

Permeation kinetics studies of the SCB-1-coated S8DS cladding tube assemblies progressed significantly with the completion of initial tests in a newly designed out-of-pile experiment and completion of the irradiation phase of the NAA 77-2 and -3 inpile permeation kinetics tests.

Measurement of some thermophysical properties of SCB-1 and modified SCB coatings were completed. Substrate improvement effort continued with electronmicroprobe analyses of chromized Hastelloy-N S8DS cladding tubes and pre-oxidation experiments to improve coating adherence.

An unusually high number of S8DS fuel elements produced early in the report period were rejected due to high hydrogen permeation rates in acceptance testing. Review of fabrication procedures failed to reveal a cause for the apparent change in fuel element quality. However, a review of the tube chromizing operation revealed that wide variability in chromizing quality was being encountered. Based on this fact and the S8DS cup-plug chromizing experience reported in the prior report period, S8DS fuel element fabrication was suspended and effort was directed toward developing an improved, non-proprietary, tube-chromizing capability. The objectives of this effort are to salvage vendor-chromized tubes by rechromizing and to provide

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a nonproprietary chromizing capability for virgin cladding tubes.

Irradiation tests NAA 115-1 and NAA 117-1 continued operating at design conditions. Annealing tests of the NAA 116-1 fuel were completed.

B. FUEL ELEMENT DEVELOPMENT

1. Fuel Development

a. Fuel Nondestructive Test Development

Nondestructive tests are required to verify fuel rod hydrogen distribution in fabrication and to determine the relative hydrogen content and distribution in irradiated fuel.

Development has been concentrated on nondestructive tests to determine the relative content and axial distribution of hydrogen in fuel rods. The nuclear hydrogen analyzer* and eddy-current equipment were both evaluated during this report period and have produced encouraging results.

Three commercial eddy-current instruments, operating at frequencies from 50 to 100 kc, were utilized in a series of hydrogen determinations. Hydrided fuel rods with N_H values from 0 to 6.4 were scanned with all three instruments using both probe and annular (feed-through) sensing coils. It was found that a linear difference in eddy-current readings was obtained for changes in N_H except at alloy phase changes. At the delta to alpha-plus-delta phase boundary, the linear relationship between eddy-current readings and hydrogen content abruptly changed slope. It is believed that this is due to the fact that the electrical resistivity of the hydride changes abruptly at the phase boundary.

Other tests were conducted with the sample fuel rods placed inside coated cladding tubes.

After a minor adjustment in zero setting, the previously obtained bare-fuel-rod eddy-current readings were repeated with the fuel rods inside a coated cladding tube. The annular coil was particularly well suited for this test and would be readily adaptable for hot-cell use. More sensitive and stable instrumentation is being procured to permit a more critical evaluation of the eddy-current method of performing hydrogen determinations, particularly in the hot cell.

Since the eddy current readings may be adversely influenced by changes in resistivity resulting from radiation damage, the nuclear hydrogen analyzer is also being investigated as a means of performing accurate hydrogen measurements with highly radioactive fuel elements (10^4 R/hr). Axial hydrogen variations along the length of sample fuel rods have been successfully measured using a single Pu-Be neutron source. Experiments are being designed to determine whether the high gamma flux from the fuel elements presently used will saturate the presently used BF_3 neutron detectors in the NHA. If so, solid-state detectors may be investigated for this application.

b. Physical Property Measurements

(1) Fuel Mechanical Properties

The fuel tensile-creep program is continuing and all tests on 0.15 wt % carbon level fuels have been completed. Six tests were completed during this report period and the results are summarized in Table 12. Four of the six tests were on 0.15 wt % C fuels. The following conclusions were determined from the data collected this report period.

1) SNAP 8 fuel with 0.15 wt % C has better creep resistance than fuel with 0.30 wt % C at 1400°F under 4,000 psi stress.

*C. E. Johnson, "SNAP 8 Progress Report, November 1964 - January 1965," NAA-SR-10792 (CRD), March 15, 1965

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TABLE 12
SNAP FUEL CREEP TEST SUMMARY,
JANUARY THROUGH MARCH 1965

Test Number	Composition		Temperature (°F)	Stress (psi)	Average Minimum Creep Rate (in./in.-hr)
	H/Zr	C (wt %)			
23221	1.72	0.30	1400	4000	10.0×10^{-6}
23132	1.60	0.15	1500	3000	19.1×10^{-6}
22211	1.72	0.15	1300	4000	0.40×10^{-6}
22221	1.72	0.15	1400	4000	0.80×10^{-6}
22111	1.60	0.15	1300	4000	0.55×10^{-6}
21131	1.60	0.15	1500	4000	3.40×10^{-6}

2) SNAP 8 fuel (1.72 H/Zr, 0.15 wt % C) has better creep resistance than material with a lower hydrogen composition (1.60 H/Zr) at a stress of 4,000 psi and 1300°F.

3) SNAP 8 fuel (1.72 H/Zr, 0.15 wt % C), stressed at 4,000 psi exhibits twice the creep rate at 1400°F as it does at 1300°F.

Hydrogen loss from the fuel during irradiation might result in a gradual decrease in hydrogen content such that the delta-beta + delta phase boundary is crossed. If the creep resistance of the resulting beta material is low, severe fuel swelling might result due to trapped fission gases.

A study was initiated to determine the creep properties of the fuel as the hydrogen content was decreased across the delta-beta + delta phase boundary. A creep-test specimen was tested at 3,000 psi and 1300°F. Hydrogen was withdrawn from the system during testing to reduce the hydrogen content of the fuel.

The initial fuel composition was 1.47 H/Zr (delta phase) and was reduced in increments, past the delta-beta transformation of 1.37 H/Zr at 1300°F to 1.29 H/Zr. Results of these tests, shown in Figure 25, indicate that the creep re-

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Figure 25. Creep Test of SNAP Fuel as Hydrogen Composition is Reduced

sistance of the fuel is greatly reduced by the presence of the beta-hydride phase.

(2) Short-Term Mechanical Tests on Fuel

Studies on the phenomenon of hydrogen loss from the fuel during compression testing* have been completed. The results can be summarized as follows:

1) Hydrogen is lost from delta and epsilon hydrides in three distinct regions — elastic, inelastic, and post-rupture.

2) The total amount of hydrogen lost from delta and epsilon hydride materials is on the order of 0.02 to 0.03 wt %.

*C. E. Johnson, "SNAP 8 Progress Report, November 1964 - January 1965," NAA-SR-10792 (CRD), March 15, 1965

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3) The total amount of hydrogen released in the elastic region of epsilon hydride is inversely proportional to the strain rate (Figure 26).

4) The hydrogen release rate in the elastic region for epsilon-hydride material is directly proportional to strain (Figure 27).

5) Release of hydrogen in the elastic region for delta-hydride material is presumably a surface phenomenon. Release of hydrogen in the inelastic region for the same material is a crack-formation type of release rather than a strain function.

6) Material with 0.5 H/Zr composition does not release hydrogen during room-temperature compression testing.

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Figure 26. Hydrogen Release (Total) as a Function of Strain Rate in the Elastic Region for Epsilon Phase Fuel

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Figure 27. Hydrogen Release Rate as a Function of Strain in the Elastic Region for Epsilon Phase Fuel

A mechanism of hydrogen release in the elastic region for epsilon hydride fuel is postulated as follows: as twin boundaries move, the uncombined hydrogen in the twinning dislocation's tetrahedral sites tends to relocate. If twinning is highly oriented, the mean free path to a free surface is reduced considerably. The uncombined hydrogen in twinned dislocations forms a cascading type of relocation which expels hydrogen nearest a free surface.

Room-temperature compression tests were conducted on 0.5 H/Zr and 1.50 H/Zr fuel. A representative stress-strain compressive curve for 1.50 H/Zr material is presented in Figure 28. By comparing the compressive properties of this delta-hydride material with previously reported epsilon material (with comparable carbon contents), it is found that the total strain to failure is reduced by a factor of 8, and the Young's modulus and yield point are increased by a factor of 2 to 3 for delta-hydride material. A similar observation has been noted for delta and epsilon hydrides tested in tension. Fuel material with 0.5 H/Zr composition exhibited compressive properties similar to non-hydrided fuel.

Poisson's ratio and shear modulus (calculated) were determined for the delta-hydride of 0.15 wt % C fuel. A Poisson's ratio of 0.27 was obtained and a shear modulus of 5.35×10^6 psi was calculated. In previously reported data on

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Figure 28. Typical Room Temperature Compressive Stress-Strain Plot for SNAP Fuel Delta Hydride

epsilon-hydride fuel, Poisson's ratio was approximately 0.40 to 0.45 and the shear modulus was 2×10^6 to 3×10^6 psi. This difference can be attributed to the fact that delta-hydride fuel does not possess the highly twinned structure (leading to slip and twin boundary movement) which is found in epsilon-hydride fuel.

c. Fuel-Coating Interaction Studies

The test designed to establish the nickel-fuel diffusion characteristics as a function of time at 1400°F continued. The metallographic results of samples removed from test during this report period are summarized as follows:

(1) 1960-Hour Sample

There was no outward evidence of diffusion of the nickel plate into the fuel before sectioning of the sample. Approximately 80% of the transverse section metallographically examined was unbonded with no indication of a nickel intermetallic. Several localized areas showed a partial diffusion at the nickel-fuel interface with an adjacent void containing a nonmetallic material. One area exhibited an almost complete diffusion of the nickel into the fuel. A brittle interface was observed as indicated by a series of fine microcracks proceeding through the intermetallic phase.

(2) 2440-Hour Sample

Complete diffusion of the nickel plate into the fuel substrate was apparent by visual examination of the outside surface. Metallographic examination of the transverse sections confirmed this and showed that the nickel had diffused entirely into the fuel to form a 4-mil-thick intermetallic zone. Gross cracking at the diffusion zone — fuel substrate was quite extensive and more continuous than previous samples, as shown in Figure 29.

This work has been continuing as an extension of the original cursory study of the effect of a nickel barrier on the fuel-coating interaction. There is evidence of severe oxidation of the fuel in support of a fuel-coating interaction theory in areas beneath pinhole discontinuities in the nickel plating. Conversely, in areas where the nickel plating was uninterrupted, no reaction was found to occur. The fact is recognized, however, that the nickel and fuel are incompatible,

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Figure 29. 2440-Hour Nickel-Plated Fuel Compatibility Sample (250X). A 4-mil intermetallic zone with gross cracking along the diffusion zone-fuel substrate boundary is observed.

as demonstrated by the rapid diffusion characteristics of the nickel into the fuel, and the subsequent cracking phenomenon. In view of this, the problem of providing a barrier to the fuel-coating interaction will be approached from a basic materials aspect.

2. Hydrogen Barrier

a. Coating Tests

(1) Permeation Kinetics Studies

A new permeation kinetic test system for more closely defining temperature and pressure effects on hydrogen permeation through ceramic

barriers was assembled. The new system consists of two furnaces, all welded stainless-steel vacuum piping, mullite vacuum retorts, a 2-inch high-vacuum pumping station, a helium-hydrogen leak detector, and a new 3-point calibration system. A vacuum of $\sim 5 \times 10^{-5}$ torr has been achieved at the retort, which is as good or better than other existing systems. This new system also has the capability of testing membranes without having to compensate for temperature gradients or cold-end-cap permeation, and permits measuring the hydrogen flows in a reverse direction (metal to ceramic instead of ceramic to metal).

As reported previously, two distinct activation energies of diffusion have been observed for the SCB-1 hydrogen barrier material. Above 1370°F, $Q = 26,700$ cal/mole $\cdot$ °K and below 1370°F, $Q = 15,700$ cal/mole $\cdot$ °K. Data previously obtained above 1370°F were, therefore, influenced to some extent by the low-temperature activation energy, because of the temperature gradient which existed on the membrane. It has also been noted that some extraneous hydrogen has been present in most of the test results (over and above normal background rates measured with no internal hydrogen pressure). This extra hydrogen has been termed A_0 and shows up as the intercept in a L vs P plot (see Figure 30). All previous data interpretation on SCB-1 involved subtracting this A_0 term from the data.

Using the new kinetic system, a permeation test was performed on the isothermal portion of a membrane. The hydrogen permeating through the gradient portion and the cold end-cap was exhausted to an auxiliary vacuum station. This test was designed to eliminate the A_0 term. Figure 31 shows the data points taken with the new system on SCB-1 membrane No. 2310 at 1500°F when the pressure was increased in 1 atm steps

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Figure 30. Hydrogen Permeation Rate vs Pressure, SCB Membrane in Temperature Gradient Showing A_o Intercept

from 0 to 6 atm H_2 . The intercept is the measured background of the isothermal portion of the membrane at zero pressure and is in no way related to A_o . Thus, it appears that previous data interpretation, involving subtracting of the A_o term, was indeed valid.

(2) Membrane Endurance Tests

A cluster of three SCB-coated cladding tubes has been on endurance test for over 15,000 hours at 1500°F and 1 atm hydrogen pressure. The permeation rate has increased slightly in the last 2,000 hours, but the rate is still acceptable. The history of this endurance test is given in Figure 32.

A second cluster of five SCB-coated membranes also increased slightly in permeation rate as shown in Figure 33. These SCB membranes, processed with the short firing cycle for the SCB coating, have been under test for over 8,000 hours at 1500°F under 1 atm hydrogen pressure differential. Tests are now being made

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Figure 31. Hydrogen Permeation Rate vs Pressure, SCB-1 Membrane No. 2310, Isothermal Test in Double Vacuum System

to determine individual membrane permeation rates.

b. Coating Process Development

An unusually high number of S8DS fuel elements fabricated during this report period were rejected because of unacceptable hydrogen permeation rates in acceptance testing. Investigation of the fabrication processes (discussed in Section IV-C of this report) showed that wide variability in cladding tube chromizing was being encountered, resulting in variable adherence of the SCB coating on the cladding tube assemblies. As was the case with the S8DS cup-plug chromizing discussed in the prior progress report, variability within chromizing lots and lot-to-lot variability has been encountered with S8DS cladding

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Figure 32. Average Permeation Rate vs Time for Five Membranes Tested at 1500°F

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Figure 33. Average Permeation Rate vs Time for Five Membranes Tested at 1500°F (Fabricated with short firing cycle)

tubes. Both the cup-plugs and cladding tubes had been chromized by the same vendor who uses a proprietary process. Based on the cup-plug experience in the prior report period, effort was directed toward developing a tube chromizing capability at Atomics International. This program has two objectives which are:

- 1) To develop a process for salvaging vendor-chromized cladding tube assemblies by rechromizing, and

- 2) To develop a process for chromizing virgin cladding tube assemblies.

Initial efforts have been directed toward adapting the AI chromizing process developed for cup-plugs.

(1) Chromized Substrate Studies

Five randomly selected samples from various S8DS cladding tube lots which had been chromized by vendor B were analyzed with an electron microprobe to determine the concentrations of the principal elements in the chromized substrate (Cr, Ni, Mo) at and near the cladding surface. The sum of the concentrations of these three elements totals approximately 93% in Hastelloy-N. It was found that this total was obtained at varying distances from the surface of the chromized substrate in the range of 4 to 16 microns, and that this point corresponded approximately with the point where the chromium content began to decrease from the chromium-rich value in the chromized layer to the usual value for Hastelloy-N alloy. It was also found that the peak concentration of chromium in the chromium-rich layer varied in the

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range of 25 to 38%. Correlation of these observations with SCB adherence data and metallography obtained on comparison samples led to two conclusions:

1) Good SCB adherence was obtained with samples which exhibited the peak chromium concentration and the maximum in the summation of the concentrations of the three elements studied at a point very near the surface of the chromized substrate.

2) Samples for which the maximum in the summation of the concentrations of the three elements studied occurred some distance from the surface of the substrate probably possessed a porous surface layer.

(2) Coating Adherence Test

An empirical destructive test was developed for evaluating SCB-1 coating adherence to S8DS cladding tubes. The method essentially consists of (1) pressing a halved section of a coated tube flat and/or (2) flexing a split section of a coated tube inside out over a mandrel and measuring the amount of enamel removed. The amount of pressing or flexing required to remove 50% of the enamel from the surface of the metal is the criterion used to determine the adherence index of the coating.

Thirty-five S8DS coated cladding tubes from S8DS-type fuel elements, which had hydrogen permeation rates varying from a low of 0.18 to a high of over 22.00 cc(STP)/hr, were sectioned and tested to investigate a possible relationship between ceramic adherence and permeation rate. Adherence values were plotted against permeation rate, as shown in Figure 34. Curves were drawn through the points to indicate what is believed to represent average, maximum, and minimum relationships between adherence and permeation rate. There appeared to be four exceptions — Tubes 3392, 3405, 3526, and 3557 — which did not fall into the area enclosed by the

curves. This is now believed to result from variability in the chrome substrate over the length of the coated tube which had not been found at the time the data in Figure 34 were being obtained.

A preliminary conclusion from Figure 34 is that a coating adherence index minimum of 3 is required to obtain fuel elements which will meet the S8DS acceptance permeation test limit of 1.0 cc(STP)/hr.

The adherence index has not been used in reporting test results in other sections of this report because the concept of adherence index is based on limited data at this time. The chromizing development test results, therefore, are reported in percentage of coating remaining after flexes of the cladding sample.

(3) Chromizing Process Development

A series of chromizing runs were made at AI using the chromizing pack developed for S8DS cup-plug chromizing. A group of sub-length, open-end S8DS cladding tubes was chromized for 90 minutes at 2135°F, and the samples were subsequently coated with SCB-1 to S8DS specifications and tested to determine coating adherence and chromizing thickness. The results are presented in Table 13. From these data, it may be observed that some variation occurred in the chromized layer thickness and in the ceramic adherence. It was also found that pre-oxidation of these chromized tubes resulted in decreased adherence as contrasted with the findings with the vendor-chromized material. The adherence of the coatings was generally good for those samples which were not pre-oxidized before coating.

Another group of sublength, open-end S8DS cladding tubes was chromized at 2135°F for 45 minutes. The time was reduced from the 90 minutes used in the previous runs to obtain a thinner chromium layer which would meet the

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Figure 34. Permeation Rate vs Adherence Index

0.005 to 0.0010-in. thickness specification for S8DS. In addition, the effect of electropolishing the tubes prior to chromizing was evaluated. Results of these tests, shown in Table 14, indicated that electropolishing the cladding tubes prior to chromizing resulted in improved coating adherence.

Rechromizing of vendor-chromized S8DS cladding tubes has shown promising results. Three of the vendor-chromized S8DS tubes were cut into three pieces which were rechromized at 2135°F for 45 minutes. Results of this rechromizing test are shown in Table 15. Removal of most of the vendor's chromized layer by electro-

polishing prior to rechromizing resulted in improved adherence of the SCB-1 coating.

Two full-scale S8DS cladding tubes were successfully chromized and two others were successfully rechromized. Each of the four membranes was electropolished prior to the chromizing (or rechromizing) treatment. The chromizing treatment parameters were 2100°F for 30 minutes and the chromized layer thicknesses ranged from 0.63 to 0.94 mils. One chromized cladding tube and one rechromized cladding tube were sectioned and the short sections were coated and tested. Adherence test results for samples taken from several positions along the

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TABLE 13
DIFFUSION LAYER THICKNESS AND ADHERENCE TEST RESULTS FOR
SHORT TUBES AFTER CHROMIZING AT 2135°F FOR 90 MINUTES

Specimen Number	Diffusion Layer Thickness (mils)	Post-Chromizing Treatment	Coating Remaining After Flexing (%)			
			1 flex	2 flexes	3 flexes	n flexes
1	1.6	None	not coated			
2-1	1.9	Pre-oxidized	60	30	10	-
2-2	1.9	None	95	90	80	50 (7 flexes)
3-1	1.7	None	100	100	100	100 (11 flexes)
3-2	NA	Pre-oxidized	0	-	-	-
7-1	1.1	None	100	95	90	50 (11 flexes)
7-2	1.3	None	90	80	50	-
7-3	1.0	None	100	100	90	50 (9 flexes)
8-1	0.8	None	100	100	100	90 (11 flexes)
8-2	0.6	None	50	40	20	-
8-3	0.8	None	50	45	40	-

TABLE 14
DIFFUSION LAYER THICKNESS AND ADHERENCE TEST RESULTS FOR
SHORT TUBES AFTER CHROMIZING AT 2135°F FOR 45 MINUTES

Specimen Number	Diffusion Layer Thickness (mils)	Pre-Chromizing Treatment	Coating Remaining After Flexing (%)			
			1 flex	2 flexes	3 flexes	n flexes
12-1	1.0	None	60	40	10	-
12-2	1.0	Electropolished*	90	85	85	80 (11 flexes)
13-1	NM	None	100	100	100	70 (11 flexes)
13-2	NM	Electropolished*	100	100	80	55 (11 flexes)
14-1	0.8	None	3	-	-	-
14-2	1.0	Electropolished*	80	70	50	-
15-1	0.8	None	3	-	-	-
15-2	0.8	Electropolished*	40	-	-	-

*Samples electropolished for two minutes at 6 volts.
NM = not measured.

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TABLE 15

ADHERENCE TEST RESULTS ILLUSTRATING THE EFFECT OF ELECTROPOLISHING AND RECHROMIZING OF MEMBRANES CHROMIZED BY THE VENDOR

Sample Number	"As-Received" Chromized Layer Thickness (mils)	Electropolish Time (min)	Chromized Layer Thickness After Electropolishing (mils)	Chromized Layer Thickness After Rechromizing (mils)	Adherence Test Results, Coating Remaining After Flexing (%)			
					1 flex	2 flexes	3 flexes	11 flexes
A-1	0.3	-	-	-	0	-	-	-
B-1	0.47	-	-	-	0	-	-	-
C-1	0.16	-	-	-	0	-	-	-
A-2	0.3	1.5	0.3	-	0	-	-	-
B-2	0.47	1.5	0.4	-	100	100	100	65
C-2	0.16	1.5	0.13	-	0	-	-	-
A-3	0.3	1.5	0.2	0.95	95	95	95	80
B-3	0.47	1.5	0.3	1.40	100	100	100	90
C-3	0.16	1.5	0.1	0.80	100	100	100	80

length of these are reported in Table 16. It was found that some variation in adherence occurred along the length of the cladding tubes. The two remaining cladding tubes (one chromized and one rechromized) were straightened, coated, and assembled to S8DS specifications for testing.

TABLE 16

ADHERENCE DATA FOR A FULL-LENGTH MEMBRANE AFTER CHROMIZING AT 2100°F FOR 30 MINUTES

Specimen Number	Adherence Test Results, Coating Remaining After Flexing (%)			
	1 flex	2 flexes	3 flexes	n flexes
1 (bottom)	60	45	-	-
2	100	95	90	85 (11 flexes)
3	85	70	50	-
4	10	5	5	-
5	15	10	5	-
6	75	70	60	50 (5 flexes)

c. Nondestructive Test Development Related to Coating, Burnable Poison Analysis

Refinements in irradiation, counting, and data reduction techniques were accomplished to improve the precision of the nuclear poison analyzer developed for measuring the total and in-

cremental poison in coated cladding tubes. A new counting device was installed on the sample traversing mechanism to indicate to the reactor operator that the cladding tubes are moving at a uniform rate through the reactor. Bare (uncoated) tubes have been interspersed between the coated cladding tubes to act as neutron dosimeters for checking the uniformity of the irradiation. New probe-type detectors are being designed to couple recently developed small-diameter photomultiplier tubes directly to the scintillation crystal. This will eliminate the lucite light-pipe and should improve the resolution of the detector and reduce undesired background radiation. The decay correction factors for both the Sm^{153} and Mo^{99} in the cladding tube have been recalculated to provide more accurate data reduction. All of these operational improvements have been successfully tested with production coated S8DS cladding tubes.

d. Advanced Coating Development

The objective of this program is to develop improved hydrogen permeation barrier materials based on the SNAP 8 reference barrier

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material (SCB). Three approaches to this problem are being studied:

1) Crystalline oxides are being introduced into the slip during the frit milling operation. The effectiveness of this approach depends on the relative hydrogen permeation properties of the mill addition oxide and of the matrix material, on the volume fraction of the crystalline oxide that can be added to the slip, and on any change in the hydrogen permeation properties of the matrix glass that may result from the dissolution of the crystalline oxide in the glass during the firing operation.

2) The composition of the frit is being adjusted in an effort to suppress crystallization of the coating material and thereby extend the long-time high-temperature use capability of the coating.

3) Methods for improving the adherence of the coating to the cladding are being developed by controlled oxidation of the cladding surface before application of the coating material.

Studies of the effect of mill additions of ceria (CeO_2) and of alumina (Al_2O_3) to SCB-1 coating material on the properties and behavior of the coatings were continued. Data on the softening points, coefficients of thermal expansion, and permeation test results for some of the materials studied are summarized in Figure 35 and in Table 17. As reported in the prior progress report,* it was found that increases in the coefficient of thermal expansion were obtained with additions to SCB-1 of 10% and 20% CeO_2 and of 10% Al_2O_3 . The softening temperatures for these compositions have also been increased from 1485°F to temperatures in the range of 1800 to 1900°F. The results of the permeation

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Figure 35. Softening Temperature of SCB-1 Material as a Function of CeO_2 Mill Additions

tests completed to date with these materials do not reflect corresponding changes. One membrane which was coated with SCB-1 plus 20% CeO_2 exhibited a hydrogen permeation rate of 2×10^{-2} cc(STP)/hr-cm² at 1400°F and a hydrogen pressure of 2 atm, while a second membrane with this coating exhibited a permeation rate of 2×10^{-3} cc(STP)/hr-cm² under similar conditions. A third membrane which was coated with SCB-1 plus 10% Al_2O_3 had a hydrogen permeation rate of 1×10^{-3} cc(STP)/hr-cm² at 1400°F and 2 atm hydrogen pressure. These values may be compared with an average value for SCB-1 coated membranes of 1.6×10^{-3} cc(STP)/hr-cm².

From these observations, it appears that significant solution of the mill addition oxides in the SCB-1 glass occurs during the firing operation with resulting changes in some properties of the coating such as the coefficient of thermal expansion. No significant decrease has been observed, however, in the hydrogen permeation rates for membranes coated with SCB-1 containing mill additions of crystalline oxides. Additional tests of membranes coated with these materials are in progress to verify the preliminary observations cited above.

*C. E. Johnson, "SNAP 8 Progress Report, November 1964 - January 1965," NAA-SR-10792 (CRD), March 15, 1965

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TABLE 17
PROPERTIES OF HYDROGEN BARRIER COATING MATERIALS
CONTAINING VARIOUS MILL ADDITIONS

Basic Coating Material	Mill Addition		Firing Temperature (°F)	Softening Point* (°F)	Coefficient of Thermal Expansion ($\times 10^6 / ^\circ\text{F}$)	Hydrogen Permeation Rate at 1400°F and 2 atm Pressure [cc(STP)/cm ² -hr]
	Amount (wt %)	Oxide				
SCB-1	-	-	2100	1485	3.1	$1.6 \times 10^{-3}\dagger$
SCB-1	5	CeO ₂	2100	1500	3.0	-
SCB-1	20	CeO ₂	2100	1835	3.9	Test specimen A. 2×10^{-2} Test specimen B. 2×10^{-3}
SCB-1	40	CeO ₂	2100	1900	4.1	-
SCB-1	3	Al ₂ O ₃	2100	-	-	-
SCB-1	10	Al ₂ O ₃	2100	>1900	3.3	Test specimen A. 1×10^{-3}
SCB-1	30	Al ₂ O ₃	2100	-	-	-
SCB-2	-	-	2200	1500	3.6	Test specimen A. 4×10^{-4}
SCB-2	5	CeO ₂	2200	-	-	-
SCB-2	20	CeO ₂	2200	>1700	4.6	-
SCB-2	40	CeO ₂	2200	-	-	-

*Softening point is defined as the temperature at the maximum of the thermal expansion curve.

†Average of many test specimens.

Work on the reformulation of the basic SCB-1 SNAP 8 coating material to provide a more stable material as well as one with lower characteristic hydrogen permeation rates has produced two compositions designated SCB-2 and SCB-23. These compositions are based on the SCB-1 formulation and are given in Table 18. For the SCB-2 formulation, the TiO₂ was re-

TABLE 18
COMPOSITION OF SCB-2 AND SCB-23

Compound	Coating (wt %)		
	SCB-1	SCB-2	SCB-23
SiO ₂	46.7	44	43.7
BaO	32.7	45	39.5
Al ₂ O ₃	7.5	11	11.0
TiO ₂	13.1	-	-
BaF ₂	-	-	6.0

moved from the SCB-1 composition and the BaO/SiO₂ ratio was adjusted from 0.7 to a value of approximately 1.0. The SCB-23 composition was obtained by replacing a part of the BaO in SCB-2 with BaF₂ to obtain a decreased firing temperature while maintaining the barium ion concentration. The softening points, coefficients of thermal expansion, and devitrification temperatures for these compositions are presented in Table 19. The effect of a range of BaF₂ additions to SCB-2 glass on the flow of the glass relative to SCB-1 is illustrated graphically in Figure 36. It was found that a 6 wt % addition of BaF₂ to SCB-1 resulted in the material having approximately the same flow properties as those of SCB-1 at 2100°F. Since this is the firing temperature for the SCB-1 coatings for SNAP 8 fuel elements and it was desired to test the glass under application conditions approximately equivalent to those previously studied with

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SCB-1, the 6% BaF₂ composition was selected for further evaluation.

Data on the softening temperature and coefficient of thermal expansion for SCB-2 glass and modifications of this glass obtained with additions of BaF₂ are graphically illustrated in Figures 37 and 38. It was found that the BaF₂ additions resulted in reductions in both the softening temperature and the coefficient of thermal

expansion over the range of compositions studied. The coefficient of thermal expansion for SCB-23, containing 6% BaF₂, was significantly greater than that of SCB-1 (3.5×10^{-6} in/in°F for SCB-23; 3.1×10^{-6} in/in°F for SCB-1). This observation indicates that a substantial reduction in the shear stress might be expected at the interface between the coating

TABLE 19
PROPERTY DATA FOR SELECTED
COATING MATERIALS

Coating Material	Firing Temperature (°F)	Softening Temperature (°F)	Coefficient of Thermal Expansion ($\times 10^{-6}/^{\circ}\text{F}$)	Devitrification After Annealing 5 Hours in a 1400 to 2200°F Temperature Gradient*	
				Amount	Temperature Range (°F)
SCB-1	2100	1485	3.1	Extensive†	1625 to 2200
SCB-2	2200	1500	3.6	None	-
SCB-23	2100	1400	3.5	Incipient§	1900 to 2100

*Specimens were clear glass fragments measuring about 1/4 in. across.

†The rim of surface crystals was approximately 6 mils thick at 1900°F.

§The surface was faintly cloudy.

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Figure 36. Relative Fusion Flow of SCB-2 Material at 2100°F as a Function of BaF₂ Concentration

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Figure 37. Softening Temperature of SCB-2 Material as a Function of BaF₂ Substituted for BaO

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Figure 38. Thermal Expansion Coefficient of SCB-2 Material as a Function of BaF₂ Substitution for BaO

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and cladding of an SCB-23-coated fuel element. Also, the reduced softening point should result in reduced stresses in the glass at elevated temperatures, as the glass could more readily adjust itself by viscous flow to the dimensions of the cladding. The detailed thermal expansion curve for SCB-23 glass is illustrated in Figure 39.

4-2-65 7568-02027
Figure 39. Thermal Expansion of SCB-23 Glass as a Function of Temperature

Three full-size SNAP 8 fuel cladding membranes were successfully coated with SCB-23 after a pre-coating oxidation treatment of 90 minutes at 1700°F. Tests of these membranes were initiated in the permeation test program with SCB-2-coated membranes and membranes coated with SCB-1 plus selected mill additions. It was found that favorable permeation test results were obtained with a membrane coated with SCB-2 which yielded a permeation rate of 4×10^{-4} cc(STP)/hr-cm² at 1400°F and a hydrogen pressure of 2 atm.

A series of tests was initiated to evaluate the effects of vacuum outgassing and preoxidation of chromized cladding tubes on the adherence of the SCB-1 coating. Three S8DS cladding tubes, chromized by a vendor, were each cut into six test samples. Two samples of each tube were

processed in selected vacuum outgas and/or pre-oxidation treatments. All of the tube samples were then coated with SCB-1 ceramic, using the S8DS coating process, and each was destructively examined to evaluate coating adherence. The experiment is summarized in Table 20 along with the data obtained. This experiment was repeated with very similar results. It was found that pre-oxidation improved the coating adherence on Tubes B and C which showed very poor adherence after only vacuum outgassing. On the other hand, Tube A showed excellent adherence regardless of the pretreatment. These tests indicate a wide variability in response to pretreatments among Tubes A, B, and C which were selected from vendor-chromized S8DS cladding tube lots.

3. Closures

a. Closure Development and Testing

Five S8DRM-1 closure seals have accrued 14,200 hours in the thermal endurance test. The cluster permeation rate is 0.47 cc(STP)/hr at 1300°F and 2 atm for an average of 0.09 cc(STP)/hr/sample. This rate represents an increase of less than 0.1 cc(STP)/hr from the rate reported at the end of the previous report period.

Five S8DS closure seals have now reached 13,200 hours of thermal endurance. The cluster permeation rate of 0.54 cc(STP)/hr at 1400°F and 2 atm is essentially the same as reported at the end of the previous quarter. The average rate is 0.11 cc(STP)/hr/sample.

It is interesting to note that both groups have essentially the same average permeation rate, though the S8DS group is running 100°F hotter.

Room- and elevated-temperature burst tests were performed to obtain weld-joint mechanical properties. Thirty-three samples of the edge-fusion design, described in the previous report,

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TABLE 20
INFLUENCE OF HEAT TREATMENT ON THE ADHERENCE OF SCB COATING
TO CHROMIZED HASTELLOY-N

Treatment	Tube Designation	Specimen	Coating Remaining After Flexing (%)			
			1 flex	2 flexes	3 flexes	11 flexes
Pre-oxidation only at 1700°F for 30 min, 110 cc/min air with 32°F dew point.	A	1	100	100	100	100
	A	2	100	100	100	95
	B	1	97	50	30	-
	B	2	100	95	95	65
	C	1	95	60	35	-
	C	2	50	35	15	-
Vacuum outgassed at 2140°F for 90 min, and fast-cooled. No pre-oxidation.	A	1	100	100	100	65
	A	2	97	90	70	40
						(5 flexes)
	B	1	0	-	-	-
	B	2	0	-	-	-
	C	1	0	-	-	-
Vacuum outgassed at 2140°F for 90 min, and fast cooled, then pre-oxidized at 1700°F for 30 min, 110 cc/min air with 32°F dew point.	A	1	100	100	100	100
	A	2	100	100	100	100
	B	1	0	-	-	-
	B	2	100	100	100	100
	C	1	90	80	40	-
	C	2	0	-	-	-

Note: All samples were taken from tubes that had been chromized by a vendor.

and nine samples of the S8DS design lip weld were tested and analysis of the data was begun. Long-term creep testing of the edge-fusion weld joint was begun at test conditions of 1400°F and 1250-psi hoop stress.

b. Nondestructive Test Development Related to Closure

A radiographic technique was developed for measuring weld penetration in the edge-fusion weld configuration which is shown in Figure 40. This technique which permits tangential exposures to be made without undercutting (darkening of the edges of the sample). The correction form produced radiographic images of the weld which resembled metallographic cross-sections. Figure 41 is an example of radiographs of the weld, without and with the correction form.

Measurements of the radiographs indicated that the average weld penetration was 0.020 inch.

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Figure 40. Edge-Fusion Design

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a. ANGLE-BEAM EXPOSURE

b. CORRECTION FORM SHOWS WELD CLEARLY AND
ALLOWS MEASUREMENT OF WELD PENETRATION

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Figure 41. Radiographs of Edge-Fusion Welds
(Tonal values are reversed from original
radiographs)

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4. Performance Tests

a. Developmental Tests

Testing of S8DS developmental fuel elements continued through the report period. The basic tests being conducted to investigate fuel element performance characteristics include:

1) Thermal soak — 400°F/hr heating to 1300°F, 10-hour dwell, cooling to 150°F at 400°F/hr (5 cycles). This test simulates the slow thermal cycles occurring during reactor checkout.

2) Ramp heating — 300°F/min heating to 850°F, 150°F/min heating to 1400°F, cooling to 200°F at 50°F/min (50 cycles). This test simulates repeated ground test reactor startup.

3) Thermal cycling — cycling about 1400°F from 1350°F to 1450°F at 50°F/min (300 cycles). This cycling simulates temperature variations occurring during reactor operation.

4) Thermal endurance — 10,000 hours of continuous operation at isothermal temperature (1300°F, 1350°F, or 1400°F).

Permeation tests, conducted at 1400°F (isothermal), are used to determine the effect of various inputs on the SCB ceramic hydrogen barrier of the fuel elements. In addition to the above tests, the effects of various levels of vibration are being investigated.

Results of tests performed through this report period are summarized in the following paragraphs.

(1) Control Tests

The 20 fuel elements, previously subjected to eight permeation tests, have completed ramp heating and thermal cycling. The multiple permeation tests caused the average 1400°F permeation rate to approximately double, i.e., increase from 0.38 to 0.70 cc(STP)/hr. Ramp

heating of the fuel elements caused the average permeation rate to increase to 1.0 cc(STP)/hr. This increase is equal to that sustained after multiple permeation tests. Subsequent thermal cycling inputs caused no damage. In fact, the average permeation rate decreased to 0.91 cc(STP)/hr. The data for these fuel elements are shown in Table 21.

(2) Thermal Test Series

A group of 14 S8DS developmental fuel elements were given thermal soak, ramp heating, and thermal cycling tests. The data are shown in Table 22. Ramp heating and thermal cycling inputs increased the average permeation rate of these elements. The overall effect of the three tests plus permeation testing was to approximately double the average permeation rate, i.e., to increase the average from 0.41 to 0.74 cc(STP)/hr. This is generally the same response which was obtained for the fuel elements in the control test (c.f. Table 21).

As in the control tests, a wide variation in response was exhibited. Some fuel elements remained stable while others showed large increases in permeation rate. Evaluation of the data, relative to manufacturing data, revealed no correlation. It appears that thermal inputs generally cause an upward trend in permeation rate, with widely random response to individual inputs. This is true for fuel elements given the control tests as well as those in the thermal test series. Considering the 34 elements under test (control plus thermal tests), 15 were delivered on July 15, 1964 and 19 were delivered September 23, 1964. Grouping the elements by delivery date, however, showed no large difference in performance.

(3) Thermal Endurance

The 34 fuel elements from control and thermal tests are now in thermal endurance test

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TABLE 21
EFFECTS OF RAMP HEATING AND THERMAL CYCLING
AFTER MULTIPLE PERMEATION TESTS ON
20 ELEMENTS

Element Number	Delivery Date	Hydrogen Permeation at 1400°F cc(STP)/hr				
		Permeation Test			Post Ramp Heat Test	Post Thermal Cycle Test
		1	2	8		
N-2487	7-15-64	0.28	0.49	0.62	0.66	0.68
N-2488	7-15-64	0.30	0.33	0.42	0.59	0.47
N-2530	7-15-64	0.27	0.28	0.43	2.4	3.4
N-2532	7-15-64	0.33	0.34	0.47	2.6	1.3
N-2539	7-15-64	0.31	0.26	0.29	0.36	0.36
N-2636	9-23-64	0.37	0.80	0.82	0.70	0.76
N-2678	9-23-64	0.37	0.57	1.2	1.7	1.4
N-2796	9-23-64	0.51	0.62	1.3	1.2	1.3
N-2806	9-23-64	0.56	0.80	1.5	1.6	0.78
N-2812	9-23-64	0.42	0.57	0.85	0.90	0.85
N-2910	9-23-64	0.88	0.90	1.1	1.1	0.98
N-2912	9-23-64	0.74	1.0	1.3	1.2	1.2
N-2994	9-23-64	0.34	0.36	0.55	1.2	1.3
N-3218	9-23-64	0.28	0.20	0.47	0.52	0.35
N-3220	9-23-64	0.26	0.33	0.42	0.43	0.42
N-3227	9-23-64	0.19	0.56	0.41	0.42	0.23
N-3233	9-23-64	0.33	0.79	0.50	1.2	0.92
N-3236	9-23-64	0.20	0.20	0.29	0.30	0.25
N-3239	9-23-64	0.23	0.24	0.31	0.31	0.37
N-3242	9-23-64	0.43	0.42	0.86	0.77	0.96
Average		0.38	0.50	0.70	1.01	0.91
Average Δ		+0.12	+0.20	+0.31	-0.10	

TABLE 22
RESULTS OF THERMAL TESTS ON ELEMENTS WITH
ACCEPTANCE PERMEATION TESTS

Element Number	Delivery Date	Hydrogen Permeation at 1400°F [cc(STP)/hr]				
		Accept- ance	Pre- Thermal Soak	Post Thermal Soak	Post 50 Ramp Heat	Post 300 Thermal Cycles
N-2490	7-15-64	0.34	0.38	0.40	0.59	0.54
N-2491	7-15-64	0.33	0.50	0.35	0.32	0.80
N-2492	7-15-64	0.53	0.56	0.51	0.93	0.85
N-2506	7-15-64	0.25	0.34	0.32	0.43	0.24
N-2513	7-15-64	0.36	0.44	0.50	0.46	0.71
N-2540	7-15-64	0.43	0.48	0.46	0.63	0.66
N-2543	7-15-64	0.30	0.28	0.26	0.22	0.27
N-2544	7-15-64	0.33	0.38	0.32	0.63	0.48
N-2548	7-15-64	0.37	0.44	0.50	0.74	0.49
N-2561	7-15-64	0.47	0.51	0.37	0.62	0.54
N-2599	9-23-64	0.31	0.40	0.45	0.43	1.03
N-2826	9-23-64	0.43	0.47	0.53	0.51	1.12
N-2665	9-23-64	0.39	0.49	0.45	0.47	0.46
N-2613	9-23-64	0.90	0.93	0.88	1.01	2.22
Average		0.41	0.47	0.45	0.57	0.74
Average Δ		+0.06	-0.02	+0.12	+0.17	

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furnaces. Operating temperatures of 1300 and 1350°F are being used to determine if performance is temperature-dependent. The fuel elements were loaded randomly in single and triple element retorts. Approximately 700 hours of operation have been accrued. Test data to date at the endurance operating temperatures are shown in Table 23. The permeation rates of all fuel elements have stabilized. Current data from thermal endurance test are generally in good agreement with pre-endurance data. There are several retorts in the 1350°F furnace which have rates higher than the pre-endurance values. Only one retort operating at 1300°F has a rate significantly higher than the pre-endurance rate. This may indicate a temperature effect but it is not reflected in the average rates. Quite possi-

bly this is merely coincidental and is part of the random performance exhibited during pre-endurance tests.

Thermal endurance testing was continued on three groups of developmental elements. The elements were described and the pre-endurance data were tabulated in the previous progress report. Briefly, these elements received only pre-endurance permeation tests prior to beginning thermal endurance test. One group of 5 elements has a spacer assembly between the fuel rod and the ceramic at the blind end of each element. These elements are undergoing thermal endurance in single element retorts. The endurance data are presented in Figure 42 which shows hydrogen permeation as a function of time at 1400°F. The endurance data were fit to the equation form:

$$\ln L = K_1 + K_2 t$$

where

L = permeation rate [cc(STP)/hr],

K_1 and K_2 are constants, and

t = time at 1400°F (hr).

These equations are plotted with the respective permeation time curves of Figure 42.

Thermal endurance data of a second group of five elements are shown in Figure 43. These five developmental fuel elements were fabricated to S8DS fuel element specifications which were current at the time of manufacture. After 2900 hours at 1400°F, the elements were removed from the furnaces and subjected to a permeation test at 1400°F, five ramp heat cycles, and another permeation test. The resulting data are listed in Table 24.

The third group includes three S8DS design fuel elements, except that the fuel rods were nickelplated to inhibit fuel-coating interaction. The endurance data for these elements are shown in Figure 44.

TABLE 23

THERMAL ENDURANCE PERMEATION
DATA FOR 34 S8DS DEVELOPMENTAL
FUEL ELEMENTS

Element Number	Hydrogen Permeation cc(STP)/hr			
	Pre-Endurance	170 hr	325 hr	670 hr
	At 1350°F			
N-2513, 2506, 2910	1.0	1.3	1.2	1.2
N-2544, 3233, 2912	1.5	1.4	1.4	1.3
N-2491, 2599, 2812	1.4	1.5	1.4	1.4
N-2539, 2543, 2806	0.76	1.3	1.2	1.2
N-2826	0.89	0.70	0.67	0.63
N-3220	0.21	0.16	0.16	0.16
N-2613	1.4	1.2	0.98	0.98
N-2487	0.24	0.52	0.49	0.50
N-2665	0.26	0.36	0.35	0.36
Average	0.45	0.49	0.46	0.46
	At 1300°F			
N-2492, 2678, 3236	0.82	0.65	0.64	0.61
N-2540, 2548, 3239	0.42	1.0	1.0	1.0
N-2532, 2561, 2636	0.79	0.62	0.65	0.62
N-2490, 2796, 3218	0.60	1.3	0.63	0.61
N-3242	0.37	0.21	0.20	0.20
N-2530	0.97	0.44	0.40	0.46
N-2488	0.15	0.15	-	0.16
N-2994	0.37	0.27	0.28	0.28
N-3227	0.06	0.07	0.06	0.08
Average	0.27	0.28	0.24	0.24

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Figure 42. Hydrogen Permeation of Fuel Elements with Spacers During 1400°F Endurance Test

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Figure 43. Hydrogen Permeation of Five S8DS Reference Design Fuel Elements During 1400°F Thermal Endurance

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Figure 44. Hydrogen Permeation of Fuel Elements with Nickel-Plated Fuel Rods During 1400°F Endurance Test

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TABLE 24
EFFECT OF RAMP HEATING DURING
INTERRUPTED THERMAL
ENDURANCE

Element Number	Hydrogen Permeation at 1400°F [cc(STP)/hr]			
	Prior to Unloading	After Unloading	After 5 Ramp Heats	After Re-Loading
N-2415	-	1.25	1.41	-
N-2424	-	1.08	0.84	-
N-2533	-	0.61	0.59	-
	1.93	2.94	2.84	2.2
N-2410	-	0.74	0.57	-
N-2521	-	0.37	0.29	-
	0.81	1.11	0.86	0.95

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Figure 45. Hydrogen Loss During 1400°F Thermal Endurance

TABLE 25
VIBRATION TEST EFFECTS ON S8DS FUEL ELEMENT PERMEATION

Element Number	Hydrogen Permeation at 1400°F [cc(STP)/hr]								
	Pre-Vibration			Post-Vibration 100% Level			Post-Vibration 200% Level		Post 2nd Vibration 200% Level
	Test 1	Test 2	Test 3	Test 1	Test 2	Test 3	Test 1	Test 2	Test 1
N-2576	0.38	0.38	-	1.1	0.78	0.85	13.5	9.9	-
N-2598	0.53	0.61	0.65	0.89	0.81	0.91	6.2	5.4	-
N-2638	0.56	0.61	-	1.1	0.93	1.0	3.5	3.7	-
N-3241	0.41	0.38	-	0.39	0.45	0.42	0.74	-	6.7
N-2550	0.36	0.40	-	0.80	0.50	0.60	7.1	5.9	-
N-2553	0.30	0.33	-	0.60	0.57	0.69	0.53	0.57	7.47
N-2554	0.28	0.33	-	0.66	0.53	0.61	2.4	2.4	-
Average	0.40	0.43	-	0.79	0.65	0.73	4.8	4.6	-
N-2813	0.41	0.63	0.50	-	-	-	0.93	-	-
N-2902	0.58	0.61	0.62	-	-	-	1.3	-	-
N-3219	0.60	0.68	0.43	-	-	-	>9.0	-	-
N-3223	0.33	0.67	0.68	-	-	-	5.2	-	-
N-2602	0.63	0.57	-	-	-	-	9.1	-	-
Average	0.51	0.63	0.56	-	-	-	>5.1	-	-

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In order to compare the relative performance of the three groups of elements undergoing thermal endurance, it is necessary to consider variations in fuel dissociation pressure and/or test temperature. One way of making this type of comparison is by use of the parameter K_{BLD}^* , which is a measure of the effectiveness of the hydrogen barrier. K_{BLD} as a function of 1400°F endurance time was calculated for each of the three groups of S8DS developmental fuel elements. Since K_{BLD} is directly related to the relative worth of the ceramic barrier, these calculations showed that fuel element N-2521 (S8DS reference design) and the three elements containing nickel-plated fuel rods had the best ceramic hydrogen barrier. This is indicated by Figure 45, which shows that the three elements with nickel-plated fuel rods have had the least hydrogen loss to date.

(4) Vibration Tests

A two-phase vibration test program was concluded. During the first phase, fuel element response as a function of input level and core tank position was determined. Results of this investigation were reported in the previous progress report.^{*} Fuel element response was random in nature with significant chatter at several frequency levels, generally below 300 cps. As much as 800 g's of shock was indicated by accelerometers mounted on the fuel elements.

During the second phase of the program, the effect of vibration on the hydrogen barrier was investigated. As shown by the data listed in Table 25, the average hydrogen permeation of 7 elements almost doubled after being subjected to vibration at 100% of the S8DS fuel element de-

sign level. This increase may have been due to the vibration test, the subsequent permeation test, or a combination of the two tests. Since the next two permeation tests failed to show an additional increase in permeation, the inference is that vibration testing at 100% design level followed by permeation testing apparently damaged the hydrogen barrier of the fuel elements, but not catastrophically. In fact, the average permeation rate after design level vibration was less than 0.80 cc(STP)/hr. Vibration testing at 200% of the design level caused considerably more damage to the hydrogen barrier as shown by the data in Table 25. Additional tests are planned to investigate the effect of vibration inputs between 100% and 200% of the design levels.

b. Qualification Tests

The progress of the S8ER and S8DS design fuel elements in their respective test programs[†] is discussed below. Thirty-one of the original thirty-three S8ER fuel elements have completed qualification test. The data are summarized in Tables 26 and 27.

The results of pre-endurance testing S8ER design fuel elements were discussed in a previous report.[§] Pre-endurance data indicate, in general, significant increases in permeation, while endurance data indicate just the opposite. Post-endurance permeation rates are generally higher than acceptance and final endurance rates (Table 26). Figure 46 shows the curves of average permeation rate vs endurance time for the 31 S8ER elements. The curve shows a continuing decreasing trend in permeation as a function of time at temperature. The average hydrogen loss for these 31 S8ER fuel elements after 10,000 hours at 1400°F is 13.5%. At 600-kwt

*C. E. Johnson, "SNAP 8 Progress Report, November 1964 - January 1965," NAA-SR-10792 (CRD), March 15, 1965

†C. E. Johnson, "SNAP 8 Quarterly Progress Report, July-September 1963," NAA-SR-8992 (SRD), November 15, 1963, pp 65-67

§C. E. Johnson, "SNAP 8 Quarterly Progress Report, October 1963-January 1964," NAA-SR-9392 (SRD), March 16, 1964, pp 95-102

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reactor operating conditions, these fuel elements would be expected to lose 5.8% of their hydrogen during 10,000 hours of operation.

Table 27 lists S8ER fuel element dimensional data at acceptance, prior to, and after 10,000 hours of thermal endurance. A statistical evalu-

ation of the data shows, in general, that the length, maximum diameter, and maximum total indicated reading (TIR) of these fuel elements increased during performance tests. This increase is attributed to the relief of the residual stresses incurred during production straightening of the cladding tubes.

TABLE 26
HYDROGEN PERMEATION RATES OF
31 QUALIFICATION-TESTED
FUEL ELEMENTS

Element Number	Hydrogen Permeation at 1400° F cc(STP)/hr			
	Acceptance	Pre-Endurance	Final Endurance*	Post-Thermal Endurance
E-821-N	0.11	1.2	0.29	0.85
E-760-N	0.05	1.3	0.36	0.80
E-804-N	0.18	0.93	0.26	0.49
E-811-N	0.58	3.8	0.39	0.42
E-841-N	0.09	1.3	0.26	0.53
E-886-N	0.25	3.4	0.28	1.0
E-897-N	0.30	1.4	0.32	0.57
E-802-N	0.46	1.7	0.22	0.86
E-810-N	0.27	2.4	0.34	0.39
E-880-N	0.07	1.3	0.25	0.72
E-888-N	0.31	2.2	0.25	0.40
Average	0.24	1.9	0.29	0.66
E-814-N	0.13	2.2	0.32	0.50
E-887-N	0.10	1.5	0.26	0.29
E-858-N	0.17	3.0	0.32	0.63
E-870-N	0.25	2.9	0.34	0.37
E-918-N	0.84	2.3	0.22	0.53
E-951-N	1.42	2.5	0.23	1.1
E-962-N	0.09	3.7	0.38	0.79
E-939-N	0.33	2.6	0.81	0.34
E-960-N	0.15	1.3	0.26	0.39
E-916-N	0.50	3.2	0.18	0.25
Average	0.40	2.5	0.28	0.51
E-861-N	0.42	1.8	0.34	1.2
E-884-N	1.13	2.2	0.37	0.63
E-972-N	0.32	3.2	0.39	0.59
E-963-N	0.23	2.3	0.37	0.50
E-992-N	0.56	3.2	0.39	0.49
E-1012-N	0.79	2.3	0.39	0.31
E-1017-N	0.94	3.4	0.44	0.66
E-987-N	1.31	2.4	0.34	1.1
E-1007-N	0.12	1.2	0.58	0.32
E-1047-N	0.28	1.6	0.32	0.33
Average	0.61	2.4	0.39	0.62

*Average of last 3 measurements during thermal endurance

Figure 47 shows average values of K_{BLD} of the 31 S8ER fuel elements vs accumulative hours

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Figure 46. Average Permeation of 31 S8ER Fuel Elements as a Function of Endurance Time

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Figure 47. Average K_{BLD} vs Time at 1400°F

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at temperature, where K_{BLD} is a measure of quality. Increasing values of K_{BLD} indicate that the ceramic is becoming a less efficient hydrogen barrier. A "perfect" fuel element would have the value of $K_{BLD} = 0$. Examination of Figure 47 shows that, at acceptance, the hydrogen barrier of these 31 elements was highly efficient. However, pre-endurance inputs and early endurance testing at 1400°F damaged these elements. Further thermal endurance tests resulted in a decreasing K_{BLD} indicating an improvement of the efficiency of the hydrogen barrier.

Fifteen S8DS design fuel elements have been permeation-tested prior to thermal soak. These permeation rates are compared with acceptance rates in Table 28. One fuel element (N-3553) was obviously damaged — apparently between tests — because a third test verified the pre-thermal soak rate. The average permeation rate, even excluding this element, shows the same upward trend observed for earlier S8DS fuel elements. Tests of these elements have been stopped pending finalized production specification for S8DS fuel elements. These elements are to be disassembled to utilize the fuel rods in the elements made to the final specification.

Tests of the 31 S8ER fuel elements after qualification are discussed below:

(1) S8ER Program — 11 S8ER Elements

a) Five elements received the following post-qualification test inputs:

- 1) Five cycles of ramp heating,
- 2) Post-ramp heat permeation testing at 1400°F,
- 3) An additional 1000 hours of thermal endurance, and
- 4) Post-endurance permeation testing at 1400°F.

The data from acceptance through post-1000-hour endurance are shown in Table 29. Post-ramp heat average permeation rates are higher than the post-10,000-hour endurance rates by a factor of two, indicating hydrogen barrier degradation. Post-1000-hour endurance permeation data further substantiate that the hydrogen barrier was damaged.

b) Five elements have undergone the following post-endurance inputs:

- 1) Permeation testing at various temperature,
- 2) Permeation scanning at 1400°F, and
- 3) Post-scan permeation testing at 1400°F.

The fuel element permeation data from acceptance through post-scan, are shown in Table 30. The permeation-temperature data for these fuel elements are shown in Figures 48 through 52. The post-endurance 1400°F permeation rates are also shown. With the exception of two 1450°F points, the data correlate well to an equation form:

$$\ln L = a + b (10^4/T)$$

where

L = hydrogen permeation [cc(STP)/hr],
 T = temperature (°R), and
 a and b = constants.

The two 1450°F points which do not correlate well are higher than would be predicted from the data at lower temperatures. This is an indication of either nonlinearity above 1400°F or hydrogen barrier damage. The values of a and b , found by the least squares method, are shown on the graphs. The average value of b is -3.56. Permeation data from enriched S8ER fuel elements (1200 to 1400°F) which received no prior test inputs gave an average value of $b = -3.55$.

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TABLE 27
AVERAGE DIMENSIONS OF S8ER FUEL ELEMENTS

	Dimensions (in.)		
	S8ER Test Levels, 11 Elements	S8DS Test Levels, 10 Elements	S8FS Test Levels, 10 Elements
Length			
Acceptance	14.467	14.466	14.468
Pre-endurance	14.468	14.471	14.472
Post-endurance	14.475	14.473	14.471
Minimum Diameter			
Acceptance	0.560	0.560	0.560
Pre-endurance	0.561	0.560	0.560
Post-endurance	0.561	0.560	0.560
Maximum Diameter			
Acceptance	0.562	0.561	0.561
Pre-endurance	0.562	0.562	0.561
Post-endurance	0.562	0.562	0.563
Total Indicated Reading(TIR)			
Acceptance	0.009	0.009	0.009
Pre-endurance	0.011	0.009	0.008
Post-endurance	0.011	0.014	0.019

TABLE 28
PERMEATION DATA OF S8DS
DESIGN FUEL ELEMENTS

Element Number	Hydrogen Permeation Rate at 1400°F [cc(STP)/hr]	
	Accept- ance	Pre- Thermal Soak
N-2701	0.21	0.15
N-2789	0.24	0.67
N-2808	0.21	0.37
N-2882	0.37	0.36
N-3525	0.23	0.19
N-3527	0.22	0.28
N-3528	0.36	0.48
N-3529	0.36	0.48
N-3534	0.40	0.56
N-3541	0.30	0.31
N-3544	0.30	0.39
N-3551	0.26	0.35
N-3553	0.36	2.05*
N-3555	0.32	0.42
N-3558	0.24	0.25
Average (all elements)	0.29	0.49
Average (excluding 3553)	0.29	0.34

*This element has been permeation tested in two different furnaces with the same result.

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TABLE 29
PERMEATION RATES OF FUEL ELEMENTS WHICH RECEIVED
POST-QUALIFICATION RAMP HEATS

Element Number	Hydrogen Permeation at 1400°F [cc(STP)/hr]					
	Acceptance	Pre-Endurance	Final* Endurance	Post Endurance	Post Ramp Heat	Post Endurance
E-821-N	0.11	1.2	0.29	0.85	1.6	0.99
E-880-N	0.07	1.3	0.25	0.72	1.0	1.0
E-897-N	0.30	1.4	0.32	0.57	1.3	1.2
E-760-N	0.05	1.3	0.36	0.80	1.3	1.2
E-802-N	0.46	1.7	0.22	0.86	1.5	0.89
Average	0.20	1.4	0.29	0.76	1.3	1.1

Average of last 3 readings

TABLE 30
DATA ON FUEL ELEMENTS PERMEATION TESTED AT VARIOUS
TEMPERATURES AND IN THE SCANNER

Element Number	Hydrogen Permeation Rate [cc(STP)/hr]												
	Acceptance 1400°F	Pre Endurance 1400°F	Final Endurance* 1400°F	Post Endurance 1400°F	1150°F	1250°F	1350°F	1450°F	Total Scan 1400°F	1-1/8-in. Closure End	12-in. Closure End	1-1/8-in. Blind End	Post Scan 1400°F
E-810-N	0.27	2.4	0.34	0.39	0.01	0.07	0.20	0.73	0.72	0.50	0.06	0.16	0.66
E-811-N	0.58	3.8	0.39	0.42	0.04	0.11	0.27	1.05	1.71	0.93	0.25	0.53	0.95
E-841-N	0.09	1.3	0.26	0.53	0.03	0.11	0.30	0.99	1.76	1.38	0.09	0.29	0.76
E-881-N	0.31	2.2	0.25	0.40	0.02	0.07	0.21	1.13	1.19	0.90	0.04	0.25	0.97
E-886-N	0.25	3.4	0.28	1.00	0.04	0.20	0.53	1.51	1.79	1.38	0.15	0.26	1.3
Average	0.30	2.6	0.31	0.55	0.03	0.11	0.30	1.08	1.41	1.02	0.12	0.30	0.94

*Average of final 3 readings

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Figure 48. Permeation-Temperature Relationship, Element E-810-N

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Figure 49. Permeation-Temperature Relationship, Element E-811-N

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Figure 50. Permeation-Temperature Relationship, Element E-841-N

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Figure 51. Permeation-Temperature Relationship, Element E-881-N

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Figure 52. Permeation-Temperature Relationship, Element E-886-N

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Figure 53. Average K_{BLD} vs Time at 1300°F

TABLE 31
DATA ON ELEMENTS WHICH WERE POST-QUALIFICATION
ENDURANCE TESTED

Element Number	Hydrogen Permeation at 1400°F [cc(STP)/hr]					
	Input	Acceptance	Pre-Endurance	Final Endurance *	Post Endurance 10,000 hr	Post Endurance 1,000 hr
E-804-N	S8ER	0.18	0.92	0.26	0.49	0.53
E-870-N	S8DS	0.25	2.90	0.34	0.37	0.72
E-918-N	S8DS	0.89	2.3	0.22	0.53	0.72
E-951-N	S8DS	1.4	2.5	0.23	1.1	0.96
Average		0.69	2.1	0.26	0.63	0.73

*Average of last 3 readings

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The permeation scan data of Table 30 disclose that typically about 72% of the hydrogen loss comes from the closure end and 20% from the blind end of a fuel element. Only fuel element E-811-N showed permeation from the center 12 inches to be greater than 10% of the total permeation. Post-scan permeation rates are lower than the total scan rates but higher than the post-endurance rates. The cause of this variation has not been determined.

Four S8ER fuel elements have completed an additional 1000 hours of endurance following qualification test. One fuel element received the S8ER qualification inputs and three received S8DS level qualification inputs. The data of these fuel elements, shown in Table 31, indicate that the thermal cycling associated with the additional permeation and endurance tests, or the endurance test itself, damaged the fuel elements by causing higher permeation rates.

(2) S8DS Program — 12 S8ER Elements

- a) Three fuel elements received post-qualification inputs mentioned previously above.
- b) Two fuel elements are undergoing an additional 1000 hours of endurance at 1400°F.
- c) Two fuel elements are undergoing gradient temperature permeation testing.
- d) Three elements will be tested in the permeation scanner.
- e) Two elements increased in permeation rate above the defined failure limit of 4.00 cc(STP)/hr at 1400°F during pre-endurance inputs and were removed from the program before endurance testing. These elements completed destructive analysis. The results of destructive analysis were discussed in a previous report.*

(3) S8FS Program — 10 S8ER Elements

- a) Four elements are undergoing an additional 1000 hours of thermal endurance testing at 1400°F.
- b) Two elements are being permeation tested with gradient temperature.
- c) The remaining four elements will be tested in the permeation scanner.

(4) Conclusions

- a) Hydrogen permeation of S8ER fuel elements decreased exponentially as a function of time during endurance test. Post-endurance permeation rates are generally higher than the final endurance permeation rates.
- b) Additional post qualification tests generally damage the hydrogen barrier.
- c) Permeation scan data show the greatest hydrogen loss (72%) from the closure end.
- d) The growth evidenced in the fuel elements during performance testing is attributed to the relief of residual stresses incurred during production straightening of the cladding tubes.

c. Environmental Tests

Nineteen S8ER fuel elements have completed 9800 hours of continuous operation at 1300°F. Operation continued to be stable, with permeation rates decreasing as a function of decreasing hydrogen dissociation pressure. This is indicated by relatively stable values of K_{BLD} . The average K_{BLD} values of the 19 fuel elements are shown in Figure 53 as a function of time. Based on this average K_{BLD} curve, no real change in barrier quality is indicated for the past 6000 hours of operation.

*C. E. Johnson, "Progress Report, SNAP 8, August-October 1964," NAA-SR-10492 (SRD), December 23, 1964, p 68

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d. S8DRM-1 Fuel Element Tests

The S8DRM-1 fuel element test program has been described in detail in a previous quarterly report.* Twenty-seven fuel elements have completed pre-endurance test and 24 fuel elements have accrued over 1000 hours of endurance test at 1300°F. Of the 27 elements completing the pre-endurance tests, 12 fuel elements were manufactured with AI-8763D ceramic hydrogen barrier and 15 were made with SCB ceramic. There are 12 of each type now undergoing endurance test. The pre-endurance test data are given in Tables 32 and 33.

The AI-8763D-coated fuel elements exhibited lower hydrogen loss rates through thermal soak, vibration, shock, acceleration, ramp heat, and thermal cycle than did the SCB-coated fuel elements. A statistical analysis of the test data showed, however, that there was little difference between the two types of fuel elements in response to these test inputs. This was not the case after six AI-8763D-coated and seven SCB-coated elements were tested in the S8DRM-1. After S8DRM-1 inputs, the AI-8763D-coated elements were damaged to the point that the SCB-coated elements had a lower average

TABLE 32
PERMEATION DATA OF ELEMENTS NOT SUBJECTED TO
S8DRM-1 SYSTEM TESTS

Element Number	Hydrogen Permeation [cc(STP)/hr]							
	Ceramic	H/Zr _{eff}	Acceptance (1400°F)	Pre-Thermal Soak (1300°F)	Post Thermal Soak (1300°F)	Post Vibration, Shock, and Acceleration (1300°F)	Post Ramp Heat (1300°F)	Post Thermal Cycle (1300°F)
E-1041N-4	AI-8763D	1.706	0.16	0.02	0.05	0.05	0.07	0.06
E-1044N-4	AI-8763D	1.702	0.14	0.02	0.02	0.03	0.12	0.10
E-1063N-4	AI-8763D	1.696	0.52	0.12	0.10	0.13	0.22	0.64
E-1064N-4	AI-8763D	1.696	0.30	0.04	0.08	0.08	0.06	0.12
E-1066N-4	AI-8763D	1.711	0.16	0.02	0.03	0.02	0.03	0.10
E-1070N-4	AI-8763D	1.704	0.14	0.02	0.02	0.02	0.03	0.07
Average	AI-8763D	1.702	0.24	0.04	0.05	0.06	0.09	0.18
Normalized Average*	AI-8763D	1.764	0.14	0.07	0.09	0.10	0.15	0.29
N-2358-4	SCB	1.748	0.37	0.13	0.77	1.05	1.39	1.23
N-2359-4	SCB	1.789	0.47	0.23	0.21	0.10	0.22	0.14
N-2360-4	SCB	1.770	0.45	0.14	0.07	0.11	0.29	0.24
N-2361-4	SCB	1.736	0.38	0.08	0.11	0.11	0.21	0.17
N-2400-4	SCB	1.794	0.39	0.12	0.18	0.36	0.52	0.67
N-2439-4	SCB	1.774	0.41	0.13	0.16	0.60	1.10	1.05
N-2442-4	SCB	1.745	0.30	0.08	0.10	0.09	0.27	0.14
N-2482-4	SCB	1.744	0.28	0.04	0.08	0.18	0.19	0.12
Average	SCB	1.762	0.38	0.12	0.21	0.32	0.52	0.47
Normalized Average*	SCB	1.764	0.13	0.12	0.21	0.31	0.51	0.46

*Normalized to 1300°F and 1 atm internal hydrogen pressure

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, October 1963-January 1964," NAA-SR-9392 (SRD), March 16, 1964

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TABLE 33
PERMEATION DATA OF FUEL ELEMENTS SUBJECTED TO S8DRM-1 REACTOR TESTS

Element Number	Ceramic	H/Zr _{eff}	Hydrogen Permeation [cc(STP)/hr]								
			Acceptance (1400°F)	Pre-Thermal Soak (1300°F)	Post Thermal Soak (1300°F)	Post Vibration Shock, and Acceleration (1300°F)	Post Ramp Heat (1300°F)	Post Thermal Cycle (1300°F)	Post S8DRM-1 Startup Reliability (1300°F)	Post S8DRM-1 Thermal Cycle (1300°F)	Post S8DRM-1 Vibration and Shock (1300°F)
E1023N-4	AI-8763D	1.686	0.24	0.02	0.03	0.05	0.07	0.03	0.09	-	0.14
E1028N-4	AI-8763D	1.710	0.30	0.04	0.02	0.04	0.05	0.03	0.16	-	0.08
E1089N-4	AI-8763D	1.703	0.17	0.02	0.02	0.04	0.02	0.02	0.06	-	0.03
E1119N-4	AI-8763D	1.712	0.21	0.03	0.05	0.02	0.02	0.02	0.07	-	0.13
E1129N-4	AI-8763D	1.698	0.22	0.04	0.05	0.02	0.04	0.03	0.10	-	0.17
R1133N-4	AI-8763D	1.714	0.12	0.02	0.05	0.02	0.03	0.04	0.66	-	1.31
Average	AI-8763D	1.704	0.21	0.03	0.04	0.03	0.04	0.03	0.19	-	0.31
Normalized Average*	AI-8763D	1.764	0.09	0.05	0.07	0.05	0.07	0.05	0.30	-	0.48
N2436-4	SCB	1.737	0.28	0.05	0.07	0.09	0.12	-	0.14	0.15	0.18
N2444-4	SCB	1.743	0.26	0.06	0.08	0.15	0.15	-	0.22	0.21	0.23
N2451-4	SCB	1.742	0.34	0.07	0.06	0.12	0.11	-	0.12	0.12	0.27
N2471-4	SCB	1.772	0.28	0.04	0.12	0.17	0.33	-	0.41	0.31	0.27
N2483-4	SCB	1.780	0.46	0.08	0.13	0.17	0.15	-	0.16	0.21	0.18
N2484-4	SCB	1.733	0.26	0.09	0.10	0.10	0.09	-	0.16	0.15	0.18
N2486-4	SCB	1.726	0.37	0.10	0.09	0.21	0.16	-	0.20	0.22	0.36
Average	SCB	1.748	0.32	0.07	0.09	0.14	0.16	-	0.20	0.20	0.24
Normalized Average*	SCB	1.764	0.17	0.08	0.10	0.16	0.18	-	0.22	0.22	0.27

*Normalized to 1300°F and 1 atm internal hydrogen pressure

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hydrogen loss rate. In fact, the hydrogen permeation rate of one of the AI-8763D-coated fuel elements exceeded the defined failure limit of 1.0 cc(STP)/hr at 1300°F.

The average data from the 1300°F thermal endurance tests of 24 S8DRM-1 fuel elements are shown in Table 34, and Figure 54 is a graph of average K_{BLD} vs endurance time. To date,

TABLE 34
S8DRM-1 ENDURANCE TEST RESULTS

Time (hr)	Average Hydrogen Permeation at 1300°F [cc(STP)/hr]	
	12 Elements, AI-8763D Ceramic	12 Elements, SCB Ceramic
Pre- Endurance	0.24	0.21
120	0.30	0.24
168	0.28	0.25
312	0.26	0.21
480	0.33	0.24
648	0.51	0.26
816	0.42	0.25
960	0.38	0.23
1152	0.39	0.24
1320	0.35	0.20

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Figure 54. S8DRM-1 Fuel Element
Performance During Endurance Test

the SCB-coated S8DRM-1 fuel elements have performed better during endurance test than have the fuel elements coated with AI-8763D ceramic. The hydrogen permeation rates are becoming stable, and the test temperature will be increased to 1400°F for the remainder of the 10,000-hour test period.

A comparison of S8DRM-1 and S8ER fuel element test performance was one objective of this test program. The difference between the S8ER fuel elements and the AI-8763D-coated S8DRM-1 fuel elements is in the closure seal design. The cup and cap of the S8ER design was changed to a one-piece cup-plug for the S8DRM-1. A comparison of a group of five S8ER fuel elements which had test inputs nearly identical to the inputs of six of the AI-8763D-coated S8DRM-1 fuel elements is shown in Figure 55. The performance of the S8DRM-1 fuel elements exceeded that of the S8ER fuel elements in that there was a significant difference between hydrogen permeation rates throughout all tests. In comparing responses to test input, the S8DRM-1 fuel elements showed only slight increases in hydrogen permeation up to thermal cycling, whereas the S8ER fuel elements showed significant increases in permeation rate after each input. The response to thermal cycling was similar for both groups of fuel elements. The data show a significant increase in fuel element integrity of the S8DRM-1 fuel element as compared to the S8ER fuel element on the basis of test results of these 11 fuel elements.

e. Special Studies

Tests to determine the amount of hydrogen cross flow which occurs in the two-zone scanner were completed. In these tests, known hydrogen flow rates were fed into the retort on each side of the orificed baffle. The diametrical gap between the fuel element and the orifice was simulated using a stainless-steel rod. For each of three diametrical gaps investigated, the cross flow was found to be a linear function of the difference in flows measured from each side of the

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Figure 55. Comparison of S8DRM-1 and S8ER Fuel Element Performance

baffle. This is shown graphically in Figure 56 for 2.0-, 5.5-, and 10.0-mil diametrical gaps. Since cross flow goes from the high to the low flow side of the baffle, measured flow rates are corrected by subtracting the cross flow from the low rate and adding this quantity to the high rate. The data form a diverging family of curves, related to diametrical gap. The orifice in the two-zone scanner baffle is sized to permit a maximum clearance of 5.5 mils for S8DS fuel elements.

Thirteen S8DS pilot fuel elements, fabricated in the prior report period, were subjected to low-level vibration inputs to determine if this input would be useful as an acceptance test. Vibration input levels to the test core vessel are listed in Table 35. After completing the

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Figure 56. Scanner Crossflow at 1400°F

TABLE 35
LOW-LEVEL VIBRATION
INPUT LEVELS*

Frequency (cps)	Magnitudes
5 to 20	3/16 in. double amplitude
20 to 400	3.4 g
400 to 2000	5.0 g

*Constant octave sweep rate requiring 5 minutes along each of 3 mutually perpendicular axes.

vibration test, the elements were permeation-tested, ramp-heated, and permeation-tested again. The results indicate that the low-level vibration test can be useful in identifying fuel elements which pass 1400°F acceptance permeation tests but would become damaged during subsequent performance tests or reactor operation. Three elements showed only a slight increase in permeation after low-level vibration,

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Figure 57. Permeation-Temperature Relationship, S8DS Element N-2560

Figure 58. Permeation-Temperature Relationship, S8DS Element N-2558

Figure 59. Permeation-Temperature Relationship, S8DS Element N-2671

Figure 60. Permeation-Temperature Relationship, S8DS Element N-2693

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whereas the remaining ten elements increased about 60% in permeation. Furthermore, the same three elements remained relatively undamaged during subsequent testing, as shown by the data listed in Table 36.

TABLE 36
PERMEATION TEST RESULTS FOR
13 S8DS PILOT FUEL ELEMENTS

Element Number	Hydrogen Permeation at 1400°F [cc(STP)/hr]					
	Pre-Vibration		Post-Vibration		Post Ramp Heat	
	1st Test	2nd Test	1st Test	2nd Test	1st Test	2nd Test
N-2724	0.31	0.38	0.43	0.39	0.50	0.52
N-2733	0.30	0.30	0.39	0.36	0.46	0.48
N-2941	0.29	0.27	0.28	0.25	0.30	0.35
Average	0.30	0.32	0.37	0.33	0.42	0.45
N-2712	0.37	0.43	0.94	0.78	0.87	1.02
N-2748	0.44	0.40	0.72	0.62	0.64	0.76
N-2753	0.51	0.65	1.03	1.06	0.92	1.10
N-2786	0.31	0.37	0.59	0.52	0.59	0.65
N-2788	0.30	0.34	0.98	0.82	1.21	1.17
N-2792	0.31	0.63	0.84	0.59	0.70	0.83
N-2921	0.33	0.43	0.69	0.57	0.62	0.66
N-2992	0.33	0.39	0.61	0.56	0.72	0.76
N-3222	0.40	0.50	0.70	0.66	0.72	0.79
N-3246	0.74	0.86	1.03	0.85	1.10	1.33
Average	0.40	0.50	0.81	0.70	0.81	0.91

Four S8DS reference design fuel elements are undergoing tests to determine if permeation at isothermal conditions can be correlated with permeation at gradient temperature conditions. In order to minimize the effect of possible thermal aging, the elements were tested at 1400°F for about 600 hours. Without removing the elements from the furnaces, a series of tests were begun whereby the element temperature was varied and the resulting permeation measured. To date, the elements have been subjected to two temperature conditions:

- 1) Isothermal
- 2) Closure end cold — the closure end was 20°F to 200°F cooler than the remainder of

the element. The resulting data are given in Figures 57 through 60 which show permeation as a function of inverse average temperature. A least-squares fit was made of the data to equation form

$$\ln L = a + b (10^4/T)$$

where

L = hydrogen permeation [cc(STP)/hr],
T = temperature (°R), and
a and b = constants.

The individual values of a and b determined for each element are shown in Figure 60. The average value of b for these 4 S8DS developmental fuel elements is -3.76. This value compares well with the average of -3.55 found for fuel elements of the S8ER design.

Additional tests are in progress to determine if the same correlation is applicable when the blind end of the fuel element is relatively cold.

5. Irradiation Tests

a. Materials Irradiation

(1) NAA-116 Series, Low-Temperature Fuel Materials Irradiation

The objective of the low-temperature fuel materials irradiations is to provide irradiated fuel samples for use in the correlation of the results of post-irradiation annealing tests with the results of high-temperature inpile tests. The purpose of the annealing tests is to study the swelling behavior of the fuel as a function of annealing temperature and time and as a function of the metallurgical parameters of the materials. These data are expected to yield information on the breakaway swelling temperature of the fuel.

Annealing studies of the fuel samples from the NAA-116-1 capsule were concluded. In these studies the annealing temperature was held constant at 1300°F and selected samples were annealed for various times up to an accumulated annealing time of 211 hours. For these

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tests the fuel samples were sealed inside heavy-wall quartz ampoules. The fuel samples were removed from the ampoules after selected annealing times, for density measurements. Each time a quartz ampoule was broken to remove a fuel sample, the contained gases were collected and analyzed to obtain data on the release of radioactive krypton.

Data on the densities and the density changes obtained for fuel samples from the NAA-116-1 capsule are presented in Tables 37 and 38. It was found that a major fraction of the density decreases which occurred during irradiation was recovered as a result of the annealing treatments. The results of hydrogen analyses of the annealed samples indicated that no significant loss of hydrogen occurred during the annealing thus eliminating hydrogen composition change as the mechanism responsible for the annealing induced density increases.

Dye penetrant tests were also performed on the annealed specimens. Only one of the four specimens (Number 20) exhibited cracks after annealing. This indicates that the measurements of the apparent densities of the fuel specimens were probably not significantly influenced by liquid entering the cracks during the measurements. The severity of the cracks in Specimen Number 20 did increase during the last annealing treatment. The cause of this behavior has not been determined.

TABLE 37

DENSITY DATA FOR FUEL SAMPLES FROM THE NAA-116-1 CAPSULE*

Sample Number	Density of Sample (gm/cm ³)						
	Pre-Irradiation	Post-Irradiation	1 hr	6 hr	22 hr	90 hr	211 hr
14	6.287	6.209	6.224	6.230	6.226	6.238	6.259
19†	6.546	6.451	-	6.484	6.472	6.475	6.520
20§	6.184	6.091	6.123	6.146	6.128	6.138	6.176
22†	6.580	6.400	6.435	6.422	6.438	6.428	6.483

*Illustrating the effect of annealing at 1300°F on fuel density (0.1 metal at. % burnup)

†20 wt % U material

§Crack was found after 90 hr annealing

The observations made to date on the fuel samples from the NAA-116-1 capsule are consistent with the hypothesis that the large decrease in density which occurred as a result of irradiation was due to the formation of internal microcracks. The mechanism for the formation of the microcracks has not been identified. The density increases which occurred during annealing of the irradiation samples are believed to be due to the collapse of these microcracks because of a stress-relief mechanism.

(2) NAA-77 Series, Inpile Permeation Tests

The NAA-77 capsule series was designed to measure hydrogen permeation rates of barrier materials during reactor operation. Advantages of this approach are: (a) the rate of radiation damage to the hydrogen permeation barrier material, if any, can be determined, and (b) the effect of the ionizing radiation field on the permeation rate can be measured.

Initial studies of the effect of reactor radiation on the performance of SCB-1 permeation barrier material is nearing completion. Four capsules (NAA-77-1 through NAA-77-4) containing SCB-1-coated membranes have been

TABLE 38

DENSITY DATA CHANGE FOR FUEL SAMPLES FROM THE NAA-116-1 CAPSULE*

Sample Number	Density Changes ($\Delta\rho_2/\rho_{post}$) [§] Due to Irradiation (%)	Density Changes, $\Delta\rho_1/\rho_{post}$, [†] After Annealing at 1300°F for Total Times Indicated (%)				
		1 hr	6 hr	22 hr	90 hr	211 hr
14	1.26	0.242	0.338	0.274	0.468	0.805
19	1.47	-	0.511	0.326	0.372	1.07
20**	1.53	0.526	0.904	0.608	0.772	1.40
22	2.81	0.547	0.344	0.594	0.438	1.30

*Illustrating the effect of annealing at 1300°F on fuel density (0.1 metal at. % burnup)

$$\dagger \frac{\Delta\rho_1}{\rho_{post}} = \frac{\rho_{annealed} - \rho_{post-irradiated}}{\rho_{post-irradiated}} \times 100$$

$$\S \frac{\Delta\rho_2}{\rho_{post}} = \frac{\rho_{pre-irradiated} - \rho_{post-irradiated}}{\rho_{post-irradiated}} \times 100$$

**Crack was found after 90-hr anneal

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TABLE 39
NAA-77 SERIES IRRADIATION HISTORY

Capsule	Irradiation		Membrane Temperature (°F)	Total Time (hr)	Neutron Dosage (nvt)		Fission Fragment Dosage/Element Inch
	Started	Ended			Calculated	Measured	
NAA-77-1	8-63	6-64	1500	4,343	2.56×10^{21}	1.54×10^{21}	-
NAA-77-2	3-64	2-65	1300	4,638	1.39×10^{21}	not available	1.29×10^{18}
NAA-77-3	3-64	2-65	1500	4,638	1.39×10^{21}	not available	1.31×10^{18}
NAA-77-4	6-64	6-64	1700	100	$\sim 6 \times 10^{19}$ 13×10^{19}	-	7×10^{16} (Membrane ruptured)
SNAP 8 design parameters 600-kwt core			~1400	10,000	1.4×10^{21}		7.6×10^{17}

irradiated. The hot-cell examination of NAA-77-1 and -4 capsules has been completed. The irradiation of the NAA-77-2 and -3 capsules was initiated.

The irradiation history of the four capsules is summarized in Table 39. Permeation rates and radiation dose data are presented in Figure 61 for the NAA-77-2 and -3 capsules. The gradual increase in the permeation rate which was observed in both NAA-77-2 and -3 has been attributed to the devitrification of the hydrogen barrier material. From comparison of these data with data obtained in long-time aging tests it appears that the devitrification process may be enhanced in an irradiation field. Long-time permeation tests have shown that devitrification of the SCB-1 barrier material does not occur in 5000 hours aging at 1500°F but does occur in 14,000 hours at 1500°F. This behavior may be contrasted with that obtained with the NAA-77-2 capsule which was operated for less than 400 hours at 1500°F; in this case, it was found that changes in the permeation rate corresponded to

those expected when devitrification occurs in the absence of radiation. The gradual increase in permeation rate which has been associated with devitrification of the barrier material is also illustrated in Figure 62 which compares the permeation rates for irradiated and nonirradiated membranes. The results for NAA-77-1 are inconsistent with the above conclusion. The data for the NAA-77-1 capsule were obtained, however, using a mathematical partition function which was designed to separate the extraneous hydrogen flow through the capsule parts from the hydrogen flow through the membrane, and it is possible that this function may also have tended to mask the gradual increase in permeation rate caused by devitrification.

Analysis of the permeation rate data obtained from the NAA-77-2 and -3 capsules yielded the kinetics constants in Table 40. From the data it may be seen that the pressure dependence of the permeation process was not affected by the irradiation, while the activation energy for permeation appears to have changed by approximately a factor of two.

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Figure 61. Permeation Rates and Radiation Dose Data for the NAA-77-2 and -3 Irradiation Capsules

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Figure 62. Comparison of Permeation Rate Data for Irradiated and Unirradiated Membranes as a Function of Annealing Time

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TABLE 40
HYDROGEN PERMEATION KINETICS DATA FOR THE NAA-77-2 AND -3 CAPSULES

	NAA 77-2		NAA-77-3	
	Pre-Irradiation	During Irradiation	Pre-Irradiation	During Irradiation
		3300 hours 4638 hours 1.39×10^{21} nvt ($E_n < 1$ Mev)		3300 hours 4638 hours 1.39×10^{21} nvt ($E_n < 1$ Mev)
Rate at 1 atm and 1500°F (cc/hr/cm ²)	1×10^{-3}	2.8×10^{-3}	0.88×10^{-3}	2.7×10^{-3}
Pressure dependence	1.0	1.0	1.0	1.0
Activation energy 1500 to 1300°F (cal/mole)	24,500	16,000	25,500	11,000

General conclusions that may be drawn from the NAA-77 irradiation capsule experiments are:

1) The ionizing radiation field in a reactor does not significantly alter the hydrogen permeation process in SNAP 8 fuel cladding assemblies.

2) Radiation does appear to accelerate the devitrification process in the permeation barrier material, presumably by enhancement of self-diffusion.

In addition, three more specific conclusions were reached:

1) Inpile annealing of SCB-1-coated membranes has an effect on hydrogen permeation rates similar to that observed in out-of-pile annealing.

2) Inpile annealing of SCB-1-coated membranes caused the activation energy for the permeation process to change in a manner similar to that observed in out-of-pile annealing. The activation energy changed from $25,000 \pm 2,000$ cal/mole to about $14,000 \pm 2,000$ cal/mole.

3) Inpile annealing of SCB-1-coated membranes does not alter the pressure dependence for the permeation. The pressure dependence remains a direct proportionality after either inpile or out-of-pile annealing within the limits of the pressure range studied.

In addition to the observations made on the hydrogen permeation rates in the NAA-77 capsules, some observations were made on the creep behavior of the cladding assembly from the NAA-77-1 capsule. Some indication of radiation acceleration of the creep rate was obtained as illustrated in Figure 63. This observation lead to a study of two permeation membranes which had been tested for 14,000 hours at one atmosphere of differential pressure. These membranes were also found to have experienced an accelerated creep rate. Figure 63 shows the creep rates of the two membranes, E808 and E822, as compared to the creep rate of NAA-77-1 and creep rate data reported for INOR-8.* The tentative conclusion drawn is that for hoop stresses of 400 to 1200 psi, neutron radiation does not appear to affect the creep rate of Hastelloy-N at 1500°F.

*R. W. Swindeman, "The Mechanical Properties of INOR-8," ORNL-2780, January 27, 1961

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Figure 63. Creep Studies of Irradiated and Nonirradiated SNAP 8 Cladding Assemblies

However, long-time low-stress creep rates are accelerated compared to the extrapolated creep rates obtained from membranes tested at higher stresses and for shorter times. In addition, the grain boundaries became decorated with a precipitate, as illustrated in Figure 64, which is probably the β phase (Ni_4Mo) of the nickel-molybdenum system. It is thought that the formation of this precipitate may be the cause for the intergranular fractures observed in all ruptured creep samples. Martin and Weir* have observed both transgranular and intergranular fractures for Hastelloy-N with intergranular fractures occurring at lower stresses where the cracks tend to propagate along the grain boundaries. The observation that SNAP 8 Hastelloy-N cladding material has not exhibited transgranular fracture during creep is being studied further.

b. Engineering Irradiation

(1) NAA-115-1 Fuel Element Engineering Irradiation

The NAA-115-1 irradiation assembly, six experiments containing twelve sublength fuel elements, continued irradiation at Hanford. The data acquisition system continued to monitor and record data from the 38 thermocouples located on these experiments. The NAA-115-1 data reduction code, developed to process this recorded data, has produced time-temperature histograms for each of these thermocouples. Data contained on the histograms indicate that each experiment has operated at design temperatures 62% of the reactor operating time. The remaining percentage of time represents periods of power determinations, calibration, stabilization, and experiment startup sequence.

*W.R. Martin and J.R. Weir, "Effect of Elevated Temperature Irradiation on the Strength and Ductility of the Nickel Base Alloy, Hastelloy-N," ORNL-TW-1105, February 1965

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TYPICAL GRAIN SIZE

TIG WELD ZONE

END CAP

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Figure 64. Hot End of Longitudinal Section of Membrane XE-808
Showing Concentrations of Grain Boundary Precipitates (250X)
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(2) NAA-115-2 Fuel Element Engineering Irradiation

The NAA-115-2 fuel element irradiation experiment closely resembles the NAA-115-1. The major difference in the second experiment is the use of the S8DS-1 reference hydrogen barrier. Other fuel element parameters of the two irradiations are identical and provide statistical confidence to the NAA-115 experiment plan.

Major efforts during the reporting period were devoted to the assembly of fuel specimens and the assembly of the individual experiments.

Eleven of the twelve fuel elements have been assembled and have met the S8DS hydrogen permeation specification. The twelfth fuel element and infuel instrumented unit, is presently being assembled. The assembly of the fuel elements into their respective experiments has been started and experiments 2, 3, 4, 5, and 6 have been completed through NaK bonding of the fuel element to the outer container.

The NAA-115-2 instrumentation system including the gas flow module and data acquisition system was completed during this report period. This system was shipped to the Hanford reactor site and installed adjacent to the operating NAA-115-1 instrumentation system.*

(3) NAA-117-1 Accelerated Fuel Element Engineering Irradiation

This effort is directed toward irradiating, at an accelerated rate, 16 fuel elements which are prototype in diameter but sublength. The test elements are 4 inches long and have effective H/Zr, wt % carbon additive, and burnup levels which span the SNAP 8 reference design. NAA-117 composition and temperatures complement the NAA-115 experiments and, in addition, include power density and burnup level parameters.

A greatly improved computer program used for data reduction has recently been completed. This program reduces the time required for analysis by a factor of approximately 20.

Using this new method, analysis of the data from the 49 thermocouples, all of which are intact, indicates that all temperatures, gas control, and data acquisition systems continued to operate as designed.

(4) NAA-121-1, Prototype Fuel Element Irradiation

Prototype full-scale fuel element performance will be simulated in these experiments which are designed for irradiation at Hanford.* The instrumentation console for NAA-121-1 was installed and checked out at the reactor site at Hanford.

(5) S8ER Post-Irradiation Evaluation

Development of equipment and techniques required in the inspection and examination of the S8ER fuel elements continued. Eddy-current equipment was designed for use in hydrogen distribution determinations. Design studies were initiated to adapt a zone-scanner hydrogen permeation test to hot-cell use.

C. FUEL ELEMENT FABRICATION

The 20 pilot run S8DS fuel elements, assembled at the end of the prior reporting period, were processed through quality control for complete evaluation. The results of the inspection with regard to major product requirements are summarized in Table 41:

The results in Table 41 demonstrated adequate conformance to specification requirements and assembly of elements for the S8DS program was initiated. A group of about 25 normal elements was assembled for the qualification and performance testing program, followed by about 50 enriched elements for the core. As these elements were being processed through inspection it was

*C. E. Johnson, "SNAP 8 Progress Report, November 1964 - January 1965," NAA-SR-10792 (CRD), March 15, 1965

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apparent that the frequency of rejections for excessive hydrogen permeation rates and the magnitude of the permeation rates, were unusually high. Element assembly operations were suspended pending completion of the inspection of all elements in process in order to evaluate the results. A summary of hydrogen permeation rate yields for the pilot run and production elements is shown in Table 42:

TABLE 41
CONFORMANCE OF PILOT RUN ELEMENTS
TO PRODUCT REQUIREMENTS

	Hydrogen Permeation Rate	Envelope Dimensions	Weld Quality	Other Requirements
Number Inspected	20	20	20	20
Number Accepted	16	20	18	20
% Accepted	80%	100%	90%	100%

TABLE 42
HYDROGEN PERMEATION RATE YIELDS

	Pilot Run	S8DS Production Elements		Total
		Normal	Enriched	
Number Tested	20	22	52	94
Number Accepted	16	16	25	57
% Accepted	80%	73%	48%	61%

Examination of the distribution of permeation rates for these elements indicated three general groupings of permeation rates:

- 1) A reasonably normal distribution of elements with permeation rates less than the specification limit of 1.0 cc(STP)/hr,
- 2) A scattering of elements with permeation rates between 1.0 and 2.0 cc(STP)/hr, and

- 3) A group of elements with gross permeation rates, from 4 to 22 cc(STP)/hr, again in a reasonably normal distribution.

Element permeation rates from 1 to 2 cc(STP)/hr may be caused by marginal blends or minor defects in the hydrogen retention barrier; however, a different failure mode is indicated for the elements with gross permeation rates. In the investigation of the cause of element failures it was decided to include the group of about 100 elements fabricated for various developmental tests since July 1964. Hydrogen permeation rate data for all S8DS elements produced since July 1964 is shown graphically in Figure 65 and is summarized in Table 43:

Evaluation of Figure 65 and Table 43 resulted in the following conclusions:

- 1) Inclusion of the 94 developmental test elements confirmed the existence of the three distributional groupings of permeation rates as was previously indicated.
- 2) Permeation rate yields steadily decreased from the early developmental test elements through the most recent enriched production elements.
- 3) The frequency of elements with gross permeation steadily increased over the same period of time.

Element fabrication procedures were reviewed in detail to determine a cause for the apparent change in the quality level of the product. Nothing abnormal was noted and the only process modification was the shift in cup-plug chromizing from a vendor to the inhouse process. This change occurred after completion of the developmental test elements and prior to assembly of the pilot run elements. Attention was directed to the tube chromizing operation because: (1) the important

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Figure 65. Element Permeation Rate Distribution

TABLE 43

ELEMENT PERMEATION RATE YIELDS

Type	Number Tested	Number Accepted	Rejects	
			Total Number	Number < 4 cc(STP)/hr
Development Test	94	77 (82%)	17	4 (23.5%)
Pilot Run	20	16 (80%)	4	1 (25%)
Production Normals	22	16 (73%)	6	4 (67%)
Production Enriched	52	25 (48%)	27	24 (61%)
Totals:	182	134 (71%)	54	33 (61%)

relationship of the chromized layer to ceramic coating quality had been established during the cup-plug coating program* (discussed in the previous report), (2) recent problems of high rejection rates on elements assembled for the

NAA-115-2 irradiation experiment were apparently solved by using tubing from a different chromizing lot, and (3) only minimal control could be exercised over the vendor because the chromizing process was considered proprietary.

The element permeation rate data shown in Figure 65 was replotted by the tube chromizing lot number in Figure 66. The information presented in Figure 66 resulted in the following conclusions:

1) There is evidence of significantly different permeation rate behavior patterns from different tube chromizing lots.

2) Data presented in Table 43 show a possible correlation between decreasing permeation rate yields and element assembly sequence, with the last elements assembled

*C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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Figure 66. Hydrogen Rate Distribution by Tube Chromizing Lot, 188 S8DS Fuel Elements Tested Since July 1964

(the enriched) exhibiting the poorest quality. Analysis of Figure 66 dispels this apparent correlation. Lot Number 1/22/64 contained 21% of all the enriched elements tested and 100% of these elements were acceptable. The probability of this occurring, when the total

enriched element yield is 48%, is remote.

The conclusion was that the tube-chromizing process does have an important effect on subsequent element permeation rate performance.

Work conducted in conjunction with the above has demonstrated a statistical correlation exists between the adherence of the ceramic coating to the tube and fuel element permeation rates. Variations in coating adherence have been found which correlate with the between-lot and within-lot fuel element permeation rate distribution (Figure 66).

Based on the investigations conducted, it was evident that the vendor's proprietary tube-chromizing process was not producing the desired chrome layer on a controlled, repetitive basis. Therefore, it was decided to accelerate the program of adapting the technology developed for in-house chromizing of cup-plugs to tube chromizing.

The scope of the inhouse tube chromizing program includes the following:

- 1) Development of a process to prepare tube assemblies for ceramic coating which will result in improved adherence of the coating to the tube assembly, and
- 2) Demonstration through accelerated performance tests, that elements produced by the new process meet S8DS product requirements.

Two development programs are being carried forward in parallel: (1) rechromizing of existing chromized tubing, and (2) the chromizing of new tubing. The status of these development efforts is discussed in detail in another section of this report. The related fabrication program has been planned to translate the results of the experimental data into the fabrication and testing of a number of full-scale elements. At the end of the reporting period, a group of 25 chromized tubes were re-chromized inhouse, were ceramic-coated, and were being processed into fuel elements. The initial group of new tube assemblies was being prepared for the chromizing operation.

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V. COMPONENT DEVELOPMENT

A. REFLECTOR AND DRIVE SCRAM MECHANISM

Safe operation of the SNAP 8 reactor system during ground testing requires a means for shutting the reactor down quickly by withdrawal of the control drums. A control-drum-scam drive mechanism has been designed which is capable of operating under the S8DS environmental requirements. This design is based on a developmental prototype.*

The S8DS system design employs a gear-reduced, 1.8°-step actuator and a scram mechanism for each of the six control drums. Each mechanism, mounted on top of the reflector, engages the upper drum shaft through a flexible coupling. Each control-drum-drive system consists of the actuator and brake unit, a scram clutch and drive spring mechanism, and a

snubber assembly to absorb the energy of the scrambled drum. Details of design and operation of the system have been reported previously.†

Performance mapping and design verification tests will be conducted on three scram-drive units, one with clockwise drive and two with counterclockwise drive, under simulated operating conditions of vacuum and temperature.

The clockwise drive unit has been subjected to a complete bench test (ambient air at room temperature) to determine if mechanical and electrical performance are within the required design parameters. Actual control-room scram times were measured. The frictional torque was 7 in.-lb. The comparison of the test data to theoretical scram times (under 0- and 6-in.-lb frictional torque) is shown in Figure 67. A constant 6-in.-lb total drive-system friction torque

Figure 67. Scram Times for S8DS Scram Mechanism

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, May through July," NAA-SR-9992 (SRD), September 22, 1964

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

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Figure 68. Clockwise Drive Unit Installed In Vacuum Chamber

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Figure 69. Performance of Two S8DS Control-Drum Actuators After Four Thermal Cycles

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value was used in design analysis. This value includes allowance for friction of the control-drum bearing, scram-mechanism bearings, and drag of the adjacent coils of the clock-type scram spring. All bench-test data were obtained by using an S8DRM-1 design actuator to wind the scram spring and drive the simulated control drum.

The clockwise drive unit has been installed in a vacuum chamber (see Figure 68) and has operated successfully through three scram cycles at 1×10^{-5} torr pressure. The S8DRM-1 actuator has been replaced by an S8DS model utilizing S8DS-type high-temperature cabling and connectors. Thermal cycling of this unit has been initiated to evaluate mechanical and electrical operation as a function of temperature.

B. CONTROL-DRUM ACTUATORS

Control-drum actuators are required to rotate and position the reactor control drums in response to signals from the controller. The reactor coolant temperature is controlled by this action. An actuator which provides a 1.8° output step is the reference design and is now undergoing design verification testing. The actuator output obtains the required torque and 0.25° step at the control drum through a gear reduction.

The indirect-drive actuator designed for the S8DS reactor is the reference design for the SNAP 8 flight system. The fabrication of test actuators of this design is now being completed.

Results of bench tests on early actuators exceeded the design requirements by adequate margin and agreed with design calculations.

The first actuator has completed over 1000 hours of endurance operation at 1025°F and 10^{-7} torr. The second unit has completed over 800 hours under the same conditions. Figure 69 shows the performance data of these actuators.

The analysis of the data shows the following:

a) The torque output and bearing friction of the two units match closely.

b) The slight variation (10%) between the two units in current required to release the brake is due to variation in the brake spring force.

c) The 1.2 amp current with 21-volt input required to produce 40 in. -oz dynamic or stepping torque is well below the normal 28-volt operating current of 1.6 amp. Bench test results indicate that the motor will produce over 90 in. -oz of dynamic torque with the normal current of 1.6 amp. The design requirement is 37 in. -oz. The measurable dynamic torque at temperature and vacuum is limited to 40 in. -oz by the test fixture.

d) The current required to release the brake is gradually decreasing. This may be attributed to the force relaxation in the René 41 spring which holds the brake in the closed position.

e) The motor bearing friction decreases from 5 to 2.8 in. -oz after 20 to 80 hours under vacuum at 1025°F . This change of friction at high temperature on the graphite-aluminum oxide motor bearing material had been demonstrated previously in sliding block tests.

C. ELECTRICAL COMPONENTS

1. Cable Harness

The objective of the cable harness development program is to test, evaluate, and recommend cable material and associated hardware for use in an integrated harness which is capable of transmitting power and electronic signals between various components of the system.

The cable harness concept for S8DS has been revised to eliminate unreliable plug-in

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Figure 70. Connector Backshell Assembly Details

connections at components. This is accomplished by supplying cable material, as recommended for SNAP 8,^{*} in short lengths as an integral part of S8DS components. The cable is connected to the component by percussive-arc welding techniques.[†] Protection of the terminations is provided by a split-back shell developed and designed for this specific application. A pictorial drawing of the connector back shell is shown in Figure 70. The end of the cable, away from the component, has ring lugs on the conductors for easy assembly to a terminal board. Terminal boards are located on the reactor system cable harness support structure and provide interconnecting means between components and the permanently mounted cable harness. After reflector assem-

bly and initial test, the connections at the terminal board are to be welded by inert gas shielded arc process. This cable arrangement results in a continuous welded circuit from the component to the remote handling connector.

2. Coil Techniques

Metal bobbins significantly increase the mechanical strength of the coil assembly.[§] Under coil assembly test conditions,[§] however, dielectric breakdown occurred in both of two tests performed on a commercial high-temperature enamel coating. Evaluation of two different coating materials is now in progress. The first material is flame-sprayed aluminum oxide. The other is Allen PBX cement which will be sprayed on the metal bobbin and then cured prior to coil formation. The test environment is 1000°F and 10^{-8} torr.

Selection of SAP II insulation for SNAP 8 application was based on previous test data,^{**} which indicated the insulation resistance of SAP II to be superior to that of PBX. Recent tests^{*} have shown that insulation resistance is reduced considerably by oil backstreaming from diffusion-pumped vacuum systems. The electrical properties of PBX cement have been re-examined in an oil-free ionization pumped vacuum system at 1100°F at 10^{-8} torr for 1000 hours. The PBX insulation resistance of 0.2 megohms remained constant throughout the test period. This value is similar to the SAP II insulation resistance previously reported for identical test conditions.

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, May through July 1964," NAA-SR-9992 (SRD), September 22, 1964

†C. E. Johnson, "Progress Report, SNAP 8, August-October 1964," NAA-SR-10492 (SRD), December 23, 1964

§C. E. Johnson, "SNAP 8 Progress Report, November 1964 - January 1965," NAA-SR-10792 (CRD), March 15, 1965

**C. E. Johnson, "SNAP 8 Quarterly Progress Report, April-June 1963," NAA-SR-8692 (SRD), September 27, 1963

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Coils prepared by wet-winding techniques^{*,†} were assembled in S8DS developmental actuator stators. Some coils experienced mechanical damage and some coils failed due to loss of electrical continuity during thermal-cycling tests. Mechanical damage resulted from inadequate lead retention strength while the loss of electrical continuity was caused by wire breakage near the solid-to-stranded, percussive-arcwelded, finish-lead joints. The following design and process modifications were made to minimize coil failures:

a) The process of chromeplating termination areas has been discontinued. This minimizes the coldworking caused by manual handling of the conductor. The improved coil drying and curing process^{*} has effectively deterred conductor corrosion so that termination chromeplating is no longer necessary;

b) Inclusion of a 3/4 turn of stranded finish lead wire located along the side of the coil for increased mechanical strength;

c) Inclusion of an expansion loop in the coil conductor near the solid-to-stranded, percussive-arcwelded, finish-lead joint to remove residual tensile stresses from the conductor; and

d) Revision of the manufacturing process to include thermal-cycling acceptance tests on completed coil assemblies.

The above changes reduced the infant mortality of mechanical damage from 10 to 4% and of electrical failures from 10 to 2%.

D. MECHANICAL COMPONENTS

1: Springs

The spring development program objective is to provide design verification of reactor spring design, establish spring characteristics at elevated temperatures and reactor environments, and generate spring material data for design optimization. Tests are being conducted to evaluate load loss as a function of operating temperature and time to permit estimates of safe operating stresses and relaxation behavior at the SNAP 8 environmental temperature and irradiation level.

The long-time load-relaxation tests of Rene' 41 springs in air atmosphere were terminated. Table 44 summarizes the results of these tests conducted on two different scram ejection spring configurations. Metallurgical analyses of these springs is in progress.

TABLE 44
LOAD RELAXATION TEST DATA,
RENE' 41 SPRINGS IN AIR

	Springs [*]		
	1	2	3
Test temperature (°F)	1,100	1,100	1,050
Initial uncorrected stress at temperature (psi)	36,400	44,450	42,550
Spring energy relaxation at 10,000 hr (%)	35.9	34.8	16.9
Load relaxation at 10,000 hr (%)	20.0	19.2	8.9
Load relaxation at 11,500 hr (%)	21.2	19.7	9.3
Load relaxation at 15,000 hr (%)	23.7	—	—

*Spring 1 is 0.222-inch-diameter wire with 14 active coils, 1.47 inches in diameter. The total length is 5.0 inches and the spring rate is 142 lb/in. Springs 2 and 3 are 0.280-inch-diameter wire with 8 active coils, 1.74 inches in diameter. The total lengths are 4.1 inches and the spring rates are 375 lb/in.

*C. E. Johnson, "Progress Report, SNAP 8, August-October 1964," NAA-SR-10492 (SRD), December 23, 1964

†C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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TABLE 45

COMPARISON OF STAUFFER-880 SPRING
LOAD RELAXATION INPILE
AND OUT-OF-PILE

Spring	Test Temperature (°F)	Irradiation	Load Relaxation at 1000 Hours (%)
27	1035/675	No	14.6
22	1035/675	2×10^{10} R 1.5×10^{19} nvt	8.2
26	1050	No	9.6
24	1055	2×10^{10} R 1.5×10^{19} nvt	14.8
28	1140	No	51.6
21	1140	7×10^9 R 1.5×10^{18} nvt	25.2

TABLE 46

BEARING MATERIAL COMBINATIONS

	Set No. 1	Set No. 2	Set No. 3
Sockets	Al ₂ O ₃ -coated Inconel-X	Al ₂ O ₃ -coated Inconel-X	Al ₂ O ₃ -coated Inconel-X
Shafts	Al ₂ O ₃ -coated Inconel-X	Al ₂ O ₃ -coated Inconel-X	Al ₂ O ₃ -coated Inconel-X
Thrust Washers	Al ₂ O ₃ -coated Inconel-X	Al ₂ O ₃ -coated Inconel-X	Al ₂ O ₃ -coated Inconel-X
Balls	Al ₂ O ₃ -coated Inconel-X with dry film*	Al ₂ O ₃ -coated Inconel-X	Solid carbon monoballs
OD			
ID	Carbon bushings	Carbon bushings	

*Sodium-silicate-bonded molybdenum disulfide plus graphite dry-film lubricant

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Figure 71. Control-Drum Bearing Running Friction vs Time

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Tests identical to those summarized in Table 44 have been started to provide verification of load-relaxation behavior, especially in the early stages for which relaxation data were not recorded in the previous tests. Metallurgical analyses of these springs will be conducted after 1000 hours of test to determine what metallurgical behavior accompanies the inflection in load relaxation/time behavior at that stage in the tests.

Load-relaxation tests of three 88% Ta - 12% W alloy springs (Stauffer 880) in vacuum environment have completed 1300 hours of test at temperature. These tests are being conducted at the same loads and temperatures as three similar springs on inpile irradiation test. Table 45 compares the inpile and out-of-pile results for the springs after 1000 hours of test. At this stage of testing the results are not consistent. Test data indicate that Stauffer 880 exhibits a greater relaxation rate than René 41 at the same stress levels and temperatures. The loading of the Stauffer 880 and René 41 springs of identical configuration was selected to produce equal stresses at each test temperature. These loadings stress the Stauffer 880 spring to 2/3 yield strength and the René 41 spring to 1/3 yield strength.

2. Bearings

a. Ground Environment Test

The S8DS ground test reactor is planned to operate in the slightly oxidizing atmosphere of 10^{-5} torr pressure for over 10,000 hours. Three different tests of control-drum bearings using different material combinations are being run at 1150°F and 10^{-5} torr air. A tabulation of bearing materials for each of the test bearings is shown in Table 46. Each of these bearing sets have run in excess of 1000 hours and the nonlubricated bearing set (No. 2) has run 4500 hours. The running frictional torque of each of

these tests has remained less than 1.9 in.-lb. The design objective torque is 2.5 in.-lb or less. Plots of the running friction versus test time of all tests are shown on Figure 71.

The bearing test mechanism is shown in Figure 72. The test bearings are located within the cylindrical mass at the right of the picture. A thrust load is applied to the upper bellows and pin as shown in the figure. The rotational drive is applied to the bellows coupling on the right which is driven by a magnetic feed-through drive. The frictional reaction torque is resisted by the fork extending from the lower bellows and measured by a force transducer outside of the vacuum system.

b. Shock and Vibration Tests

A shock and vibration test was run to determine the design integrity of a set of S8DRM-1 control-drum bearings using carbon-graphite solid monoballs and dry-film lubricated sockets. The SNAP 8 reflector and dynamically related masses were mocked up to simulate prototype design loadings. The test fixture and instrumentation are shown in Figure 73.

The shock and vibration program shown in Table 47 was imposed to the fixture base. Vibration accelerations as high as 120 g were recorded at the bearings.

The bearings survived the shock and vibration testing successfully with no cracking or spalling of the carbon-graphite monoballs. Figure 74 shows the post-test condition of these bearings.

3. Gear Mockup Tests

The reference design for the SNAP 8 control-drum-drive system specifies a metallic pinion gear train to transmit actuator torque to the reflector drums.

In a space environment, the phenomenon of surface adhesion might occur where these "clean" surfaces are in contact under load and

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Figure 72. Bearing Performance Test Mechanism

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Figure 73. Bearing Vibration Test Apparatus

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TABLE 47
BEARING SHOCK AND VIBRATION TEST CRITERIA

	Duration	Number	Level
Shock	Shape and duration of the input pulse must be one of the following: Triangular - 10 millisec Rectangular - 5 millisec One-half sine - 8 millisec	3 in each direction of X, Y, and Z axes (18 shocks)	35 g
Vibration Logarithmic Sweep	10 minutes	Once in each of X, Y, and Z axes	16 to 100 cps at 6.0 g peak 100 to 180 cps at 0.0118-in. double amplitude 180 to 2000 cps at 19.0 g peak
Sinusoidal Resonance	5 minutes at each resonance from sweep	—	One-half of sweep amplitude
Random Vibration	5 minutes	X	20 to 200 cps at 2 db/oct 200 to 700 cps at -17.5 db/oct 900 to 2000 cps at 0.15 g ² /cps

at elevated temperature. The nature of the adhesion can be by diffusion, chemical reaction of intermetallic formation, and/or surface plastic flow resulting in asperity interlock. It is necessary, therefore, to prevent the occurrence of these conditions, or if they should happen, the type of interface bond formed should be so weak that shearing can readily occur with a minimum of surface damage.

The approach to the solution of this problem has been to use known dry-film lubricants such as graphite, MoS₂, and MoSe₂, which are sodium-silicate bonded so that a low-vapor-pressure, low-shear-strength, surface-coating system is achieved.

A test program to evaluate the performance of specific dry-film lubricants at gear operating environments is in progress. In these tests, 300 series stainless-steel specimens, coated with the lubricant, slide against one another under 1150°F and pressure of 10⁻⁵ to 10⁻⁸ torr test conditions. Normal load on the specimens is 40 psi and the test procedure is to initiate sliding 10 minutes during each hour for approximately 150 hours (about 1200 inches of travel). Originally, specimens were kept separated during initial pump-down and heat-up to operating temperature; however, a change in procedure was made in that specimens are now kept in contact under 40-psi normal load during the pump-down and heating cycle. The test data obtained during the reporting period are summarized in Table 48.

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Figure 74. S8DRM-1 Control-Drum Bearings After Shock and Vibration Test

TABLE 48
GEAR MOCKUP TEST DATA, 1150°F

Test Number	Dry-Film Lubricant	Travel (in.)	Pressure (torr)	Time at Temperature (hr)	Initial Friction	Friction (Running)	Test Status
3-US	MoSe ₂	12,365	10 ⁻⁷	5626	—	0.23	Continuing
8-US	MoS ₂	11,670	10 ⁻⁷	160	<1	0.18	Completed
12-GS	MoS ₂	13,340	10 ⁻⁵	262	<1	0.2	Completed
13-GS	MoS ₂	6,720	10 ⁻⁵	142	<1	0.27	Continuing

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One of the original tests, on a MoSe₂-coated system (Test 3-US), has completed the programmed cycle and is continuing on as an endurance test. The endurance test involves the maintaining of the test specimens at test conditions, but sliding is actuated only once each 24-to 72-hour period. Failure is considered to occur when the coefficient of friction exceeds 1.0. This test has completed more than 5600 hours in this phase and is still continuing. A single similar test performed on a MoS₂-coated system* was terminated after 1758 hours of test because the coefficient of friction exceeded 1.0.

Three other tests of MoS₂-coated systems (Tests 8-US, 12-GS, and 13-GS) were all started with the specimens in contact during pump-down and heat-up. In these tests, the starting coefficient of friction was greater than 1.0, but dropped to the typically observed values of 0.2 to 0.4 after short periods of sliding. After Test 12-GS had completed its programmed cycles, the temperature and pressure were brought to room conditions. The specimens were then pumped down to 10⁻⁵ torr and reheated to 1150°F with the normal 40-psi load. Initial actuation showed the coefficient of friction to be 0.2, the same as it had been at shutdown. The same outlined procedure was performed on Test 8-US, except pressure levels were 10⁻⁷ torr and the same result was observed. From this, it appears that initial starting friction can be excessively high and that either a high-temperature vacuum bakeout and/or burnishing operation is necessary to eliminate this high starting friction condition.

Three 10,000-hour tests of reference design gear materials (Inconel-X vs Hiperco 27), both coated with MoS₂ plus graphite dry-film lubricant, were initiated for confirmation of design application. The test fixtures and setup are the

same as those used for the gear mockup tests. The environment is 1000°F at 10⁻⁷ to 10⁻⁸ torr pressure. Sliding is initiated after alternating dwell times of 80 and 88 hours. The results, to date, are given in Table 49. Testing on Sample 8R was discontinued after 3796 hours because of a vacuum failure. The remaining two tests are continuing and the data to date tends to confirm the friction results of the previously run short-time tests.

TABLE 49
REFERENCE GEAR LUBRICANT 1000°F DATA,
MoS₂ PLUS GRAPHITE DRY-FILM COATING
ON INCONEL-X AND HIPERCO-27

Test Number	Time at Temperature (hr)	Coefficient of Friction	Status
8R	3796	0.2	Discontinued
9R	3240	0.25	Continuing
10R	2876	0.35	Continuing

E. SENSORS AND INSTRUMENTATION

1. Instrumentation

a. Drum Position-Sensing System

The drum position sensing system measures the reactor control-drum angular position and provides an output signal for diagnostic purposes. The desired system accuracy is ±1% of full scale. Since the sensor is mounted on the reflector assembly, it must be capable of operating in the reactor environment throughout the 10,000-hour life of the nuclear system.

(1) Transmitter

A rotary variable-reluctance (differential transformer) type of sensor is being developed for this application. The stator contains one primary and two secondary windings. The magnetic sensor rotor is shaped so that the coupling

*C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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to one secondary winding, hence its output voltage, is at a maximum when the output voltage from the other secondary winding is at a minimum. The demodulator is designed in such a manner that the output voltages are added in series-bucking, changing polarity at the center of the range of rotation.

Two prototypes of the variable-reluctance design have been fabricated and assembled. Figure 75 is a photograph of an assembled sensor. Thermal-vacuum testing has been initiated, from which preliminary data indicate that the sensors have exceeded all performance requirements.

Coil insulation resistances for both sensors have been well above the required one megohm minimum. Figure 76 shows the coil insulation resistance as a function of time at 800°F (lowest temperature measurements on each sensor). Coil winding resistances have shown no meas-

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Figure 75. Drum Position Sensor

urable change (at constant temperatures) throughout the initial 6 thermal cycles and 180 hours at 800°F.

Zero shift as a function of temperature on the first thermal cycle was less than one degree on both sensors. Data for S/N 2702 is shown in Figure 77. The higher null voltage, compared to the previous S8DRM design, is attributed to increased eddy-current losses resulting from the more massive structure of the sensor. This

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Figure 76. Drum Position Sensor Insulation Resistance (Coil to Case) vs Time

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Figure 77. Drum Position Sensor Output vs Shaft Angle

has no deleterious effect on the system demodulator output, however, as shown in Figure 78.

Shaft torque at room temperature decreased slightly as a result of the first 6 thermal cycles and 180 hours at 800°F. Table 50 shows the results of torque measurements. A maximum torque of 16 in.-oz is allowable for the sensor. Improved fabrication methods over the previous sensor design are credited with the low torque

values. Bearing diametral clearance in these sensors is ~0.0006 inch at room temperature.

(2) Demodulator

The demodulator provides regulated a-c excitation to the sensor primary winding. The a-c output from the secondary winding varies as a function of sensor shaft angle, in a linear relationship over the angular range of 154 degrees, with a phase reversal at the null (midpoint) position. The demodulator converts this a-c signal to a 0 to 5 vdc ramp over the angular range.

Figure 78. S8DS Position Sensor-Demodulator System Output vs Shaft Angle

TABLE 50
POSITION SENSOR PROTOTYPE SHAFT
TORQUE CHARACTERISTICS*

Sensor Number	Direction	Torque Initial Values Before Thermal-Vacuum Test (in.-oz)	Torque Values After 6 Thermal Cycles and 180 Hours at 800°F (in.-oz)
2701	Clockwise	0.75	0.45
	Counter-clockwise	0.70	0.45
2702	Clockwise	0.70	0.55
	Counter-clockwise	0.65	0.50

*Room temperature in air

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A prototype demodulator was assembled (see Figure 79). The final S8DS units will be electrically identical to this prototype. Changes were made in the regulator reference voltage range and a filter capacitor value to optimize system stability. These modifications were incorporated in the system drawings.

After these modifications were made, system testing was conducted using the prototype demodulator and the prototype position sensor (Sensor No. 2702), showing extremely good stability with sensor temperature variation. Between 800 and 530°F, less than 2 degrees of arc equivalent change occurred in demodulator output signal (see Figure 80). The response of the regulator circuit to changes in sensor temperature is illustrated in Figure 80. The excitation voltage varies approximately three volts over the temperature range to maintain a constant secondary sum voltage of about 13 volts. Further testing is proceeding to define all important system characteristics.

Various demodulator components and sub-circuits were irradiated to SNAP 8 low flux levels (10^{13} nvt neutrons, 10^7 R gamma). Preliminary data on diode pairs of the type to be

used in the detector-converter indicated a differential voltage stability (change in differential forward voltage drop) of 0.001 volt, which is within performance requirements. A complete converter circuit from the demodulator, also irradiated under operating conditions, showed a drift in output of 0.003 volts. This represents approximately 0.12% of full scale signal. The output magnetic amplifier indicated excellent stability of all characteristics, with measured closed loop gain variation within 0.1%.

b. Limit Switch

(1) Shorting-Bar Switch

Thermal-vacuum tests were continued on three developmental units, which have now accumulated 3700 hours at 1000°F (10^{-6} torr), and 50 thermal cycles during a period of 700 additional hours, for a total of 4400 hours environmental testing. During these tests there has been no measurable change in switch point (within 0.002 in.) or spring force vs distance characteristics (within 5%). The maximum calculated stress in the springs is approximately 45,000 psi in the compressed condition.

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Figure 79. Prototype Demodulator

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Figure 80. S8DS Demodulator Regulator Input and Output vs Sensor Temperature

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An additional switch, fitted with a PO3N graphite contact, was added to the test. This is the material specified for the S8DS design switch and possesses higher physical strength than the previously used PO3 HT.

Parts for six units of the S8DS design are being fabricated. The S8DS design is similar to the development model but incorporates the following improvements:

- 1) Use of a screw adjustment of the clearance of the graphite contact within the movable contact assembly, replacing the shims used previously. This minimizes the possibility of binding due to thermal effects.

- 2) A larger diameter graphite roller which provides plunger antirotation guidance, operating in slots in the housing. This replaces the metal tabs previously used and eliminates any possibility of metal-to-metal rubbing in the switch.

- 3) Use of a larger diameter alumina-coated pin on which the roller rotates. This provides an alumina-graphite combination at the point of highest expected bearing loads.

(2) Snap-Action Switch

Tests on three modified switches (graphite plunger and René 41 overtravel spring) were continued until the switches had accumulated 3700 hours at 1000°F and 50 thermal cycles in vacuum in the unactuated position. Calibration at the end of this time showed a setpoint shift at approximately 0.010 in. with a corresponding degradation in actuation force. Development effort on this type of limit switch has been discontinued.

2. Ground Instrumentation (S8DS Precision Position-Sensing System)

The S8DS system requires six precision drum-position sensing channels for use in reac-

tor experiments to be conducted throughout the reactor test. System accuracy of $\pm 0.1^\circ$ is desired.

a. Reference Design (Step-Counting and Verifying System)

The step counting and verifying system utilizes a position sensor, demodulator, and digital voltmeter to verify drum motion when the actuator is pulsed by the controller and to provide a digital display of position by counting these steps. The system design was described previously.*

The logic design of the system was completed. Compatibility tests on breadboard versions of input and output interface circuits, (pulse selector logic, digital error amplified, and digital counter), were completed. System tests were conducted in conjunction with an S8DS scram kit, in which it was demonstrated that the sensor-demodulator possessed sufficient resolution when working with the digital error amplifier. Input pulse noise problems were successfully solved with the use of isolation relays and time delay logic circuitry.

Assembly of a prototype counting system, utilizing a printed circuit board on which most interconnecting wiring is etched, is approximately 75% complete.

b. Backup System (Variable Capacitance Transducer with Demodulator)

The backup system utilizes a capacitive transducer having no magnetic coils.* System tests utilizing temperature compensating techniques on the demodulator were conducted, in which the transducer was temperature-cycled from 70 to 615°F. Figure 81 is a graph of system performance using Sensor No. 002, in which error from ideal output is plotted as a function of sensor shaft angle. The data illustrate the

*C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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Figure 81. Effect of Temperature Compensation on Capacitive Sensor System

improvement in performance obtained with the temperature compensating adjustment. Average deviation from initial room temperature data was $\pm 0.04\%$ of full scale, with a maximum deviation of 0.009% (equivalent to 0.13 degrees arc) at a shaft angle of 140°.

F. CONTROLS

1. Controller

The reactor controller is required to provide properly sequenced outputs to three control-drum actuators which in turn, control the reactivity of the reactor. The direction, rate, and order of control drum rotation are controlled by the state of the input signals to the controller. The design goal for the controller is a minimum of 10,000 hours of operation over which it will receive a significant total integrated radiation dosage. The primary purpose

of this program is the development of a controller of reliable performance under these conditions.

The design approach selected utilizes magnetic core-transistor circuitry. Five basic modules are required; a logic module, shift driver, oscillator, relay driver, and relay. The magnetic approach permits a greater than 50% reduction in semiconductor devices as compared to an all-transistor approach. This is a significant factor in increasing the overall reliability of the controller.

a. Prototype Controller

Individual subassemblies utilizing packaged circuit modules were checked out for functional operation. These include the time base generator, the actuator sequencer, the phase sequencer, and the relays. All subassemblies are being interconnected for final functional testing prior to assembly into a packaged unit. Design of the packaged unit is 50% complete.

The reliability test assembly, which incorporates the basic controller modules in a simple logic circuit, has operated without a failure for 4440 hours. The purpose of the test is to obtain module reliability data under actual operating conditions. The test has accumulated 24×10^6 cycles. There are seven of the basic core-transistor-logic modules in this assembly so that approximately 31,000 module-hours have been accrued without failure.

A power switching relay endurance test was completed. These relays switch power to the actuator phases and brake. The purpose of this test was to determine the relay operating life at simulated actuator load conditions. Five radiation-hardened latching relays continuously switched a 6 ohm, 20 mh load on each set of contacts. This load represents the maximum load due to a cold actuator. More than 100,000 cycles were completed without malfunction. This

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number of cycles meets the design requirements for 10,000 hours of operation. Detailed test results are shown in Table 51.

TABLE 51
POWER SWITCHING RELAY
ENDURANCE DATA

Relay	Cycles at Failure
1	230,000
2	160,000
3	265,000
4	290,000
5	110,000

An irradiation of the four basic controller circuit modules was performed. Three logic modules (CTL), two shift drivers (CTI), two relay drivers (RD), and one oscillator (OSC) were irradiated. A total dose of 6.4×10^{13} nvt and 1.9×10^7 rads was accumulated. Measurements taken before and after the irradiation showed performance to be within satisfactory limits. The irradiated logic modules and relay drivers were substituted into a phase-sequencer subassembly and normal operation resulted. The oscillator and shift drivers were substituted into the time-base-generator subassembly and normal operation was also attained there.

b. S8DS Controller

The S8DS controller is a rack-mounted unit designed primarily for use in the S8DS system test. The unit provides four discrete stepping rates, three of which are variable over a wide range. The unit provides visual status information and a variable setpoint capability which is used to vary the point at which stepping rate changes are made. Manual rate selection and manual overrides are included where required. This controller will be used to study and determine the optimum reactor startup cycle. The

results from this study will provide the final performance requirements for the reference design.

Fabrication of this controller has been initiated. Construction of the various circuit cards is complete. Wiring of the drawer assemblies is underway.

2. Temperature Detector

The temperature detector provides voltage signals to the temperature switch which are proportional to the reactor outlet coolant temperature. The detector, in conjunction with the switch, establishes the limits of the reactor deadband and commands the controller to drive the drums whenever the outlet coolant temperature passes outside this deadband.

a. Reference Design

The reference design is a platinum resistance temperature detector (RTD) which forms one leg of a three-wire Sieman's bridge. As the resistance of the detector changes with temperature, the output of the bridge to the control winding of a high-gain magnetic amplifier in the switch varies accordingly. The bridge is adjusted to null at that detector resistance corresponding to the temperature at the middle of the deadband.

Thermal testing at 1300°F of the evaluation RTD's continued. These detectors are maintained at 1300°F to simulate long-term operating conditions. They are checked for drift every 500 hours by calibration in a triple point cell (0.01°C) utilizing precision laboratory resistance measuring techniques. Accuracy of this calibration converted to equivalent temperature at 1300°F is $\pm 0.2^\circ\text{F}$. Six detectors from an original 25 remain within acceptable limits (-4°F shift in 10,000 hours), and their drift in terms of equivalent setpoint shift is shown in Figure 82. All data is referenced to zero after an initial

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Figure 82. Resistance Temperature Detector Drift Data

1000-hour stabilization period. This stabilization will be performed by the manufacturer on future units prior to acceptance. Prototype detectors being procured are similar in design to the RTD's evaluated in these tests. Detector No. 3 was damaged in handling at 6500 hours and was removed from test.

The temperature detector NaK test loop has been completed and preparations are being made for filling and flow testing with NaK. This loop will be used for thermal cycling and vacuum endurance tests of detectors.

b. Backup Design

The backup design is a thermocouple sensor which also operates into a high-gain, magnetic-amplifier switch. In addition, this design requires a reference potential source and an associated highly regulated, low voltage supply.

Stability testing of three chromel-alumel thermocouples continues with drift within 5°F for 7000 hours and testing of various noble metal junctions is being initiated. Due to the complexity of the reference source and power supply required, however, the entire thermocouple approach is less attractive than the RTD approach at this time.

*C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

3. Temperature Switch

The temperature switch receives the output from the RTD's and provides signals to the controller whenever the reactor outlet coolant temperature passes outside the deadband limits as defined by two preset points.

Testing of the S8DS design switch continues. Repeatability of the switch point over the operating temperature range of 50 to 120°F is $\pm 1.5^\circ\text{F}$. Deadband width is from 16 to 75°F. The null point or center of the deadband may be adjusted $\pm 250^\circ\text{F}$. The problem of setpoint stability for all possible settings* has been solved by adjusting bias winding turns for coarse setting and bias winding current for fine setting. In the S8DS switch design, adjustment of winding turns is accomplished by taps on the winding and adjustment of current is accomplished with a potentiometer in series with the winding.

G. MATERIALS DEVELOPMENT

The objective of this study is to evaluate coatings for the SNAP 8 beryllium neutron reflectors which will provide high infrared radiation emittance over long periods in high vacuum. The performance of the coatings is being tested by means of long-term environmental tests at 1150°F and vacuum in the 10^{-7} torr range. Emittance of the samples is measured periodically during the course of the tests. The exposure time for the tests is 10,000 hours.

A total of 36 beryllium samples is included in the environmental testing, the results of which are shown in Table 52.

The total normal emittance of anodized coatings ranged from 0.55 to 0.67 over the last 1990 hours of testing. The "bare" beryllium samples retained their initial tarnish and samples coated with oxidized silicon-beryllium alloy retained their initial emittance of 0.67 to 0.82

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TABLE 52
EMITTANCE COATINGS FOR BERYLLIUM,
LONG-TIME TEST DATA*

Number of Samples	Type of Coating	Initial Emittance Range	Total Test Time (hr)	Latest Emittance
10	Anodized beryllium	0.76 to 0.87	3330	0.55 to 0.67
5	Anodized beryllium	—	1000	0.52 to 0.58
3	Uncoated beryllium (natural tarnish)	0.15 to 0.23	1000	0.13
3	Oxidized fused silicon beryllium	0.63 to 0.84	1000	0.67 to 0.82
15	Plasma-sprayed chromium oxide	0.88 to 0.90	1000	0.88 to 0.90

*Total normal emittance measured at 800°F

after 1000 hours of testing. Plasma-sprayed chromium oxide coatings exhibited the highest emittance obtained thus far in this test (0.88 to 0.90). This is an encouraging performance since previous tests of plasma-sprayed chromium oxide coatings on beryllium have spalled during testing at 1150°F.* The environmental test is being continued.

H. SAFETY COMPONENTS

1. Reactor Destruct Technology

An explosive destruct system is needed to provide the means for reactor core vessel rupture and dispersal of the fuel elements at any time during launch, reactor startup, or after useful orbital operation of the reactor. The development of the destruct system involves the selection and performance verification of explosive materials, the design and testing of explosive transport mechanism, the evaluation of reactor dispersal efficiency, and the development of initiation mechanisms and controls.

Based on literature surveys and liaison with government agencies experienced in the explo-

sives industry, it was concluded that the selection of explosives for the SNAP 8 destruct system should be restricted to the class of high explosives used by the military. In general, the military explosives are noted for their explosive power, reliability, and safety during handling. These explosives are organic compounds and highly reliable firing trains have been developed for most of them. Inorganic explosives, on the other hand, do not normally possess the explosive power of the organics and experimental data are limited. For these reasons the investigation of metal-oxidant inorganic explosives has been terminated for the present.

Initial investigations have shown that no practical explosive is known which is stable enough to withstand storage in the temperature and irradiation environment adjacent to the SNAP 8 reactor vessel. Consequently, the instrumentation compartment was tentatively selected as the storage location for the explosive. Any alternate location with comparable temperature and radiation environment can also be considered. The 10,000-hour irradiation exposure in the instrumentation compartment is 1×10^7 R (gamma), 1×10^{13} nvt (fast neutron) and the average temperature is 160°F. There are a number of organic explosives which have demonstrated a functional threshold above this total dose at room temperature. Some of these were also irradiated at 160°F for short time periods. Performance after long term (10,000 hours) storage in the instrumentation compartment environment, however, requires verification. A screening test program is being set up to determine which explosives perform better under these conditions. The materials to be tested are those which make up modern high-explosive firing trains, including main charge explosives, booster materials, and detonator materials. In

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, April-June 1963," NAA-SR-8692 (SRD), September 27, 1963

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addition, a number of propellants to be used to transport the destruct charge from its storage location to the reactor vessel are included. Based on existing thermal and radiation stability data the following materials were selected for participation in the screening test program:

- 1) Main charge: DATB, TATB, Tacot
- 2) Booster materials: HNS, tetryl
- 3) Detonator materials: KDNBF, 95/5 lead styphnate + lead azide
- 4) Solid propellants: ANP 3095, AK-14, AN584J

The test plan is being formulated and requirements for test equipment are being evaluated. Conceptual design of a destruct charge transport mechanism was initiated.

2. Electrically Actuated Band Release Device (EABRD)

The EABRD is an electrically fired one-shot actuator for release of mechanical energy. It is suitable for many safety applications, such as initiation of destruct charge release, shield plug removal, and reflector ejection.

The EABRD will separate on ground command, electrical malfunction command, or programmed electrical timer command. Electrical current is supplied to a heater located inside the structural housing. The heat reduces the structural strength until it is exceeded by the applied tensile load and the housing separates. The latest configuration EABRD is shown in Figures 83 and 84.

a. EABRD Housing

Previously, the EABRD had been designed for tensile load of 1500 lb, which resulted in a relatively thick-walled (0.020 in. nominal) tube section. A re-analysis was made, showing that a 600-lb tensile load would be enough to overcome

static and dynamic loading in orbit. A 25% relaxation factor was added for spring degradation, which with a safety factor of 25% for overall loading results in a design load of about 1000 lb. This load, combined with strength data for Rene' 41 from vendor literature and tensile tests on SNAP 10A EABRDs, resulted in a redesigned EABRD with a 0.012-in. nominal wall thickness.

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Figure 83. SNAP 8 Electrically Actuated Band Release Device (EABRD)

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Figure 84. X-Ray Photograph of EABRD

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Preliminary tests of this configuration show very good separation characteristics (see Table 53).

A test program is in progress to determine both the short-term tensile strength of Rene' 41 EABRD housings, and long-term stress-rupture and creep effects. Preliminary results show that unit strength decreases with material thickness due to surface effects (see Table 54). These results correlate with published data, which give lower strength values for Rene' 41 sheet than for bar stock. Additional long-term tests now in progress will give sufficient data in order to

make a Larson-Miller plot of stress-rupture strength in the tubular configuration.

b. Heater

Evaluation of the heater design and development of fabrication processes were continued. In the present design (Figure 84), a 0.040-in.-diameter tantalum wire is passed through the center of a lavite bobbin and a 0.020-in.-diameter tantalum wire is wound around it; the tantalum wires are joined together at one end of the bobbin and joined to individual lead wires

TABLE 53
EABRD FIRING TEST DATA

EABRD Sample Number	Housing Material	Temperature (°F)	Tensile Load (lb)	Heater Resistance (ohms)	Line Resistance (ohms)	Wall Thickness (in.)	Heater Wire Diameter (in.)	Material	Time to Separate (sec)	Total Energy (watt-sec)	Remarks
22	Rene' 41	650	650	0.294	0.54	0.0125	0.020	Tantalum	11.7	2800	Satisfactory separation
24	Rene' 41	70	600	0.310	0.54	0.012	0.020	Tantalum	16.4	4000	Satisfactory separation

TABLE 54
RENE 41 TENSILE AND STRESS-RUPTURE TEST DATA

Sample Number	Temperature (°F)	Environmental Pressure	Load (psi)	Area (sq. in.)	Stress (psi)	Failure Time (hr)	Remarks
10	900	Ambient	3930	0.0250	157,000 ultimate	—	0.021-in.-wall short-term tensile test. Yield strength = 138,000 psi
None	900	Ambient	702	0.0053	132,400 ultimate	—	0.009-in.-wall SNAP 10 housing. Yield strength = 95,200 psi
None	900	Ambient	750	0.0054	139,000 ultimate	—	0.0095-in.-wall SNAP 10 housing. Yield strength = 103,700 psi
0	880	10 ⁻⁵ torr	3600	0.0250	144,000	660	Stress-rupture test
14	900	10 ⁻⁵ torr	3800	0.0250	152,000	1188	Stress-rupture test
23	900	Ambient	1865	0.0128	145,000 ultimate	—	0.011-in.-wall short-term tensile test. Yield strength = 134,000 psi

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TABLE 55
PERCUSSIVE WELD TEST DATA

Type of Weld	Number of Samples	Load to Rupture (lb)	Remarks
0.020-in. Tantalum to 0.040-in. Tantalum	5	14 to 27	4 samples broke at or near the weld at a minimum of 23 lb. The 5th broke at the holder at 14 lb.
0.020-in. Tantalum to 0.040-in. Copper	6	23.5 to 24.5	All broke at or near the weld.
0.040-in. Tantalum to 0.040-in. Copper	6	28.5 to 38	All broke at or near the weld.

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Figure 85. Metallograph of 0.020-in. Tantalum to 0.040-in. Stainless-Steel-Clad Stranded Copper Wire Weld Specimen Potted With PBX-Alkophos, After 7 Days Storage

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(stranded copper) at their other ends. Tantalum was selected because of its high melting point, high resistivity, and ease of winding. The central wire was made larger to reduce its heating effect. The assembly uses a PBX-Alkophos potting compound to insulate it from the housing.

Satisfactory techniques were developed to make the wire terminations by the percussive welding method. Tantalum-to-copper and tantalum-to-tantalum weld specimens were tensile tested to evaluate the joining technique. The test results listed in Table 55 indicate satisfactory weld strength. Similar specimens showed sound welds when sectioned and examined metallographically (Figures 85 and 86).

A program was initiated to determine compatibility of the tantalum and copper wires with the PBX-Alkophos potting compound. Welded sections of tantalum-to-tantalum and tantalum-to-copper wire were potted and cured for various times, at temperatures from 250 to 800°F, in both air and vacuum. The extreme case was considered to be maintaining the potted assembly in the wetted condition for three days prior to air curing. After one-week to one-month shelf storage, the specimens were sectioned and examined metallographically (Figures 85 and 86). No evidence of corrosion was found in the total of 32 samples.

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Figure 86. Metallograph of a 0.020-in. Tantalum to 0.040-in. Tantalum Weld Specimen Potted With PBX-Alkophos, After 30 Days Storage

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I. SHIELD

1. Introduction

The program during the past year has maintained fairly broad coverage of shielding development problems with emphasis recently being directed toward manrated applications. As investigations of certain basic lithium hydride performance questions, such as material compatibility, hydrogen migration tendencies, and hydrogen loss rates through punctures, have been accomplished, emphasis has been directed toward the development of fabrication and design techniques. Two structural mockup studies have been in progress: one, to investigate internal support concepts and lithium hydride (LiH) stabilization methods; the other, NaK line penetrations and an aggregate gamma-shield region. The use of a novel method of achieving a bond between LiH and the shield structure by means of backfilling shield voids with lithium metal was also investigated.

In the materials area, a high-temperature braze material, Haynes Stellite 157, was found to have an apparently high resistance to attack by molten LiH. The study of gamma shielding materials continued to be broadly based and is directed by the results of analytical studies of optimum shielding designs.

2. Braze Compatibility with Lithium Hydride

The extremely corrosive action of molten LiH on the majority of commercial brazing alloys has been observed in shield development programs for many years. Until recently, there has been no known braze which could withstand attack for even a few hours. A search of commercial products lead to the selection of Haynes Stellite 157 for evaluation. This material is composed of 70.5 Co, 21 Cr, 4.5 W, 2.4 B, and 1.6 Si. Several 304 stainless-steel braze joint

configurations were prepared and fired in hydrogen at 2200°F. Good braze flow was attained and the resultant braze on the joints was excellent. The specimens were then exposed to LiH at 1300°F for 100 hours to simulate a shield casting cycle. Post-exposure visual examination revealed no evidence of attack of either the braze material or the base metal. One T-shaped specimen was bent flat without indication of braze failure. A longer term exposure of specimens is in progress.

One problem with this braze material is its high brazing temperature (2200°F). This presents limitations on its application to shield casing fabrication.

3. Hydrogen Migration Studies, Gas-Phase System

Evaluation of the results of the 2000-hour, gas-phase hydrogen migration test^{*} was completed. In this test, six chambers containing various amounts of LiH were connected to a common manifold. Each chamber was maintained at a different temperature between 700 and 1100°F. The test conditions are indicated in Table 56.

TABLE 56
GAS-PHASE MIGRATION
TEST CONDITIONS

Capsule	LiH Weight (gm)	Average Temperature (°F)
1	500	753
2	125	793
3*	60	500
4	25	1000
5	62	908
6	10	1105

*Capsule 3 was a control capsule removed before initiation of the test

*C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

A summary of the results of the chemical analyses is shown in Table 57. Each result is an average of at least three determinations from various regions of each capsule.

A plot of the hydrogen composition as a function of temperature is shown in Figure 87. It is significant that only a small trend (~5%) toward hydrogen relocation is indicated, even though the large amount of cooler LiH would be expected to contain more than enough free lithium to react with all the hydrogen from the 1100°F material. The relatively high level of oxygen in Capsules 4 and 6 gives an indication that minute inleakage may have occurred during the course of the test, but this does not affect the conclusion that the amount of migration was, at most, very small.

This test concludes the study of hydrogen migration in LiH shields. There is no indication that a problem exists in SNAP shields because of the phenomenon. The effect of a radiation field on the process has not been experimentally investigated; however, no work in this area is considered necessary at this time.

4. Gamma Shield

The use of uranium alloys as gamma shielding in manned systems requires thorough in-

vestigation of the thermal properties of the selected alloys. The thermal stability of massive shielding regions under thermal transient conditions associated with reactor startups and earth shadow cyclic effects is a major performance question. An initial effort involved the measurement of the thermal expansion characteristics of U-10 Mo specimens under thermal cycling from room temperature to 1650°F, at a rate of 300°F/hr. In this test, a Leitz dilatometer was employed. The mean linear coefficient of thermal expansion over the range was 8.7×10^{-6} in./in.-°F.

The production of secondary gamma rays through a neutron capture in tungsten and uranium gamma shielding causes a loss of effectiveness as thicknesses approach those required for certain manned shield applications. Since these gamma rays are, in most part, caused by the capture of resonance energy neutrons, the use of a poison additive, such as boron, is not particularly effective. There is considerable interest in the use of lead and lead compounds as gamma shielding for certain applications. Lead has an extremely low rate of secondary gamma-ray production.

To avoid the phase change and other problems that would be associated with the use of molten lead, a lead glass offers possibilities in the

TABLE 57
GAS-PHASE MIGRATION
CHEMICAL ANALYSIS

Capsule	OH* (wt %)	Li (wt %)	H (wt %)	CO ₃ (wt %)
1	1.31	86.6	12.4	0.12
2	1.39	86.5	12.3	0.10
3	1.75	86.4	12.1	0.10
4	2.97	85.8	12.2	0.11
5	1.71	86.4	12.1	0.07
6	4.84	84.1	11.9	0.09

*The chemical analysis method does not distinguish between the hydroxide and oxide forms. This should be considered as the OH equivalent value.

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Figure 87. Hydrogen Content of Gas-Phase Hydrogen Migration Test After 1968 Hours

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600°F and above temperature range. The feasibility of casting regions of lead glass was investigated on a small scale. A 5-lb charge of the oxides of lead and boron were melted in a platinum crucible. The composition corresponded to $4\text{PbO} \cdot 1\text{B}_2\text{O}_3$ or about 11 wt % B_2O_3 . The melting point of this glass is about 920°F. The melt was poured into a preheated Pyrex beaker and was cooled. To maintain integrity of the casting in the event of cracking, a stainless-steel metal-wool matrix was incorporated into the casting. The resulting casting is shown in Figure 88. The casting was very strong, free of cracks, and had a density of 6.91 gm/cc.

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Figure 88. Lead Glass Casting with Metal Wool Stabilizer

5. Lithium-Metal Backfilled Shield

The postmortem evaluation of the lithium-metal backfilled shield* and the assessment of the lithium on the thermal conductivity of the shield were completed. Removal of the shield casing was conducted in the large drybox at the shield casting facility. The atmosphere in the box (97% N_2 , 3% O_2 , 10 ppm water) is reactive

with the lithium metal, but is safe from the combustion standpoint.

Major questions in the evaluation of this shield were the ability of Li metal to wet and penetrate into the cracks in the LiH block and the degree of bonding afforded interfaces. X-ray results, reproduced in Figures 89 and 90, indicate that excellent penetration of shield voids occurred.

Upon removal of the shield casing in the dry-box, Li metal was found to exist in a distinctly metallic phase. The bond between the shield casing and the LiH was extremely tenacious, requiring multiple cuttings of the casing shell and considerable prying to remove individual pieces of casing. It is significant that this bond was maintained through very high-level vibration and shock tests immediately prior to the disassembly activity. Soon after removal of individual portions of the casing, the Li metal changed to a deep red color — apparently the nitride form. The shield is shown with casing removed, in Figure 91. Further disassembly revealed the cast block to be extremely tough and resilient. Attempts to cleave the block by means of chisels were essentially futile. The as-cast shields can normally be cleaved in this manner.

The highly beneficial effect of the lithium metal on the structural properties of the cast shield is considered to warrant additional investigation. The long-term effect of Li metal/LiH compatibility is currently being investigated in the laboratory.

The evaluation of the thermal transient response data on the shield was completed. The method employed was to determine the best fit value of effective bulk thermal conductivity

*C. E. Johnson, "Progress Report, SNAP 8, August-October 1964," NAA-SR-10492 (SRD), December 23, 1964

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Figure 89. Cast LiH Shield Before Lithium Metal Backfill

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Figure 90. Cast LiH Shield After Lithium Metal Backfill

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Figure 91. Lithium-Metal Backfilled Shield

which analytically reproduces the experimental transient. At temperatures below 700°F, where the shield was not completely filled with Li metal, satisfactory results were not obtained. In the 700 to 900°F range, however, good comparison was obtained. The best fit conductivity was found to be 3.7 Btu/hr-ft-°F, which compares with a value of 2.9 from tests on this shield before introduction of the lithium metal.

6. Shield Structural Mockup No. 1*

Evaluation of this mockup was completed in the large drybox facility. Initially, a circumferential cut was made in the shield casing above the points where the reactor support spider was attached to the shield mounting ring. The entire

upper shield casing was found to be free from the cast block. This was undoubtedly due to the stainless-steel wire screening that had been lightly spotwelded to the inside of the shield casing. On the other hand, in the shield lower head, the screen had been more heavily spotwelded to the casing and considerable force had to be applied to separate the casing and hydride block. It appears feasible to use this method to obtain or eliminate bonding to the casing wall, as desired.

The cast block was cut to a depth of about five inches in the plane of the axis. The block was then split, revealing the cross section shown in Figure 92. The crack patterns in the

*C. E. Johnson, "SNAP 8 Quarterly Progress Report, May-July 1964," NAA-SR-9992 (SRD), September 22, 1964

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Figure 92. Shield Mockup No. 1, Cross-Section View

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Figure 93. Shield Mockup No. 1 with Support Leg Removed

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shield were thus clearly revealed. The crack patterns in this shield were of particular interest because of the use of metal wool stabilizer instead of the conventional honeycomb. It is concluded that the extent of cracking in this shield would not be expected to affect its shielding capabilities significantly. It has not been established that the LiH stabilization technique can be adapted for a larger shield (Ex: S8DRM-1, 40 inches) because the diameter of this test shield was only 22 inches.

After cleavage of the block, the disassembly proceeded to the removal of a reactor support leg. As was previously observed from x-rays, the support legs were adequately supported by the cast LiH so that column buckling was resisted during shock and vibration tests. The shock and vibration tests were severe enough, however, to cause tension-induced fatigue cracks at numerous hole locations in the support tubes. In all cases, these incipient failures are radial from the hole locations and in a plane perpendicular to the tube axis. Figure 93 shows a support leg removed from the shield.

7. Shield Mockup No. 3

Fabrication of this assembly* progressed to 90 percent of completion. The top view of the assembly before loading the aggregate tungsten material is shown in Figure 94. The NaK lines with glass tape to simulate a higher performance insulation are also shown. A bottom view of the same assembly is shown in Figure 95.

8. Rib Test Specimen

The casting of the rib test specimen was completed. Evaluation of the criss-cross matrix and other features of this small test speci-

men has not yet been accomplished. The titanium-dioxide opacified porcelain-enamel emissivity coating appears to have withstood the distortions and other conditions associated with the casting cycle. Further evaluation is required to assess the overall performance of this coating and the post-casting emissivity characteristics.

Figure 94. Shield Mockup No. 3, Top View

Figure 95. Shield Mockup No. 3, Bottom View

*C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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VI. SNAP 8 EXPERIMENTAL REACTOR (S8ER)

A. INTRODUCTION AND 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 design of the flight reactor, especially with regard to reactor stability, temperature coefficients, power coefficients, and hydrogen loss rate.

Reactor operations were continuous throughout the report period until final reactor shutdown at 0604 hours on April 15, 1965.* Table 58 gives a summary of S8ER operations through final shutdown.

TABLE 58
SNAP 8 EXPERIMENTAL REACTOR
OPERATIONS SUMMARY

Total operating time	500 days (12,000 hours)
Total time above 400 kw at 1300°F	365 days
Time at 600 kw at 1300°F	100 days
Longest continuous run (September 18, 1964 to April 15, 1965)	Over 5000 hours
Energy release	Over 5.1×10^6 kwh
Operating efficiency from first power operation on November 22, 1963 to com- pletion of power operation on April 15, 1965	91%

B. OPERATIONS

The S8ER established what is believed to be the longest record for continuous reactor oper-

ation on February 12, 1965 when it exceeded the longest known record (147 days) held by the Yankee reactor at Rowe, Massachusetts.

The S8ER operated continuously during the report period. Reactor power level was maintained at 450 kw except for fourteen days during which reactivity change rates at 600 kw were measured. Operation at 450 kw was performed at a core outlet temperature of 1300°F with temperature increases of 200 and 300°F across the core, and at a core outlet temperature of 1200°F with a temperature increase of 200°F across the core. This test provided data to measure the effect of core temperature conditions on reactivity change rates.

S8ER operations during the report period included:

- 1) Completion of a total of 100 days of operation at 600 kw with a core outlet temperature of 1300°F,
- 2) Completion of a total of a year of operation at 400 kw or greater with nominal core outlet temperature of 1300°F,
- 3) Completion of a total of 10,000 hours of operation at 400 kw or greater, and
- 4) Completion of a total of 500 days of nuclear operation.

C. EVALUATION OF EXPERIMENTAL RESULTS

1. Xenon Poison Worth

The xenon transient experiment measurements have been reevaluated and compared with subsequent xenon transients observed during operational power changes to produce corrected worths for equilibrium xenon poison as shown in Figure 96.

*L. M. Fead, L. D. Felten, and V. L. Rooney, "SNAP 8 Experimental Reactor Operations and Test Results," NAA-SR-10903 (CRD), to be published

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Figure 96. S8ER Xenon Poison Worth

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Figure 97. Variation of the S8ER External Source Strength With Degree of Subcriticality

TABLE 59
S8ER HYDROGEN LOSS RATES

Dates	Nominal Conditions			Hydrogen Loss Rate (#/day)
	Power (kw)	Outlet Temperature (°F)	ΔT (°F)	
1-3 to 1-21-65	450	1300	200	3.19
1-21 to 2-4-65	600	1300	200	3.99
2-4 to 2-17-65	450	1300	200	2.48
2-17 to 3-3-65	450	1300	300	1.10
3-3 to 3-10-65	450	1300	200	3.09
3-10 to 3-26-65	450	1300	300	1.50
3-26 to 4-15-65	450	1200	200	0.60

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The error of 7% possible in the above absolute values is the same error possible in measurement of absolute power level.

2. Samarium Poison Worth

Final evaluation of the samarium poison experiment measurements has resulted in the determination of $\$1.47 \pm 31\%$ as the worth of the initial samarium oxide prepoison above equilibrium samarium poison worth.

3. Hydrogen Loss Rate Measurements

Hydrogen loss rates measured for 450 and 600 kwt operation are presented in Table 59. The loss rates were determined by a first order least-squares fit corrected for samarium, uranium depletion, and fission product buildup effects.

4. Hydrogen Loss Rate Comparison

An experiment at 450 kwt, 1300°F core outlet temperature, to compare the hydrogen loss rates at 200 and 300°F temperature increases

across the core was repeated. The first run of the experiment indicated that the hydrogen leakage was generally concentrated at the outlet (hot) end of the reactor.* The second run of the experiment indicated that the hydrogen leakage was generally concentrated at the inlet (cold) end of the reactor. The core exposures were 3.5×10^6 and 4.9×10^6 kwh, respectively, for the experiments. The difference in experimental indications is tentatively attributed to fuel phase-change phenomena which render standard hydrogen loss formulas inadequate.

5. Source Strength Determinations

The experimentally observed strength of the Po-Be neutron source in S8ER is shown in Figure 97. The number of neutrons reaching the core from each curie of polonium-210 in the source was equivalent to 3 microwatts. The neutron population level in a given subcritical configuration was about $3 \times 1/(1 - k)$ microwatts per Po-Be curies.

*C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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

A. INTRODUCTION AND SUMMARY

The SNAP 8 developmental reactor mockup (S8DRM-1) is a nonnuclear prototype system which will be used to verify the reference design. As this system is the first attempt to verify the SNAP 8 nuclear system, some failures are to be expected. Any failed components will be replaced or redesigned, if necessary, for further testing.

The main objectives of the S8DRM-1 program are to: (1) assist the development program in the design of reference systems and components; (2) test the adequacy of fabrication techniques to provide reference systems and components; (3) demonstrate the capability of assembling all individual systems and to assemble all systems to form the nuclear system; (4) demonstrate that systems and components can operate satisfactorily before, during and after simulated environmental conditions; and (5) demonstrate that systems and components can withstand simulated environmental conditions.

During this reporting period, the startup reliability test was completed.

B. ACTUATORS

All actuator stator coils were wound using 27-gauge wire. The units were encapsulated and satisfactorily passed all tests. Three actuators were completely fabricated and passed dimensional checks. These units are now being subjected to an acceptance test.

C. BEARINGS

New reflector control drum bearings were fabricated. They are similar to the S8DS bearings in design and diametral clearances. The

lower bearings were inspected and are acceptable. The upper bearings are in the inspection stage.

D. SYSTEM TESTS

The objective of this project is to perform nonnuclear environmental and operational tests on the S8DRM-1 system to provide data for the development of nuclear systems in the areas of startup, performance, and endurance. Testing includes the simulation of thermal-vacuum, shock, and vibration environments. The purpose of these tests is to determine whether the assembly will perform in accordance with design criteria. The tests are performed in addition to component and assembly acceptance tests; the latter demonstrate that each component is operational, but do not guarantee that the same component will sustain reliable operation when integrated into a nuclear system subjected to long-term endurance and environmental conditions.

During this reporting period, a detailed examination of the reactor system was conducted to determine any further effects of the shock and vibration tests. The S8DRM-1 was then subjected to a performance record test, a startup reliability test, and another performance record test, in that order.

A complete inspection of the reactor system following the shock and vibration tests revealed malfunctions in addition to those immediately observed and listed previously:^{*}

- 1) A circular section of shim cloth was cut from the original section of cloth (see Figure 98). Note that the shim cloth has

^{*}C. E. Johnson, "SNAP 8 Progress Report, November 1964-January 1965," NAA-SR-10792 (CRD), March 15, 1965

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Figure 98. S8DRM-1 Gamma Shield Shim Cloth After Reactor System Shock and Vibration Test

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Figure 99. Typical S8DRM-1 Control-Drum Drive Coupling After Reactor System Shock Vibration Test

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been cut completely in two and the inner portion has been ground into a powdery form at the outer periphery. The apparent cause was the relative movement between the bottom of the gamma shield and the top of the neutron shield during vibration tests.

2) Two of the three coupling assemblies failed (see Figure 99). The exact cause of these failures is under investigation.

3) Three of the four insulators used to separate the reflector from the core vessel failed during Z-Z axis vibration at qualification levels (see Figure 100). The failure is attributed to the low impact strength of the insulator material coupled with a relative movement between the stationary reflector and the core vessel during the vibration tests. A simpler, more adequately designed insulator-shim will be used on S8DS.

A portion of the S8DRM-1 program consists of demonstrating 20 automatic startups. The objective of the startup reliability test is to demonstrate repeatability of the automatic startup sequence. The first series of nine startups was noted in an earlier progress report.* Nine more startups are being completed during this reporting period and two additional startups will be performed.

The S8DRM-1 assembly was placed in a chamber (Figure 101) at atmospheric pressure and ambient temperature. The startup procedure duplicated that of a nuclear startup. The startup programmer, controller, and relay box (supplies power for firing squib igniters) were utilized in the startup sequence. The results of the tests are tabulated in Table 60.

Prior to startup sequence Number 2, the controller (a breadboard model) was inadvertently

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Figure 100. Typical S8DRM-1 Insulator-Spacer After Reactor Assembly Shock and Vibration Test

*C. E. Johnson, "SNAP 8 Progress Report, August-October 1964," NAA-SR-10492 (SRD), December 23, 1964

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Figure 101. S8DRM-1 Assembly Mounted in Environmental Test Chamber
for Startup Reliability Test

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TABLE 60
S8DRM-1 TEST SUMMARY

Test	Results	Cause of Exception	Apparent Effect Exception Would Have on Orbital Startup
Startup Sequence No. 1	Entire system functioned satisfactorily with no exceptions.	—	—
Startup Sequence No. 2	The system performed satisfactorily with one exception: one squib igniter did not fire.	Squib connector had a faulty connection.	Successful startup — Each drum-lock release actuator contains redundant squib igniters. The firing of one squib is sufficient to operate the actuator
Startup Sequence No. 3	System functioned satisfactorily with two exceptions: 1) one squib igniter did not fire; 2) one startup drum did not latch in the full-in position.	a) Trouble shooting after Test No. 5 revealed relay box fusistor failure. b) Lower drum support bearing caused excessive "drag" on drum. The exact cause of bearing malfunction is to be determined during installation of new bearings at a later date.	a) Successful startup — Each drum-lock release actuator contains redundant squib igniters. b) Successful startup — The drum attained a position approximately 98% full-in.
Performance Record Test No. 1	Entire system functioned satisfactorily with no exceptions.	—	—
Startup Sequence No. 4	System functioned satisfactorily with one exception: one squib igniter did not fire.	Trouble shooting after Test No. 5 revealed relay box fusistor failure.	Successful startup — Each drum-lock release actuator contains redundant squib igniters.
Startup Sequence No. 5	System functioned satisfactorily with one exception: two squib igniters did not fire (not located on same drum lock release actuator)	Three of the fusistors in the relay box failed, causing loss of power to the igniter circuits.	Successful startup — Each drum-lock release actuator contains redundant squib igniters.
Startup Sequence No. 6	Entire system functioned satisfactorily with no exceptions.	—	—
Performance Record Test No. 2	Entire system functioned satisfactorily with no exceptions	—	—
Startup Sequence No. 7	System performed satisfactorily with one exception: limit switch "B" did not operate as required.	Relative motion between the switch housing and the contacts	Successful startup — An automatic switching device is built into the programmer system to override a failure of either limit switch "A" or "B."
Startup Sequence No. 8	Entire system functioned satisfactorily with no exceptions.	—	—
Startup Sequence No. 9	Entire system functioned satisfactorily for 105 minutes. The controller malfunctioned at this time necessitating a premature shutdown.	The cause of this malfunction is under investigation.	Incomplete startup — The reactor functioned properly but the breadboarded controller malfunctioned.
Performance Record Test No. 3	Entire system functioned satisfactorily with no exceptions.	—	—

triggered by electromagnetic interference (EMI) generated by a welding machine located in the test building. A controller used in a flight system would normally be designed to resist EMI.

The objective of the performance record tests is to demonstrate the operational status of the reactor system before and after all signifi-

cant system tests. These tests also include a check of dimensional clearances, wiring circuit internal resistances, wire insulation resistances, and the rotational torque required to rotate the spring-driven startup drums. The results of the performance record tests were satisfactory; the startup reliability test had no effect on system operation.

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VIII. SNAP 8 GROUND PROTOTYPE SYSTEM (S8DS)

A. INTRODUCTION AND SUMMARY

The current objectives of the S8DS program are: to determine the steady-state and transient performance characteristics of a SNAP 8 flight reactor at a variety of operating conditions including higher-than-rated power and temperature; to demonstrate satisfactory operation of the improved fuel element design and the automatic reflector control features; to develop and demonstrate techniques of automatic startup and control; to determine the nuclear and thermal operating characteristics of SNAP 8 shielding materials in a typical radiation and temperature environment; and, if possible, to test the reactor as a component of a nuclear electric power generating system.

Reactor and neutron shield fabrication and assembly of acceptance test tooling progressed. The tooling vessel, the shield can, the beryllium reflector drums and shims, and a large quantity of the reflector bracketry were completed. Drum-drive actuator hardware neared completion and the winding of actuator coils was started.

Control-drum worths were calculated for 3- and 4-inch thick drums and stresses were determined for the reactor vessel and neutron shield.

B. SYSTEM ANALYSIS

1. Control Drum Worths

Figures 102 and 103 show the calculated differential and integral drum worths for the S8DS reactor. The drum configurations examined are the nominal shim (3.125-in.-thick drum) and the one-inch shim (4.125-in.-thick drum). The values, scaled from SCA-4A critical experiments, are estimated to be accurate to $\pm 10\%$ to $\pm 15\%$.

2. S8DS Flow Profile Data

At the conclusion of the S8DS baffle development program, tests were run to determine the flow distribution in the core model at several flow rates. These data were characterized by a large scatter band, $\pm 20\%$, around a least-squares fit of the data. It was noted that the repeatability of the data points was improved, being less than $\pm 10\%$. This variance is probably attributable to the properties of the noise analysis electrodes.

Figure 102. Differential Drum Worth vs Drum Position

Figure 103. S8DS Integral Drum Worths

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TABLE 61
S8DS OPERATIONAL PHASE, REACTOR VESSEL MINIMUM MARGIN OF SAFETY

Part	Design Condition			Maximum Stress (psi)	Allowable Stress (psi)	Margin of Safety
	Pressure (psia)	Mass (ft-lb)	Temperature (°F)			
Core Vessel Cylinder	70	—	1,350	3,550	3,700	+0.04 (creep)
Upper Plenum	70	50	1,350	2,865	3,700	+0.29 (creep)
Lower Plenum	70	—	1,150	8,013	10,200	+0.27 (creep)
Inlet Nozzle	70	50	1,150	6,143	10,200	+0.66 (creep)
Core Vessel	—	—	Operation of S8DS for 12,000 hours at 1350°F	3,550	3,500	-0.01 (creep)
Internal Reflector Cladding	—	—	600 kw for 10,000 hours at 1350°F	(1) 6,900 (2) 7,460 (3) 6,700	8,400 (Stress Rupture)	(1) +0.30 (2) +0.20 (3) +0.34

The original calibration tricuspid channel, as well as each instrumented fuel rod, contained six electrodes. A least-squares fit of reciprocal velocity vs transit time gives an average root-mean square (RMS) deviation of 0.006 for individual electrode pairs. When these were combined into one curve fit, however, the RMS deviation was 0.068, indicating different calibrations for the individual electrode pairs. A test was then made reversing the flow in the calibration channel and the transit times at comparable flows were compared to the forward flow data. Differences were noted for several of the electrodes. These differences can be treated as a net delay in response for the pair. In one direction the delay will decrease the observed time, while in the reverse direction the observed time will be increased. The delay time would be one-half the time differences. When this correction was applied to the original calibration data the RMS deviation for the combined curve fit was 0.025.

The next step was to apply the forward-reverse technique to the instrumented fuel rods in the hydraulic model. This was done by binding three fuel rods together and performing tests similar to those described above. Although the core calibration data is not completely analyzed, the results should decrease the scatter band to $\sim \pm 10\%$.

3. Stress Analysis of S8DS Core Vessel Assembly

The detailed stress analysis of the core vessel assembly covering all significant conditions of operation was completed. The analysis is divided into two major parts:

a. Operational Environment

The nuclear system is designed for 10,000 hours of continuous operation at a core temperature of 1350°F and internal pressure of 50 psia (ultimate design pressure = $1.4 \times 50 = 70$ psia),

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TABLE 62
S8DS LAUNCH ENVIRONMENT, MAXIMUM ALLOWABLE LOAD FACTOR

Part	Design Condition			Allowable Stress (psi)	Critical Stress Condition	Allowable Design Load Factor (g)
	Load	Pressure (psia)	Temperature (°F)			
Core Vessel Cylinder	Lateral	70	200	3,500	buckling	43 ultimate 31 operating
Grid Plate - Lower	Vertical	—	200	49,500	yield	110 yield 46 operating
Grid Plate - Upper	Vertical	—	200	26,000	yield	56 yield 48 operating
Lower Grid Plate Support	Vertical	—	200	26,000	yield	196 yield 170 operating
Upper Grid Plate Support	Vertical	—	200	26,000	yield	115 yield 100 operating
Reflector Positioning Lug	Lateral	—	200	26,000	yield	106 yield 92 operating
Reflector Support Bracket	Vertical	—	200	26,000	yield	74 yield 64 operating
Baffle Plate	Vertical	—	200, 1,150	26,000 15,000	yield yield	high high
Reflector Support Lug Analysis	Vertical	—	200	26,000	yield	86 yield 75 operating

and internal pressure of fuel vacuum. An analysis showing the effects of operating the reactor for 12,000 hours is included.

b. Launch Environment

Although the S8DS will not be subjected to the launch environment, the system was analyzed to determine the ability to withstand launch. Major structural consideration during the launch phase includes acceleration, shock, and vibration load factors. Since the dynamic characteristics of the S8DS structure are not completely known at this time, the analysis for launch environment will only indicate the maximum design load factor "N" to which the components may be subjected without failure.

A summary of the margins of safety, which includes a factor of safety, is presented in

Tables 61 and 62. In the analysis of SNAP 8 structures and components, two factors of safety are used: the first is 1.4 based on the ultimate strength (rupture, stability, and creep) of the material in question; and the second factor is 1.15 based on the yield strength of the material. The operating load when multiplied by these numbers equals the ultimate design load and the yield design load, respectively. The margin of safety (MS) is the ratio of the excess strength to the required strength, or the percentage by which the strength of a member exceeds the design load.

$$\begin{aligned} MS &= \frac{F - \sigma}{\sigma} \\ &= \frac{F}{\sigma} - 1, \end{aligned}$$

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TABLE 63
S8DS SHIELD AND CONFINEMENT VESSEL MARGINS OF SAFETY

Part	Allowable Internal Pressure at		Thermal Stress (psi)	Margin of Safety
	Room Temperature (psi)	Operating Temperature (psi)		
Head Assembly - Lower Ring	6.50 (yield)	43.1 (fatigue)	38,532 (tension)	+0.05
Diaphragm	7.19 (yield)		30,420 (compression)	+0.32
Head Assembly - Upper Cylinder	17.8 (yield)			
Ring	11.4 (yield)			
Head	5.51 (yield)	28.5 (fatigue)	22,754 (compression)	+0.33
Confinement Vessel Upper Head		217 (buckling)	46,680 (compression)	Negative
Conical Body		227 (buckling)		
Lower Head		191 (buckling)		

where

F = allowable stress, and

σ = calculated stress (including safety factor).

In terms of ultimate and yield strengths, the margin of safety is calculated as follows:

$$MS(\text{ultimate}) = \frac{F_{\text{ult}}}{\sigma_{\text{ult}}} - 1$$

$$MS(\text{yield}) = \frac{F_y}{\sigma_y} - 1$$

$$= \frac{F_y}{1.15 \sigma_{\text{ult}}} - 1$$

The design is governed by the lowest margin of safety.

4. S8DS Shield and Confinement Vessel

A similar analysis of the S8DS shield and confinement vessel was completed. Margins of safety were calculated using a factor of safety of 1.15 on the material yield strength and 1.5 on the ultimate strength.

The results indicate the following limits should be set on shield internal pressure:

a) 5.51 psi, if no yielding of the structure is desired at room temperature; and

b) 28.5 psi, if no failure in 500 shutdown-startup cycles is used as a failure criteria.

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The margin of safety calculations indicate that some yielding of the confinement vessel upper head could occur due to thermal stresses, but will not be detrimental to normal confinement vessel operation.

A summary of the margin of safety and allowable load is presented in Table 63.

C. REACTOR CORE VESSEL AND NEUTRON SHIELD

Fabrication of the reactor core vessel and neutron shield progressed with the completion of the tooling core vessel (Figure 104), the fabrication of the shield can (Figure 105), and the initiation of casting the Li^7H in the shield can.

D. REFLECTOR AND DRIVES

Fabrication of the reflectors progressed with the completion of the beryllium drums and shims and most of the reflector bracketry.

Several modifications have been incorporated in the reflector and drives to improve functional reliability and to provide greater operational flexibility.

All drum-drive and scram mechanism bearings have been re-evaluated and some of the bearing clearances have been increased to an extent consistent with the S8DS control-drum position accuracy requirements. The scram mechanisms have been modified, as shown in Figure 106, to provide the means for locking out the scram capability and driving the drum to the full-in position by remote control. The incorporation of these additional features is based on the following considerations:

1) In the event of an electrical failure in the scram clutch, the locking arrangement will allow the drum connected to that clutch to be moved by the actuator and to be used as a nonscramming startup drum.

2) In the event of an actuator failure, the remote drum drive arrangement will allow the drum to be moved to the full-in position where it can then be locked and used as a nonscramming startup drum.

3) During the S8DS tests it may be desirable to operate the reactor for a period with the scram capability removed from one or more drums.

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Figure 104. Completed S8DS Reactor Tooling Vessel

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Figure 105. S8DS Shield Casing

While the remote scram lockout and drum drive features do not offer drum rotation capability for all possible combinations of scram-drive system component failures, it does provide the means of overcoming the most probable component failures. The final backup is the capability of replacing a complete reflector half assembly.

E. DRUM-DRIVE ACTUATORS

Drum-drive actuator hardware neared completion and the winding of the actuator stator coils was started. Two development actuators were successfully fabricated and tested (see Section V-B) to prove the assembly techniques to be used on the system parts.

F. CABLE HARNESS

The plan for percussion welding of the S8DS cable harness to the component connectors after the components were installed on the reactor was tried and shown to be unsuccessful (see Section V-C). The harness design was modified to utilize bench-installed pigtailed on the components and terminal boards for the harness-to-pigtail connections. Lugs on the pigtailed and the harness will be bolted to the terminal boards for checkout and subsequently welded together to improve the reliability of the connection.

G. ASSEMBLY AND ACCEPTANCE TESTS

Two laminar-flow-type "clean" work benches (Figure 107) were installed in the SNAP 8 assembly and acceptance test laboratory environment

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Figure 106. S8DS Scram Mechanism

for conducting ambient acceptance tests of components. A large 20- by 28-ft laminar flow "clean" room for subassembly of the reflector halves and assembly and integration of the final reactor system was designed and fabrication initiated.

A digital data logger system capable of monitoring and recording up to 300 channels of information (Figure 108) was also installed in the acceptance test laboratory. Signal conditioners are incorporated so that thermocouple temperature measurements are recorded directly in degrees Fahrenheit, and drum rotation directly in degrees of rotation. The data logger will also monitor and record a-c or d-c voltage measurements and resistance measurements. Automatic ranging is incorporated in the a-c, d-c, and resistance portions of the data logger. The signals may be intermixed and programmed from a front panel patch board. Results are recorded on a Frieden Flexowriter and punched tape.

Two universal test controllers were fabricated (see Figure 109) for acceptance testing control drum actuators. They are manually controlled to allow for flexibility in acceptance testing of control-drum actuators and do not incorporate temperature actuated rate change mechanisms or other automatic drum reversal mechanisms. Change in direction of drum rotation is accomplished by manual switches on the front side of the panel. They do provide means for determining the operating characteristics of each actuator coil under ambient and thermal-vacuum conditions. In addition, the design is such that the pulsing rate can be varied.

A scram mechanism programmer was fabricated (see Figure 109) to automatically rewind the scram mechanism, reverse drum direction, and allow the drum to step to the full-in position. The programmer is designed to allow for an automatic scram from the full-in position or from the 75-degree position. It also incorporates a feature which permits a manual scram

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Figure 107. Laminar Flow "Clean" Work Bench

from any position. The programmer, like the test controller, is utilized in ambient and thermal-vacuum acceptance tests of the scram mechanism.

H. TEST SUPPORT EQUIPMENT

Tests on the main heat transfer pumps, conducted at the large component test-loop, indicate that the existing pump blowers are not adequate. The coil temperatures at full pump power of 40 kwe and NaK temperature of 1100°F

were 550°F for one pump and 527°F for the other. Blowers with larger capacities will be installed to maintain cooler coil temperatures.

I. FACILITY MODIFICATIONS

A thermal analysis of the final design of the vacuum vessel indicated that additional cooling coils were required on the bottom plate of the chamber. These additional coils have been installed and the vessel has been delivered to the facility for installation.

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Figure 108. Digital Data Logger System

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Figure 109. Universal Controller and Scram Programmer for S8DS Actuator Test

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