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SNAP REACTOR,
SNAP PROGRAM
M-3679 (56th Ed.)

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SNAP REACTOR PROGRAMS

PROGRESS REPORT

MAY-JULY 1968

(Title Unclassified)

By

J. G. ASQUITH

Exempt from CCRP Re-review Requirements

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Programs Progress Report, February-April 1968,"
AI-AEC-12674.

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I. INTRODUCTION

A. PROGRAM OBJECTIVES

The SNAP Reactor Program contains those tasks grouped under the SNAP 8, ZrH Reactor Technology, General Supporting Technology, and Aerospace Safety Programs. These four programs in total comprise the work leading toward the development, test, availability, and application of the zirconium-hydride reactor to space power.

The total SNAP Program objectives include:

- 1) Providing a zirconium-hydride reactor technology base for application with mercury-Rankine power-conversion equipment to a space system with mission capability in the early 1970's.
- 2) Providing the technology base and engineering for an uprated zirconium-hydride reactor with flexibility for use with thermoelectric, organic-Rankine, mercury-Rankine, or gas-Brayton-cycle power-conversion equipment.
- 3) Continually upgrading the reactor-shield performance as a means of achieving improved power system performance.
- 4) Achieving the reactor system performance within acceptable safety considerations.
- 5) Maintaining the nuclear reactor space power system work directed toward long-range national space objectives.

Specific near-term objectives are:

- 1) Continuation of the existing base technology component program supporting the immediate operation of the S8DR reactor.
- 2) Completing and testing of the S8DR reactor, with operation at 600 kwt and 1300°F outlet temperature and 1 Mwt and 1100°F outlet temperature.
- 3) Engineering of an uprated zirconium-hydride reactor with a 600 kwt, 20,000 hr, 1300°F outlet temperature capability. This reactor will have a 4π shielding capability, utilize an integral primary loop, and

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therefore, be compatible with all candidate primary power conversion systems.

- 4) Conceptual engineering of a 25-kwe thermoelectric system utilizing the uprated reactor and compact lead-telluride thermoelectric modules.
- 5) Performance of tests on components including fuel, 4π reflectors, burnable poisons, pumps, bearings, and shields to demonstrate the technical base for the uprated zirconium-hydride reactor.
- 6) Continuation of work with user agencies to incorporate the more significant mission-imposed constraints on the reactor program and to disseminate current knowledge of the program status to the using groups.
- 7) Performance of the required safety analyses and development testing to incorporate appropriate safety requirements into the uprated reactor.

B. PROGRAM SUMMARY

1. Fuel Development

Out-of-pile performance testing of S8DR fuel elements fabricated with normal uranium is continuing with 46 elements successfully completing the pre-endurance thermal-cycle tests.

Process parameters for coating Incoloy-800 clad fuel elements were established. Sixteen elements were fabricated and have begun thermal-cycle tests. Hydrogen leakage performance has been excellent after the process parameters were properly characterized. The biaxial stress-rupture tests of Incoloy-800 cladding tubes were completed and are reported.

In-pile operation of the NAA-120-4 and NAA-121-1 fuel-irradiation experiments is continuing at the Hanford facility.

2. Component Development

Design verification testing of three S8DR actuators and four sets of reflector bearings is approaching the 6000-hr point of the scheduled 12,000-hr tests.

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The dc conduction NaK-pump design for the zirconium-hydride reactor thermoelectric system was re-evaluated on the basis of the reduced pressure requirement for the primary loop. This resulted in selection of a two-throat pump which is considerably lighter than the earlier three-throat design.

Design of the 4π -shielded reflector version using a europia-nickel cermet as the control poison was completed. Samples of the cermet were successfully prepared.

3. SNAP 8 Developmental Reactor

Acceptance testing of the S8DR reflector assembly was completed. Minor modifications were incorporated into the manual controllers to minimize the possibility of operational scrams due to malfunction of the special safety network. The electric current output of the actuator controllers was increased to assure additional margin.

The S8DR core achieved dry criticality on June 22, 1968. Criticality was reached with 169 of the total 211 elements loaded in the core. Dry critical physics tests and control-drum calibrations were performed.

Checkout of the S8DR primary loop was completed and NaK was successfully circulated at operating temperature.

4. Reference Zirconium-Hydride Reactor

Conceptual design studies of the 4π -shielded reactor were concluded. The resulting reactor contains 295 S8DR-type fuel elements and has a beryllium-oxide radial reflector with ten poison-backed rotating control drums. The reactor can provide 600 kw of thermal power at 1300°F for longer than the 20,000-hr lifetime design objectives. Higher powers and longer lifetimes can also be accommodated at lower outlet temperatures.

5. Thermoelectric System

Conceptual design studies of the reference reactor power-conversion system (PCS) have been completed. The system utilizes the compact thermoelectric converter concept developed for the Atomic Energy Commission by the Westinghouse Astronuclear Laboratory. The resulting design divides the PCS into four independent converter assemblies each generating 6.3 kwe and containing 24 tubular

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thermoelectric modules. Each loop is thermally connected to the reactor by a NaK-to-NaK heat exchanger.

Parametric studies of the system were performed to characterize off-design operating conditions and to optimize unshielded system weights and radiator areas over the range of operation.

6. Aerospace Safety

The electrical and mechanical portions of the SNAPTRAN reactor system have been installed in the SNAP Engineering Test Facility (SETF). Pre-operational checkout is in progress.

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II. FUEL DEVELOPMENT

A. INTRODUCTION AND SUMMARY

One of two primary objectives of this program is to develop fuel elements to meet the requirements for the 600-kwt SNAP 8 reactor listed in Table II-1. In addition to these requirements the fuel-element cladding must resist corrosion in the NaK environment and maintain dimensional integrity over the life of the core.

TABLE II-1
PERFORMANCE GOALS FOR SNAP 8 REACTOR FUEL ELEMENTS

Lifetime	12,000 hr
Peak power density (at peak central temperature of 1500°F and peak fuel burnup of 0.3 met. at. %)	5.0 kw/ft of fuel-rod length
Hydrogen permeation	5% of H ₂ from fuel alloy
Checkouts required of hydrogen-barrier coating and closure seal	Thermal, mechanical, startup transients
Peak cladding temperature	1400°F
Reliability at 50% confidence level	99%

The second main objective is to understand and, as necessary, upgrade present ceramic-barrier metal-clad fuel-element technology to satisfy the requirements of the 20,000-hr, 600-kwt zirconium-hydride reactor system. Improved barriers, closures, and clad materials are under study to improve fuel-element performance. Irradiation experiments to further explore the behavior of the fuel and cladding materials and confirm fuel-element improvements are part of this program. The following paragraphs summarize progress this quarter.

Ten normal development elements clad in 15-mil Hastelloy-N have successfully completed the S8DR preproduction testing sequence and are now undergoing thermal endurance.

Process parameters for coating Incoloy-800 cladding and tentative process parameters for coating end-plugs were established. Incoloy-800 elements were

TABLE II-2
PERMEATION HISTORY OF HASTELLOY-N 15-mil-CLAD ELEMENTS

Element	Leakrate [cc(STP)/hr normalized to 1400°F, 2 atm H ₂]							Type of Element
	Initial	Post- Input "A"*	Post- Input "H"*	Post- 720-hr Endurance	Post- Input "H"*	Post- 500-hr Endurance	Post- 1000-hr Endurance	
8000	0.36	0.30	0.28	0.65	0.60	0.65	0.59	Standard S8DR
8001	0.19	0.67	0.37	0.21	0.35	0.31	0.27	
8002	0.40	0.32	0.40	0.52	0.71	Decanned		
8005	0.35	0.30	0.38	0.62	0.64	0.64	0.56	
8013	0.21	0.15	0.27	0.15	0.36	0.18	0.18	Double-clad, 304 SS
8016	0.38	0.12	0.29	0.22	0.39	0.16	0.16	
8021	0.32	0.31	0.44	0.27	0.41	0.35	0.34	
8006	0.17	0.18	0.26	0.15	0.34	0.15	0.13	4-mil 304 SS sleeve
8008	0.42	0.43	0.47	0.42	0.66	Decanned		
8011	0.21	0.18	0.23	0.19	0.20	0.10	0.09	2-mil Ni/Gd sleeve

*Inputs "A" and "H" represent selected thermal-cycle combinations described in previous reports.

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fabricated with both "cup-plug" and "butt-welded" final closures. Eight elements of each type have successfully passed S8DR acceptance tests and continue into the preproduction testing sequence.

Thermal tests on fuel coated with vapor-deposited Gd_2O_3 continue to show good adherence.

The NAA-120-4 and NAA-121-1 fuel-irradiation experiments continued in-pile operation. The test matrix for the NAA-140 fuel-irradiation experiment was established and design calculations were started on the experiment thermal model.

Pre-endurance thermal tests were successfully completed on the 46 normal S8DR performance test elements. These elements are now undergoing thermal-endurance testing.

B. FUEL ELEMENT MATERIALS EVALUATION

1. Cladding and Barrier

a. SCB Coating Hastelloy-N Performance

Ten 15-mil-clad developmental elements^{(1)*} successfully completed the S8DR preproduction test sequence. The permeation data are given in Table II-2. Eight of the elements have accrued 1000 of a planned 5000 hr of thermal endurance. The other two were decanned and examined for any anomalies after the initial test program. None were found. The stainless sleeve from Element 8008 appeared unchanged since the time of assembly. No evidence of a fuel-coating reaction was seen on the SCB coating. The leakrates of the remaining eight elements appear to be decreasing with time on the thermal endurance run.

A postmortem examination was made on the four double-clad endurance elements (rods encapsulated in 304 SS for fuel-coating reaction protection) which completed 10,000 hr of thermal endurance during the last report period. Visual examination of the elements before decanning revealed nothing abnormal. The dimensions of the stainless-clad rods were essentially unchanged. The stainless tubing was sectioned at midlength for removal of the fuel rods and the

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

four rods slipped out of the tubes. Three had a pinkish gold hue, the fourth (from Element 4971) did not. In general all four looked unusually clean; original machining marks were readily seen.

The halves of each stainless tube were then helium leakchecked for end-cap weld quality. Both welds from the stainless tube of Element 4971 leaked quite badly, but the welds on the other tubes were tight, which probably accounts for the difference in extent of oxidation (color) noted previously. Isochores were then run on the fuel rods. Near agreement between the isochore measurements and the computer values based upon the integrated leakrates was found for the percent hydrogen lost (Table II-3).

TABLE II-3
DOUBLE-CLAD FUEL ELEMENT HYDROGEN LOSS
AFTER 10,000 hr AT 1400°F

Element No.	Rod No.	Hydrogen Loss (wt %)	
		Integrated Permeation Rate	Isochore
4971	370-D2	6.6	4.3
4973	366-C4	5.7	5.7
4974	354-B3	6.4	7.8
4975	352-D1	5.3	6.0

The SCB coating on the Hastelloy-N tubes was a uniform dark bluish-green color typical of that obtained with hydrogen aging, and in general looked quite good. No iridescent films or deposits typical of the fuel barrier reaction were seen. The AI-8736D coating on the ID of the stainless tube from 4971 had changed considerably to a grayish-black amorphous appearing substance. Metallographically this stainless tube showed some severe intergranular attack on the ID, which was not seen on the three uncoated tubes. The causes of these reactions were not followed to any depth inasmuch as AI-8763D is not the reference coating material. In summary these elements performed quite well, with no cladding deficiencies noted, except for the AI-8763D-coated stainless tube.

b. Alternate Cladding

(1) Properties

The short-term biaxial stress-rupture testing of the as-received Incoloy-800 tubing was completed and is summarized in Table II-4 and Figure II-1. Samples are now being fabricated for the short-term tensile and long-term creep testing of as-coated material. The literature is also being reviewed for applicable data to supplement those being generated by this program.

(2) Process Development

The work aimed toward improving the processes for coating Incoloy hardware continued. Fabrication of the system for preoxidizing the alloy before coating was completed. A study was made on the effects of processing variables, such as oxidation temperatures, oxidation times, and rates of airflow on the adherence and coating quality. Firing-cycle times and temperatures for coated tubes were also examined. The method for coating tubes was established and detailed process flowsheets were issued. The range of parameters investigated and the selected values for oxidization and coating steps are shown in Table II-5. Adherences in excess of 23 flexes, before and after aging, along with good coatability have been obtained using the selected parameters. A comparison of the improved adherence over that on Hastelloy-N is shown on Figure II-2. A comparison of the respective substrates is also shown.

The process for coating cup-plugs is nearly established. Acceptable coating has been attained using essentially the same process which was established for tubes, with the exception of oxidation time which was reduced to 3-1/2 min. The range of variables and the tentatively selected parameters are shown in Table II-6.

As a result of these parametric studies and from x-ray diffraction data obtained on some substrates the following conclusions can be drawn:

- 1) Grit-blasting before oxidation improves both the uniformity and degree of adherence.
- 2) Pre-oxidation must be used to obtain acceptable adherences.

TABLE II-4
INCOLOY-800 BIAxIAL STRESS-RUPTURE TESTS
(Heat 810535)

Sample No.	Condition	Test Temperature (°F)	Hoop Stress (psi)	Diametral Strain-to-Failure (%)	Rupture Life (hr)	Average Creep rate (%/hr x 10 ⁻²)	Failure Location
800-1A	As received	1400	7950	19.5	195	10.0	1 in. from probe end
800-2A				20.6	195	10.6	
800-3A				7.88	195	4.04	
800-4A				10.3	163	6.32	
800-5A			6600	13.0	599	2.00	Main body
800-6A				17.7	599	2.95	
800-7A				20.0	462	4.30	
800-8A				14.7	629	2.30

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Figure II-1. Stress-Rupture Tests on Incoloy-800
As-Received Tubing (Heat 810535)

TABLE II-5
TUBE-COATING PROCESS VARIABLES
INCOLOY-800

Parameter	Range Investigated	Selected
Presandblasting	S8DR Cycle	S8DR Cycle
Oxidation Air Flowrate	0 to 830 cc/min	500 $\pm$ 25 cc/min
Oxidation Temperature	1850 to 1950°F	1900 $\pm$ 15°F
Oxidation Time	5 to 25 min	15 $\pm$ 1 min
Coating Cycle Time	3-1/2 to 5 min	4-1/2 $\pm$ 1/4 min
Coating Cycle Temperature	2100 to 2150°F	2150 $\pm$ 20°F

Figure II-2. Coating-Adherence Comparison of
Hastelloy-N and Incoloy-800
(Reduced 38%)

TABLE II-6
PLUG-COATING PROCESS VARIABLES
INCOLOY-800

Parameter	Range Investigated	Tentative Selection
Presandblasting	S8DR Type and S. S. White Garnet Grit Al_2O_3 -SiC Grit	S. S. White SiC Grit
Oxidation Time	3-1/2 to 15 min	3-1/2 $\pm$ 1/4 min
Oxidation Temperature	1900°F	1900 $\pm$ 15°F
Oxidation Air Flowrate	500 cc/min	500 $\pm$ 25 cc/min
Coating Temperature	2150°F	2150 $\pm$ 20°F
Coating Time	4-1/2 min	4-1/2 $\pm$ 1/4 min

- 3) Flowing air must be used to establish uniform oxide on both tubes and cup-plugs (and blind-end caps for blind-end closure elements).
- 4) Cr_2O_3 formation is critical for good coating adherence.
- 5) $\text{Cr}_2\text{O}_3/\text{Fe}_3\text{O}_4$ ratio should approximate 2 (based on x-ray diffraction data) to obtain good coatability.
- 6) Increasing the coating firing temperature slightly improves the solution of the oxide substrate into the coating, which improves the chemical bond and adherence.
- 7) The final coating quality is comparatively insensitive to the limits of the process parameters.

Two major accomplishments were achieved in the area of blind-end closure welding. One of these was a machining technique for facing off the ends of the coated tube and blind-end cap without chipping the glass. The success of this type closure depends upon maximizing the amount of coating behind the weld area which is subsequently fused together by the heat imparted in making the tube weld. Heretofore the coated parts were cut with either a lathe or Conosaw and then hand-polished to remove the chipped areas. A new lathe technique, using a 3000-rpm spindle speed and a 5° tool-rake angle, was developed which

provides a square face on both tube and coating with negligible chipping and no secondary operations required.

The second major accomplishment was in tooling and assembly techniques for making the butt-welded closure. A fixture was designed, built, and successfully used for positioning and joining the cladding tube and blind-end cap by spot-welding. This operation is done to insure proper alignment of components and to eliminate additional fixturing during the electron-beam closure weld. The necessary spot-welding parameters were determined using a miniature weld head with adjustable head pressure, and a copper electrode with a spherical tip. Electron-beam closure welds on approximately 20 consecutive Incoloy elements have now been successfully made using this combination spot-weld electron-beam technique. The statistical process capability of the procedure will be determined.

(3) Fuel-Element Testing

The five preprototype double-blind-end Incoloy-800 fuel elements (assembled December 1967) completed the S8DR-type preproduction test sequence and have been decanned. The permeation history of these elements is given in Table II-7. Two of the elements were zone-scanned before decanning. As can be seen from

TABLE II-7
INCOLOY-800 DOUBLE-BLIND-END FUEL ELEMENTS

Element No.	Leakrate [cc(STP)/hr normalized to 1400°F, 2 atm H ₂]					Permeation Scan Results % of Total Leakage		
	Post-Thermal Soak	Post-Input "A"	Post-Input "H"	Post-720-hr Thermal Endurance	Post-Input "H"	Closure End 1-1/4 in.	Center	Blind End 1-1/4 in.
800-13-2	0.12	0.11	0.39	0.49	0.37	43	26	31
800-8-6	0.82	1.04	1.80	2.54	2.20	-	-	-
800-14-17	0.85	0.58	1.32	1.37	1.40	-	-	-
800-15-5	0.88	0.70	1.05	1.72	1.76	-	-	-
800-5-1	1.71	1.05	2.46	3.65	3.03	11	74	15

TABLE II-8
INCOLOY-800 DEVELOPMENTAL ELEMENTS

Type	Element No.	Leakrate [cc(STP)/hr normalized to 1400°F, 2 atm H ₂]		
		Post-Thermal Soak	Post-Input "A"	Post-Input "H"
Incoloy-800 Double-Blind-End	9077	0.63	0.46	
	9079	0.37	0.43	
	9083	0.33	0.39	
	9091*	0.65	0.65	
	9092*	0.36	0.46	
	9093*	0.36	0.45	
	9094*	0.58	0.56	
	9095*	0.64	<u>0.96</u>	
	Average		0.55	
Incoloy-800 Standard Cup-Plug	9082	0.82	0.75	
	9084	0.23	0.38	
	9085	0.22	0.33	
	9086	0.71	0.72	
	9087*	0.38	0.58	
	9088*	0.51	0.55	
	9089*	0.35	0.35	
	9090*	0.31	<u>0.40</u>	
	Average		0.51	
Hastelloy-N Double-Blind-End	6042	0.35	0.38	0.35
	6062	0.75	0.75	0.84
	6063	0.48	0.60	0.71
	6081	<u>0.77</u>	<u>0.88</u>	0.80
	Average	0.59	0.65	

*Coated without the usual samarium.

the data the major source of leakage from Element 800-5-1 (the one with the greatest leakage) was from the tube proper. The distribution of leakrate for 800-13-2 (lowest leakage) was found to be comparable to that normally found in Hastelloy-N-clad elements.

Upon decanning, copious amounts of a thin film were found in Elements 800-5-1, 800-8-7, and 800-13-2. As can be noted, the highest and lowest leaking elements are among these three. Destructive examination of the cladding tubes revealed very poor coating quality based upon current standards. (These tubes were the product of very early coating process development.) It was obvious that the first coats of ceramic applied to these tubes was grossly pitted. In some cases the second coats covered the pits but did not appear to do so in others. A significant amount of "orange peel" was also noted. Glass coverage of the blind-end-type closure welds looked quite good. Though the performance of four of the five elements was poor (attributed to coating quality), Element 800-13-2 effectively demonstrated the feasibility of replacing the cup-plug with the blind butt-welded end closure.

The initial ten fuel elements (five cup-plug and five double-blind-end), fabricated with the current improved processes and methods, exhibited excessively high permeation rates during acceptance testing despite the fact that the fabrication appeared to have gone quite smoothly. Subsequent testing, post-mortem examinations, and process reviews established that during lathe machining of the coated tubes the cladding was being chucked too tightly, resulting in microcracking of the glass coating. The chucking procedure was modified and a teflon insert incorporated to provide a softer and more uniform grip during this operation.

Sixteen additional elements were subsequently fabricated, eight cup-plug and eight double-blind-end, and are currently being subjected to the preproduction test sequence. For additional comparison four double-blind-end, 15-mil-wall Hastelloy-N elements were fabricated and are being tested with the Incoloy-800 elements. The permeation data obtained thus far are summarized in Table II-8. To date, no real difference between the Incoloy-800 cup-plug and double-blind-end elements can be seen. A 100% fabrication yield was obtained with each group.

Although the glass sealing of the Hastelloy-N double-blind-end elements appears successful, the visual quality of the welds was comparatively poor. Parametric studies of this weld are being made to determine how to improve it, but the chromized surfaces offer a significant obstacle.

2. Fuel

a. Phase Studies

The data on the $\alpha + \delta$ -to- δ phase boundary and δ -to- γ' transformation kinetics in the Zr-U-H system were assembled and correlated in a topical report draft. This report will be issued during the next quarter.

b. Burnable Poison Studies

Control requirements of ZrH reactors can be reduced by the addition of a burnable poison to the fuel elements. The poison provides partial control by burning out at a rate that compensates for some of the reactivity losses. The objective of this task is to develop a process technique for the application of from 0.05 to 0.5 gm of gadolinium to each fuel rod; the technique should permit a controlled variation of poison content over the length of the element. Studies are being made to determine the feasibility of two methods of incorporating the poison into the fuel elements; the methods are (1) application of the poison to the surface of the fuel, and (2) its inclusion in an inner metallic sleeve.

One method being investigated for applying the poison to fuel is the vapor deposition of gadolinium oxide, Gd_2O_3 , by electron-beam heating. Eight specimens of fuel 0.52-in. diam x 4.25-in. long were coated with Gd_2O_3 by this method using a substrate temperature of $\sim 550^\circ F$ and a Pierce-type electron-beam gun. Four were coated on one side to a thickness of 0.2 to 0.3 mil, corresponding to that which would be obtained from ~ 0.7 gm of Gd uniformly distributed over a full-length rod. The other four were coated in four steps with 90° manual rotation of the specimens between depositions; the thicknesses obtained in these depositions ranged from 0.05 to 0.3 mil. The primary objective of this phase of the study is to obtain a preliminary evaluation of the vapor-deposition method based on short-term thermal testing of these fuel specimens. Secondary objectives include the preliminary design of coating equipment and start of a survey of nondestructive methods of determining the quality and distribution of vapor-deposited Gd_2O_3 coatings.

The testing and examination of two of the four fully coated specimens were discussed in the previous quarterly report. The remaining two specimens were thermal-soaked at 1400°F for 1000 and 1500 hr, respectively. This thermal treatment was subsequent to four thermal-shock tests. The specimens were visually examined through the encapsulating quartz at 500-hr intervals. Visual examination after 500 hr showed that the coating was still intact. It has reverted to its original light gray color and the small uncoated areas at the ends of the specimens, which originally indicated that some oxidation occurred during coating, were very bright and oxide-free. This observation suggests that the $\text{Gd}_2\text{O}_{3-x}$ coating had combined with oxygen dissolved in the fuel to re-form the stoichiometric oxide, Gd_2O_3 . Subsequent visual examinations indicated no spallation of the coating from the fuel and no free Gd_2O_3 in the capsules. Upon removal from the quartz tubes, the adherence of the coating on the rods was good although some could be removed by vigorous rubbing. The appearance of the specimens was unchanged from the 500-hr examination but x-ray diffraction of the coating showed that it had transformed to the low-temperature cubic modification. Metallographic examination showed that adhesion of the coating to the fuel was excellent.

The four partially coated specimens were assembled into sublength S8DR elements and subjected to S8DR acceptance testing followed by thermal aging at 1400°F. The initial hydrogen-permeation rates were satisfactory and ranged from 0.09 to 0.31 cc(STP)/hr. Two of these specimens have been removed from the cladding after 1080 and 1512 hr at 1400°F respectively. In both cases the coating was adherent but had changed from the original light gray color to nearly black. This observation, in conjunction with the results obtained from the fully coated specimens, suggests that the coating is partially reduced by the relatively high pressure of hydrogen. X-ray fluorescence and metallographic examination of the two coated tubes failed to reveal any effect on either the barrier or the gadolinium oxide coating. X-ray diffraction of the poison coatings on the specimens again showed them to be cubic. Metallographic examination again showed good adhesion of the coating to the fuel. Electron-microprobe examination of the 1080-hr specimen showed no detectable interdiffusion of the coating and fuel.

On the basis of these short-term thermal tests the adherence of vapor-deposited Gd_2O_3 coating appears to be satisfactory and essentially independent of the surface roughness of the fuel. It is not adversely affected by the initial thermal shock or thermal soaking at temperatures to 1400°F in 1500 hr. The transformation of the coating from the monoclinic to the low-temperature cubic form in less than 1000 hr at 1400°F is contrary to the expectation, reported earlier, that no detectable amount of cubic Gd_2O_3 should be formed in any conceivable reactor lifetime. However, this transformation does not affect adherence of the coating, and may be beneficial since the coefficient of expansion of the cubic form, $10.5 \times 10^{-6}/^\circ\text{C}$, is within the range reported in earlier work on ZrH fuel, $(9.5 \text{ to } 11.5) \times 10^{-6}/^\circ\text{C}$. The apparent partial reduction to a substoichiometric sesquioxide is not considered significant with respect to reactor operation but may be indicative of diffusion effects which have not been otherwise observable in these tests.

There are two basic types of electron-beam application methods, the Pierce and "work-accelerated" methods. The Pierce method was used in the coating of the specimens described above. The work-accelerated method will be used to coat specimens for further investigations. Tentative fuel configurations have been prepared for both methods so that equipment specifications can be prepared as soon as possible. The equipment specifications will be based on the experience obtained using both methods.

A survey has been started of nondestructive methods of determining the distribution and quality of vapor-deposited Gd_2O_3 coatings. Three methods appear to be particularly promising: X-ray fluorescence, alpha-particle absorption, and beta back-scatter. A single preliminary experiment was performed to determine whether the fluorescence method should be further investigated.

C. IRRADIATION TESTS

1. NAA-120-4

The NAA-120-4 irradiation experiment is statistically designed to evaluate SNAP fuel performance as a function of temperature and fuel composition at 0.9 met. at. % burnup. The experiment continued in-pile operation during the report period. Capsules M and V-1 are being operated using cladding

thermocouples for control due to in-fuel thermocouple anomalies, discussed in the previous report. In addition the pressure line on V-1 has apparently plugged. The pressure varies little, regardless of temperature changes, remaining relatively constant at ~ 2.2 psi. Because of these problems the phase-change experiment will not be performed on Specimen V-1. The equipment needed to perform the phase-change experiment on Specimen V-2 is currently being installed. This includes NaK trap, charcoal filter, calibrated volume, and vacuum-system plumbing. The V-2 phase-change experiment will be carried out during the next report period with exact time being dependent upon reactor operating schedule.

A detailed examination of the pure-helium startup data taken during the experiment leads to the conclusion that an appreciable amount of fission gas has been released in the capsules, based on the following:

- 1) At low fuel temperatures, the fuel-to-cladding temperature drops have been decreasing with time. This is to be expected as the initial vacuum between fuel and cladding is replaced with fission gas.
- 2) At higher fuel temperatures, the fuel-to-cladding temperature drops have been increasing with time. This also is to be expected as the hydrogen in the gas annulus is diluted with fission gas.

The major source of fission gas is believed to be the 20 wt % fuel sleeves on the ends of the support rods, which have twice the burnup of the test specimens.

2. NAA-121-1

The primary objective of the NAA-121-1 experiment is to evaluate the performance of three S8DR-type fuel elements under approximated S8DR reactor operating conditions. A fourth element with low hydrogen content will be used to evaluate the effect of δ -to- β phase transformation after significant fission-product accumulation. An out-of-pile test is being made to differentiate between thermal and radiation effects on fuel-element behavior. This test consists of subjecting four fuel elements to thermal treatments which simulate as nearly as possible the thermal treatment of the in-pile elements.

TABLE II-9
NAA-140-1 IRRADIATION EXPERIMENT TEST MATRIX

Subcapsule No.	Specimen No.	Cladding		Gap		Length		Poison		End-of-Life	Temperature		
		Hastelloy-N	Incoloy-800	Large	Small	Full	Sub	Foil	Surface		S8DR	Linear Gradient	Isothermal Peak
1	1		X		X		X		X				X
	2	X		X			X		X			X	
	3		X	X			X	X				X	
2	4	X			X		X	X					X
	5	X		X			X	X				X	
	6		X	X			X	X				X	
3	7	X		X		X		X		X	X		
4	8	X		X		X		X			X		
5	9		X	X		X		X		X	X		
6	10		X		X		X	X					X
	11	X		X			X	X				X	
	12		X	X			X		X			X	

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During the quarter, irradiation and the out-of-pile tests continued. Four thermocouples have now failed in the irradiation experiment, three on Capsule No. 4 and one on No. 2. Loss of these thermocouples does not jeopardize experiment operation.

The four out-of-pile test elements have accumulated 3200 hr of operation with about 2500 at design temperature. The hydrogen-permeation rates measured from out-of-pile Element No. 5185 indicate that the 5% maximum allowable hydrogen loss (to prevent β -phase transformation) will be reached in about 5000 hr at reactor operating temperatures. Two other out-of-pile elements (4468 and 5187) will accrue about 7500 hr before the 5% value is reached.

3. NAA-140-1

The primary objective of the NAA-140-1 irradiation experiment is to obtain irradiation performance data on parameters representing future long-life fuel elements. This experiment will be designed to operate for 20,000 hr at design power and operating temperatures representing a 600-kwt (211-element core) SNAP reactor with 1300°F NaK-outlet temperature.

Preliminary layout drawings and the preliminary heat-transfer analysis for the experiment have been completed. The present test specimen matrix for the NAA-140-1 experiment is given in Table II-9. The matrix and preliminary-design layouts for the subcapsules containing the sublength fuel elements indicate one isothermal and two gradient-profile elements per subcapsule. The first element in each subcapsule is to have an isothermal profile and is to be instrumented with a pressure line. This arrangement allows the required pressure lines to be brought out of the elements without excessive hydrogen loss. The ability to detect barrier failure will be dependent on the leakrate, pressure, and instrumentation sensitivity. Calculations to determine the anticipated range and variations in pressure to be expected during the test are being made so that a good evaluation of the problem can be made. The preliminary layouts were approved and will serve as a starting point for detailed design.

The preliminary heat-transfer analysis was directed toward developing a heat-transfer model for the full-length S8DR-type fuel element, using operating data from this experiment, as the fuel elements are the S8DR type. The resulting

cladding thermocouple temperature profile defined by the calculations was compared to the actual profile of the NAA-121-1. Figure II-3 gives the results of the thermal analysis and the in-pile data for the NAA-121-1 600-kw subcapsule at 0.85Q design power on pure helium and 1.00Q at operating temperatures. The two are very close to agreement and appear to verify the thermal model. To add further credence to this model similar calculations are being made on the NAA-121-1 660-kw subcapsule.

Figure II-3. Cladding Thermocouple Temperature Profile

D. FUEL ELEMENT TESTING

1. Preproduction Element Performance Tests

The 12 S8DR preproduction fuel elements continued in uninterrupted endurance test at 1400°F during this report period. As previously reported,⁽¹⁾ these elements were removed from the endurance test after 5000 hr and subjected to five thermal-ramp heating cycles to 1400°F (thermal input "A"). The elements have now completed more than 2100 hr of the second 5000-hr endurance cycle.

The time plot of Figure II-4 illustrates the outstanding hydrogen-retention performance of these elements. The average hydrogen leak (normalized to 2 atm H_2 and 1400°F) and hydrogen-barrier performance parameter (K) have remained essentially constant during the test period to date. The total H_2 loss after 8000 hr at 1400°F is about 3.7%. This corresponds to $\sim 1.3\%$ at core-temperature profiles and is far less than the 5% allowable.

Figure II-4. S8DR Preproduction Fuel Element Hydrogen Permeation

2. S8DR Element Performance Tests

The non-nuclear performance testing of S8DR normal fuel elements continued during this report period in accordance with procedures previously described.⁽²⁾ Twelve Phase-I fuel elements, representative of start-of-core production fuel elements, have completed 4000 hr of endurance testing at 1400°F; twelve Phase-II fuel elements, midcore production, have completed more than 3300 hr of endurance testing; and 12 Phase-III fuel elements, end-of-core production, have completed the first 2000-hr endurance test cycle. The pre-endurance test data on Phase-III elements are shown in Table II-10. Figure II-5 compares the hydrogen-permeation rate of elements in the three phases with the "Special 7" and S8DR preproduction fuel elements.

The hydrogen-retention performance of these S8DR production fuel elements is, in general, comparable to that of the preproduction elements. The performance parameters (i.e., permeation rate, barrier parameter, hydrogen loss) of Phase-I elements are slightly higher than for preproduction elements. However, the Phase-II and Phase-III elements equal or exceed the preproduction performance.

TABLE II-10
 PHASE-III S8DR PERFORMANCE TEST DATA HYDROGEN
 PERMEATION [cc(STP)/hr Normalized to 1400°F, 2 atm]

Element No.	Pre-Acceptance	Final Acceptance	Post-Acceptance	Post-Input "H"
170	0.41	0.39	0.40	0.42
238	0.40	0.63	0.48	0.59
309	0.40	0.41	0.49	0.72
348	0.36	0.40	0.33	0.34
357	0.15	0.32	0.28	0.38
359	0.20	0.32	0.32	0.33
360	0.11	0.23	0.40	0.25
361	0.61	0.32	0.27	0.44
399	0.13	0.26	0.40	0.31
504	0.56	0.60	0.63	0.80
521	0.29	0.41	0.49	0.62
581	0.45	0.50	0.77	0.38
Average	0.34	0.40	0.44	0.47

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Figure II-5. S8DR Production Fuel Element Hydrogen Permeation Compared with "Special 7" and Preproduction Fuel Elements

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The ten Phase-V fuel elements are being endurance-tested at 1300°F to evaluate the effect of temperature on long-term performance. These elements have now completed 2800 hr of endurance testing. The leakrate through the first 2000-hr endurance cycle is plotted in Figure II-6. The hydrogen-permeation rate shown is normalized to 1400°F at 2-atm pressure for direct comparison with the other S8DR tests.

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Figure II-6. S8DR Fuel Element Hydrogen Permeation for Ten Phase-V Elements at 1300°F

The vibration, shock, and acceleration testing of the Phase-IV fuel elements was concluded. The three fuel elements initially subjected to all of the specified inputs* were all damaged as indicated by higher hydrogen-permeation rates and confirmed by x-ray and visual inspection after decanning. Eight additional fuel elements were subjected to individual mechanical inputs to identify the input(s) causing the damage. Pairs of fuel elements were subjected to one each of the following four mechanical inputs: lateral and longitudinal sinusoidal and random vibration. The eight elements were permeation tested, zone-scanned and x-rayed before and after the mechanical inputs. Table II-11 is a summary of all test results. The two elements subjected to the longitudinal sinusoidal vibration input, No. 85 and 86, showed a large increase in hydrogen-permeation rate, indicating damage to the hydrogen barrier. The closure ends of both rods were badly chipped as seen in x-ray and subsequent visual inspection. These data indicate that the longitudinal sinusoidal vibration input is a major contributor to the damage of fuel elements resulting from the total vibration, shock, and acceleration input. No further testing on the Phase-IV fuel elements is planned.

*See Table II-8 in Reference 2

TABLE II-11
PHASE-IV S8DR PERFORMANCE DATA SUMMARY

Element No.	Mechanical Input Received	Leakrate [cc(STP)/hr normalized to 1400°F, 2 atm H ₂]				Damage Noted in X-ray	1400°F Zone Scan [cc(STP)/hr]				Rod Weight Chipped Off (grams)	
		Pre-Input	Post-Input	Increase				Blind	Center	Closure		
				Caused by Input	Average							
70	Total vibration, shock, and acceleration input per Table 2 DTP-S8DR-28-001	0.39	1.42	1.03	1.08	Yes	Pre	Not Tested			Not Applicable	
							Post	0.84	0	0.54		
78			0.52	1.30		0.78	Yes	Pre	Not Tested			Not Applicable
								Post	0.85	0.17	0.38	
81		0.54	1.96	1.42		Yes	Pre	Not Tested			1.7	
							Post	0.68	0.42	1.29		
82	Lateral sinusoidal vibration, 2 sweeps	0.61	0.57	-0.04	0.02	No	Pre	0.15	0.23	0.25	†	
							Post	0.24	0.11	0.29		
84		0.42	0.50	0.08		No	Pre	0.11	0.25	0.14	†	
							Post	0.20	0.12	0.21		
85	Longitudinal sinusoidal vibration, 1 sweep	0.39	1.18	0.79	0.92	Yes	Pre	0.12	0.18	0.13	2.1	
							Post	0.52	0.14	0.68		
86		0.46	1.51*	1.05		Yes	Pre	0.17	0.25	0.26	2.5	
							Post	0.79	0.14	0.84		
87	Lateral random vibration, 10 min	0.35	0.77	0.42	0.30	No	Pre	0.14	0.09	0.20	†	
							Post	0.28	0.09	0.32		
351		0.33	0.51	0.18		No	Pre	0.10	0.06	0.08	†	
							Post	0.11	0.05	0.11		
90	Longitudinal random vibration, 5 min	0.54	0.59	0.05	0.06	No	Pre	0.15	0.11	0.23	†	
							Post	0.23	0.10	0.28		
93		0.70	0.76	0.06		No	Pre	0.20	0.15	0.16	†	
							Post	0.26	0.14	0.21		

*Initial post-input permeation check was 0.64; recheck after zone scan was 1.51.

†Not decanned

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III. COMPONENT DEVELOPMENT

A. INTRODUCTION AND SUMMARY

Both design verification testing of the S8DR components and technology development for the reliable high-performance components for the reference zirconium-hydride reactor are continuing. These components include the bearing, actuator, 4π shieldable reflector, shield, and NaK pump.

Design verification testing of the three S8DR actuators and four sets of reflector bearings has passed the quarter mark and is approaching the halfway point of the scheduled 12,000-hr tests. The post-test examination of the third and the last S8DR scram mechanisms was completed and indicated no problem areas. Test of the position sensors and limit switches is nearing completion.

The evaluation testing of nickel-clad silver and BeO-hardened copper conductor materials has satisfactorily completed over 4300 hr at 1200°F after 400 thermal cycles to 1200°F. After 6000 hr at 1200°F the magnetic properties of the Armco iron compared favorably with that of Hyperco-27. The former material has mechanical properties more suitable for the projected launch environment.

The preliminary phase of the performance test, consisting of 27 thermal cycles and 1050 hr at 1500°F in 10^{-8} torr, was completed on the high-temperature test bearing with acceptable friction torques. Only slight wear resulted on the surfaces of the reference high-temperature bearing materials (aluminum oxide and carbon-graphite) after 900 in. of travel at 1600°F and 10^{-8} torr.

Based upon previous studies a reference design of the 4π shieldable reflector drum was prepared. Samples of rare earth oxide (60 wt %) and nickel cermet with a continuous nickel matrix were successfully fabricated by a hot-pressing process. This represents significant progress toward the fabrication of the europium-oxide-nickel cermet absorber for the control drum.

The ZrH particle-density studies for the combined neutron-gamma shield material have shown that ZrH density can be varied up to 78.3% of theoretical. Massive loss of lithium hydride from the 1000°F ZrH-LiH aggregate sample had taken place on both hydrogen-relocation tests after 4000 hr. The temperature differential between the hot and cold samples of the ZrH-LiH aggregate

accelerates the LiH loss. Literature studies for potential new shield materials did not locate a practical alternate for the lithium-hydride neutron attenuator and only lithium barium-hydride was considered a possible contender as a combined neutron-gamma shield material.

The dc conduction NaK pump design was re-evaluated on the basis of the reduced pressure requirement of the primary loop. The resultant two-pass pump system represents considerable weight reduction for the pump and the total system.

B. CONTROL-DRUM ACTUATOR

1. S8DR

The 12,000-hr design verification testing of the three S8-modified S8DR actuators⁽²⁾ continues. Table III-1 gives the current status and Table III-2 the data to date.

All units are operating satisfactorily. No unusual operational conditions were observed.

2. Actuator Technology Development

a. Coil Reliability

The tests continued on the three candidate materials — nickel-clad silver, Cube alloy, and Oxalloy.⁽³⁾

The first test run, designed to produce failure in the conductor materials by thermal cycling to 1300°F, was converted into a steady-temperature 1400°F test after more than 400 thermal cycles did not produce a failure. The failure of one Ni-clad-Ag conductor coil after 1084 hr of continuous testing was described previously.⁽¹⁾

An investigation was made to determine the maximum temperature at the fracture points. A coil similar to the failed coil, fabricated with thermocouple junctions at the conductor failure points, was inserted into the test setup in place of the failed coil. With the control thermocouple undisturbed, other thermocouples were arranged around the test specimens. Test runs were made at normal, slow, and the maximum heatup rates to determine their effect on temperature distribution of the coils under test. The results of this test, made

TABLE III-1
STATUS OF S8DR ACTUATOR DESIGN VERIFICATION TESTING

SN No.	Phases Completed			Phase in Progress (Hours of 3000-hr Endurance at 1125°F Completed)
	Initial 55 Thermal Cycles	1000-hr Endurance at 1125°F	1000-hr Dwell at 1125°F	
011	x	x	x	2724
013	x	x	x	2544
010	x	x	x	2160

TABLE III-2
ACTUATOR PERFORMANCE AT 1125°F

	SN No.	Brake Current (amp)		Stepping Current* (amp)		Insulation Resistance Ω	
		Pull-in, Minimum	Release, Maximum	CW Minimum	CCW Minimum	Brake	Minimum Phase
Start of test after 1000-hr dwell	001	1.55	0.35	2.9	2.9	7 M	300 k
Latest data		1.80					
		1.66	0.30	3.0	2.6	1.6 M	1.4 M
Start of test after 1000-hr dwell	013	1.44	0.68	3.3	2.8	7.5 M	5 M
Latest data		1.56					
		1.62	0.62	3.0	2.6	0.95 M	4.5 M
Start of test after 1000-hr dwell	010	1.28	0.34	3.0	3.2	10 M	200 k
Latest data		1.38					
		1.46	0.38	3.3	2.8	1.3 M	1.2 M

*To step 40 oz-in. load

to determine if unexpected transient high local temperatures, caused localized melting of the silver core, are shown in Table III-3.

There was no evidence of overshoot or temperature excursions during the test run. In all testing the coil-lead temperature was among the lower-to-average temperature readings. Based upon this information melting of the core was discarded as a mode of failure.

A second test, converted from the 1200°F thermal-cycle test after 407 thermal cycles and 339 hr at temperature, has been run at a constant temperature of

TABLE III-3
COIL HEATING TEST

Rate of Rise (°F/min)	Maximum Temperature (°F)	Temperature Spread (°F)
Slow (≈4)	80 (at 1400°F)	80
Normal (≈11)	90 (at 500°F)	80
Fast (≈40)	150 (at 400 to 800°F)	70

1200°F for 4311 hr. The remaining four Cube-alloy coils and the five Ni-clad-Ag coils have shown no significant change in resistance. Testing is continuing.

The failed Cube-alloy coil reported previously⁽¹⁾ was disassembled to determine the cause of the reduced resistance. The investigation was not conclusive because, although the short was determined to be between the coil body and the start lead, the exact point of shorting could not be located. This test, being a material test, had a compromised conductor insulation without a separate insulated lead wire. The coarse woven insulation present on both the start lead and the coil wire had apparently separated sufficiently to allow a contact to develop.

A third test is being made with coils wound with Oxalloy conductors on rectangular-shaped alumina bobbins which are mounted in a S8DR actuator housing. The purpose of this test is to obtain long-term (12,000-hr high-fatigue-loading (500 thermal cycles) testing of Oxalloy wire at 1125°F. These coils were previously subjected to 200 thermal cycles and 350 hr at 1125°F. In the current test the coils have accumulated an additional 187 thermal cycles and 6195 hr at temperature. Operation to date has been satisfactory with no indication of conductor failure or insulation degradation.

b. Magnetic Materials

The previously reported⁽³⁾ testing of ring samples of Hyperco-27 (a 27% cobalt material) and Armco iron (a pure iron) continues with over 6000 hr at

1200°F. The magnetic properties of both samples were measured after 6000 hr and are in essential agreement with the data reported after the 1042 and 3136 hr of test. These measurements will be repeated after 9000 and 12,000 hr at 1200°F.

Two Hyperco-27 stator shells were tested at ambient temperature, 800, 1000, and 1200°F, to determine the effect of long time at high temperature on magnetic properties. One shell, SN007, had prior service at 1025°F for 13,100 hr, while the other, SN005, had minimum service of 500 hr at 1025°F and above. Figure III-1 shows the test setup and Figure III-2 gives performances curves of the two stators at ambient and at 800, 1000, and 1200°F. Stator SN005 has better performance (magnetic flux) at the higher values of magnetizing force than SN007. The crossing of the magnetic curves is typical of variations between material heats. There is some decrease, with temperature, of performance of the two stators. Because the stators were completely machined with all the necessary holes and reduced sections, the difference in performance between the two stators is probably due to variations in machining and material heats rather than a change due to time at temperature.

Figure III-1.
Test Setup, Stator
Magnetic Test

C. CONTROL-DRUM BEARINGS

1. S8DR Design Verification Tests

Design verification testing of the S8DR control-drum bearings started last quarter continues toward the 12,000-hr goal. Four sets of Al_2O_3 (coated on Inconel-750 shafts) against solid P5 carbon-graphite reference design bearings are under test. Three sets are at ground-test pressure of 1×10^{-5} torr (two at 1150 and one at 1250°F) and one set is at simulated space pressure of 1×10^{-8} torr

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Figure III-2. Magnetization Curves for Stators 5 and 7

Figure III-3. S8DR Control-Drum Bearing Design Verification Tests

(1150°F). The 10^{-5} -torr tests have accumulated over 3000 hr at vacuum and the 10^{-8} -torr over 2000. All tests are in the operational test phase with friction-torque measurements being made at 100-hr intervals. Each test completed 100 hr at temperature (1150 or 1250°F) and 30 thermal cycles before the operational test phase. Test 1, 2, and 4 at S8DR design temperature of 1150°F also underwent a short-term over-design-point test (500 hr at 1250°F), and Test 3 operates throughout at 1250°F, 100 above design point. Test procedures specify a visual inspection of bearing components after 3000 hr at test, but inspection has been postponed until the friction-torque values have stabilized. Figure III-3 shows the friction-torque history of each test and illustrates the increasing friction torque of Tests 1, 2, and 3 during operational testing. Table III-4 summarizes operational test phase data shown in the figure.

TABLE III-4
S8DR CONTROL-DRUM BEARING FRICTION-TORQUE SUMMARY

Event	Test No.			
	1	2	3	4
Maximum friction torque (in.-lb)				
at 1150°F	2.9	3.75	-	4.5
at 1250°F	1.7	2.5	3.3	4.6
Equivalent drum cycles (270°F)	65	73	80	43
Total time, hr:				
at 1150°F	3104	2995	-	1830
at 1250°F	623	623	3710	672
at vacuum	4100	4080	4140	2780
Vacuum, torr	10^{-5}	10^{-5}	10^{-5}	10^{-8}

2. Advanced ZrH Bearings

The purpose of the advanced ZrH bearing program is to upgrade the bearing technology from the 1150°F environment of SNAP 8 to the 1500°F 20,000-hr advanced reactor environment. A prototype bearing set using Al_2O_3 (LA-2) on Ta-10W vs P5 carbon-graphite has completed a thermal-cycle test, and a series of material evaluation sliding-friction tests are in progress.

a. UHV Bearing Thermal-Cycle Test

The Al_2O_3 (LA-2) on Ta-10W vs P5 carbon-graphite ultrahigh vacuum (UHV) bearing thermal-cycle test has been completed. A summary of the final data is shown in Table III-5. Figure III-4 shows the friction-torque and corresponding friction-coefficient values for the entire test. Data scatter at ambient temperature is rather large, but 1500°F data appear stable.

TABLE III-5
ULTRAHIGH VACUUM BEARING TEST SUMMARY

Time at vacuum (10^{-8} torr), hr		2300
Time at 1500°F, hr		1050
Thermal cycles to 1500°F		27
Number of bearing cycles (110°)		180

Maximum Friction-Coefficients/Friction-Torques			
Pressure (torr)	Temperature (°F)	Maximum Friction-Torque (in.-lb)	Corresponding Friction-Coefficient
ambient	70	23.7	0.74
10^{-9}	70	24.0	0.74
10^{-9}	800	15.0	0.46
10^{-9}	1500	9.5	0.29

Following the thermal cycle test the bearings were disassembled and inspected. No sign of Al_2O_3 deterioration was found. Figure III-5 shows the post-test bearings. A slight but adequate carbon-graphite transfer film has developed on the shaft, socket, and thrust-sleeve Al_2O_3 surfaces. The thrust face of the ball shows evidence of considerable rotation, while the ball OD shows little sign of rotation. All surfaces can be judged to be in exceptionally good condition.

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Figure III-4. Advanced ZrH Reactor Bearing UHV Thermal-Cycle Test Data

Figure III-5. Advanced ZrH Reactor
Bearing After Test

b. Sliding-Friction Studies

Wear tests on four bearing couples - Al_2O_3 (LA-2) on Ta-10W against AXF-5Q and P5 carbon-graphite, and solid TiC (K-162B) against AXF-5Q and P5 - have been completed. Bearing couples were subjected to 900 oscillation cycles, equivalent to a bearing-life travel of 400 in. at 1600°F and 1×10^{-8} torr, while under a 150-psi load. Table III-6 shows friction coefficients for the four tests. The post-test condition of the bearing couples is shown in Figures III-6 and -7.

TABLE III-6
SLIDING-FRICTION WEAR TEST SUMMARY

Bearing Couple	Travel at 1600°F (in.)	Time at 1600°F (hr)	Maximum Friction Coefficient		
			Ambient	800°F	1600°F
Al_2O_3 vs P5	~900	5	0.56	0.35	0.17
vs AXF-5Q	~900	5	0.56	0.43	0.15
K162B vs P5	~900	3	0.61	0.61	0.42
vs AXF-5Q	~900	3	0.65	0.52	0.70

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a. AXF-5Q CARBON-GRAPHITE vs Al_2O_3 (LA-2) ON Ta - 10W

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b. P5 CARBON-GRAPHITE vs Al_2O_3 (LA-2) ON Ta - 10W

7696-4036

Figure III-6. Sliding-Friction Wear Test Bearing Couples
(Carbon-Graphite vs Aluminum Oxide)

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a. AXF-5Q CARBON-GRAPHITE vs TiC (K-162B)

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b. P5 CARBON-GRAPHITE vs TiC (K-162B)

7696-4027

Figure III-7. Sliding-Friction Wear Test Bearing Couples
(Carbon-Graphite vs Titanium Carbide)

All base specimens have fully developed carbon-graphite transfer films providing a good bearing surface. Wear of carbon-graphites was minimal in all four cases. Chipping of the Al_2O_3 coatings (Figure III-6) occurred in removing the specimens from the test chamber following the wear test.

A long-term sliding-friction test is now being made on the two Al_2O_3 bearing couples (Al_2O_3 vs AXF-5Q and Al_2O_3 vs P5) at 1×10^{-8} torr and 1600°F . A seven-step test under way is as follows:

- 1) Five bearing-oscillation cycles at 10^{-8} torr and 70°F .
- 2) Five cycles at 10^{-8} torr and 800°F .
- 3) Five cycles at 10^{-8} torr and 1600°F daily for two weeks.
- 4) Five cycles at 10^{-8} torr and 70°F .
- 5) Five cycles at 10^{-8} and 1600°F followed by a two-week dwell period.
- 6) Five cycles at 10^{-8} torr and 1600°F .
- 7) Repeat of Steps 1 through 4.

Approximately 400 hr at 1600°F have elapsed with the test currently in the dwell phase. Table III-7 summarizes test data.

TABLE III-7
SLIDING-FRICTION OPERATIONAL TEST DATA

Test Event	Friction Coefficients	
	Al_2O_3 vs P5	Al_2O_3 vs AXF-5Q
At ambient (initially)	0.52*	0.67
At ambient (after 400 hr at 1600°F)	0.60*	0.59
At 800°F	0.32	0.41
At 1600°F (initially)	0.25	0.32
(after 400 hr at 1600°F)	0.22	0.22
Total time at 1600°F , hr	440	440

*8.0-lb load; normal = 16.0 lb

D. INSTRUMENTATION

1. Position Sensors

Testing was begun on four developmental position sensors of the modified design. Two completed their testing last quarter, having been thermally cycled to 800°F 200 times, and subsequently operated for 5000 hr at 800°F and 10^{-6} torr. The second two units have completed 10,868 hr and 55 thermal cycles of a combined 55 thermal-cycle 12,000-hr endurance test to 800°F. Operation continues satisfactorily. The minimum insulation resistance for both units is above 60 MΩ. All units, including SN 006 which has now accumulated 21,537 hr, have insulation resistance values well above the 100Ω minimum acceptable for S8DR operation. The contact resistances have remained below 3Ω. The maximum acceptable contact resistance for S8DR is 25Ω.

2. Limit Switches

Long-term (12,000-hr) testing is in progress on nine switches selected from previously run short-term static and dynamic tests. The new test has now accumulated 6019 hr of test time, 4942 switch-operation cycles, and 37 thermal cycles from room temperature to the 1000°F test temperature. The latest test data and a description of the test items are listed in Table III-8. Figure III-8 shows the insulation resistances of each switch throughout its total test history. During the last 1800 hr switches SN 003, 005, 068, 069, 065, and 066, all with the same test history, underwent an insulation-resistance reduction of approximately one decade. SN 032 and 001 with several thousand more total test hours went down in insulation resistance a factor of approximately two. The previously observed significant superiority of the alumina insert units over the other units is no longer apparent although some slight edge still seems to exist. The vented units clearly do not now show any improved performance.

E. S8DR SCRAM MECHANISM

The completion of the third scram mechanism test at 100°F above the expected S8DR operational temperatures was reported in the previous progress report. Evaluation of the test data and post-test examination of the detail parts have been completed.

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Figure III-8. Switch Insulation Resistance vs Time

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TABLE III-8
LIMIT SWITCH TEST DATA SUMMARY

SN	Description	Applied Voltage (vdc)	Test Time (hr)		Resistance (k Ω)*	
			Previous	Total	Contact	Insulation
001	Production unit	1	5,518	11,537	0.421	3.5
003	Alumina insert	24	2,771	8,790	0.432	56
005	Alumina insert	24	2,771	8,790	0.526	56
006	Development unit	24	15,518	21,537	0.481	1.3
032	Production unit	1	5,518	11,537	0.385	5.9
065	Vent-holes, drilled	1	2,771	8,790	1.323	13
066	Vent-holes, drilled	1	2,771	8,790	0.396	14
068	Production unit	1	2,771	8,790	0.401	25
069	Production unit	1	2,771	8,790	0.344	42

*Resistance measured between pin and case, lower value of the two pins listed.

TABLE III-9
MEASURED SCRAM TIME ON NO.3 UNIT

Scram Clutch Temperature	Scram Time (sec)	
	To 23°	To Full-Out
200°F or less		
Initial, median	0.272	0.557
During test: minimum	0.233	0.482
During test: maximum	0.326	0.670
After test	0.240	0.478
930 to 980°F		
Initial, median	0.218	0.451
During test: minimum	0.180	0.411
During test: maximum	0.246	0.499
After test	0.115	0.245

The measured scram time of the control drum from the full-in position to 23° from full-in and to the full-out position were as shown in Table III-9. These values are well within the S8DR operational requirement of 0.500 sec to 31° (revision from the original 23°) and 2.00 sec to full-out position.

All parts of the scram mechanism were in operating condition. The only parts showing appreciable wear were the latching teeth on the snubber shaft and the latch pawl. The wear was reasonable after more than 1450 scram cycles, compared to the 55 maximum expected during S8DR operation. Minor chipping on some carbon-graphite bushings and aluminum-oxide coating on bearing shafts were also evident. These chips were in the noncritical areas of the bearings.

F. SHIELD MATERIALS DEVELOPMENT

1. Liquid-Lead Studies

The objective of the lead shielding program is to investigate the problems associated with the use of liquid Pb and lead-bismuth eutectic alloy as high-temperature gamma shielding for the SNAP Reactor Program. The initial phase of the program was to study the effect of long-term compatibility of the liquid metals with typical structural alloys. The current phase of the program is two-fold:

- a. To determine the change in mechanical properties of the structural alloys (1020 carbon steel, Croloy 2-1/4 and Type 446 stainless steel) after exposure to liquid Pb for up to 10,000 hr.
- b. To establish the level of carbon needed in Pb to prevent decarburization of the structural alloys and the resultant degradation of mechanical properties.

The mechanical properties test procedure was finalized, the capsules selected for testing, and work begun on removing the Pb from the capsules in preparation for machining tensile specimens from the tubular specimens.

A literature search was begun for data on the solubility of C in Pb. Also work was begun on measuring, by chemical methods, the amount of C in the Pb removed from selected capsules which had been exposed in the 10,000-hr compatibility test.

2. Aggregate Neutron-Gamma Shield Materials

The objective of the aggregate neutron-gamma shield materials program is to investigate the characteristics of mixtures of lithium-hydride with zirconium-hydride and LiH tungsten.

The current program has three major activities:

- a. The density particle-size study;
- b. Evaluation of hydrogen relocation in ZrH-LiH; and
- c. Physical and mechanical properties tests.

During fabrication of the initial ZrH-LiH test specimens, the ZrH had been crushed into coarse particles (up to about 1/4-in. mesh) and poured into the test capsules without vibration to densify the material.

This study was begun to determine the maximum density of ZrH which could be vibration-packed into a capsule using single sieve fractions and combinations of sieve fractions. Mechanical packing studies using single-diameter spherical particles have shown that a density of about 62.5% of theoretical density would be obtained. This value was essentially independent of the particle diameter and density.⁽⁴⁾

Since the ZrH particles are not spherical it was realized that to exceed the 62.5% limit a combination of sieve sizes would be required. Table III-10 shows the density data for various ZrH particle sizes and combination of sizes, including as-crushed ZrH. The procedure followed was to vibrate the larger particles to maximum density using a Syntron Platform Vibrator, Model VP80, and then add weighed amounts of the smaller material until the density ceased to increase. (Where a third, smaller-size particle was used the procedure was repeated.

The maximum density (4.42 gm/ml, 78.3% of theoretical) was obtained by the addition of -100 mesh powder to the +10 mesh material in the ratio 71.2 wt % of the +10 mesh and 28.8 wt % of the -100 mesh. This combination of particle sizes should produce a LiH-ZrH aggregate casting with a density of about 4.7 gm/ml and a wt % hydrogen of 2.12 which is roughly equal to the hydrogen content of stoichiometric ZrH_2 . The use of the LiH-ZrH aggregate as a substitute for ZrH_2 would eliminate the high dissociation pressure and difficult fabricability associated with massive ZrH_2 shapes.

TABLE III-10
DENSITIES OBTAINED FROM PARTICLE SIZE
COMBINATION STUDY

Particle Size*	Final Weight, gm	Final Volume, ml	Density gm/ml	% Theoretical
a	748.5	225	3.32	58.8
ab	823.5	240	3.43	60.7
abc	923.5	255	3.62	64.2
a	748.5	220	3.40	60.2
ac	1073.5	265	4.05	71.7
ace	1098.5	268	4.09	72.4
a	748.5	225	3.32	58.8
ad	1029.5	245	4.20	74.3
ade	1103.5	260	4.24	75.0
a	804.0	240	3.35	59.3
ae	1129.0	255	4.42	78.3
b	602.5	205	2.93	51.8
bd	947.5	245	3.86	68.3
bde	992.5	265	3.86	68.3
b	628.5	208	3.02	53.5
bc	803.5	240	3.35	59.3
bce	928.5	265	3.50	61.9
b	617.0	205	3.00	53.1
be	852.0	215	3.96	70.1
c	629.0	208	3.01	53.3
ce	754.0	230	3.27	57.9
As-crushed	-	-	3.72	65.8

*a = +10 mesh
b = -10, +20
c = -20, +80
d = -80, +100
e = -100

The H_2 -relocation study is aimed at investigating the extent of hydrogen redistribution in LiH-ZrH aggregates under a temperature gradient. The detailed observation of the specimens from the 1000-hr portion of the test was given in a previous report.⁽¹⁾

The 4000-hr portion of the test has been completed. The two capsules were cut open and the samples examined. The 1000°F sample from the 800 to 1000°F thermal-gradient capsule was found to have lost considerable LiH, exposing or freeing whole particles of ZrH. The LiH was found to have collected mainly in the 6-in. space between the two samples. The 800°F sample was essentially unchanged in appearance, except for a thin layer of LiH condensed on the end facing the hotter sample. The companion samples to these after 1000 hr of testing had exhibited only minor changes in appearance; however, these 4000-hr samples did very closely resemble the two 500 to 1000°F samples after 1000 hr.

The 1000°F sample from the 500 to 1000°F capsule had completely disintegrated leaving a pile of loose ZrH particles. The LiH was again found condensed in the region between the two specimens. The 500°F specimen was essentially unchanged except for a fairly thick layer ($\sim 1/16$ -in.) of LiH condensed on the end facing the hotter sample.

These samples plus those from the 1000-hr test will be submitted to chemical analysis for hydrogen-content measurement. The results of these analyses will be used to evaluate the merits of continuing the test to the planned 10,000 hr.

3. Combined Neutron-Gamma Shield Compounds and Alternate Neutron Shield Materials

The family of materials considered for use as shielding for the SNAP reactor today is limited basically to LiH as the neutron shield material, and W or U for the gamma-shield material. Current studies are investigating the potential usefulness of liquid Pb or Pb-Bi as a gamma-shield material, and LiH-ZrH or LiH-W aggregates as a combined neutron-gamma shielding materials.

With the increased interest in a manned reactor, visualized with a two-region shield where the lower or secondary shield experiences significantly reduced neutron and gamma dosages, it becomes possible to consider materials

TABLE III-11
NEUTRON SHIELDING COMPOUNDS

No.	Weight Percent Hydrogen	Compound	Formula	Molecular Weight	Density (gm/cc)	Melting Point (°F)	$\Sigma r/\rho$
1	12.7	Dimethylamine borohydroalane	$C_2H_{11}NBAl$	86.92	-	156	0.110
2	13.0	Bis (triethylammonium) decahydrodecaborate	$C_{12}H_{42}N_2B_{10}$	322.59	-	446	0.118
3	13.3	Bis (tetraethylammonium) decahydrodecaborate	$C_{16}H_{50}N_2B_{10}$	378.70	-	-	0.113
4	13.3	Nonaborane	B_9H_{15}	112.41	-	36	0.127
5	13.4	Lithium aluminum tetraethyl	$C_8H_{20}LiAl$	150.10	~0.9	-	0.115
6	13.7	1,4-Dimethyl-1,4 diaziniabicyclo [2,2,2] octane bis-octa hydrotriborate	$C_6H_{26}N_2B_6$	191.15	-	-	0.122
7	14.0	Dimethylamine bis-borohydroalane	$C_2H_{14}NB_2Al$	100.75	-	313	0.125
8	14.1	Hexamethyl triethylene tetramine bisborane	$C_{12}H_{36}N_4B_2$	257.91	0.94	247	0.120
9	14.1	Tetramethylammonium hexahydro- hexaborate	$C_8H_{30}N_2B_6$	212.18	0.88	-	0.126
10	14.2	Bis-trimethylaminealane	$C_6H_{21}N_2Al$	148.03	0.86	-	0.119
11	14.3	Cyclotriborazane	$H_{12}H_3B_3$	86.48	-	-	0.124
12	14.8	Hexamethyl triethylene tetramine tetra kis-borane	$C_{12}H_{42}N_4B_4$	285.58	0.90	410	0.125
13	14.9	Pentamethyl di-n-propylene triamine trisborane	$C_{11}H_{36}N_3B_3$	242.83	0.90	367	0.126
14	15.0	Pentamethyl diethylene triamine trisborane	$C_9H_{32}N_3B_3$	214.78	0.95	367	0.126
15	15.2	Lithiummethyl trimethylboron	$C_5H_{14}BLi$	91.86	-	-	0.125
16	15.3	Octamethyl triethylene diamine diammonium bis-octa hydrotriborate	$C_{14}H_{52}N_4B_6$	341.42	0.86	~662	0.129
17	15.9	Heptamethyl diethyleneamine diammonium bis-octa hydrotriborate	$C_{11}H_{45}N_3B_6$	284.34	0.86	244	0.132
18	16.0	Beryllium methylhydride	CH_4Be	25.05	-	-	0.141
19	16.1	Tetramethyl piperazinium bis-octa hydrotriborate	$C_8H_{36}N_2B_6$	225.24	-	-	0.134
20	16.2	Aluminum borohydride trimethylamine	$C_3H_{21}NB_3Al$	130.63	-	174	0.132
21	16.5	Decamethyl triethylene tetra ammonium tetrakis-octa hydrotriborate	$C_{16}H_{74}N_4B_{12}$	452.48	0.88	662	0.136
22	16.5	Octamethyl di-n-propylene tri- ammonium tris-octa hydrotriborate	$C_{14}H_{60}N_3B_9$	367.91	0.86	662	0.136
23	16.6	Trimethylammonium hydrotriborate	$C_3H_{12}NB$	72.94	0.81	201	0.135
24	16.6	Octamethyldiethylene triammonium tris-octa hydrotriborate	$C_{12}H_{56}N_3B_9$	340.86	0.85	581	0.136
25	16.7	Tetraethylammonium borohydride	$C_8H_{24}NB$	145.08	0.93	-	0.134
26	16.9	Aluminum borohydride	$H_{12}BAl$	71.53	-	84	0.120
27	16.9	Hexamethyl ethylene diamine bis- octahydroborate	$C_8H_{38}N_2B_6$	227.24	0.84	662	0.139
28	17.5	Beryllium borohydride trimethylamine	$C_3H_{17}NB_2Be$	97.80	-	95	0.144
29	18.0	Tetramethyl ammonium borohydride	$C_4H_{16}NB$	88.98	0.82	-	0.143
30	18.5	Lithium borohydride	H_4BLi	21.78	0.66	536*	0.165
31	20.7	Ammonium hydrotriborate	$H_{12}NB_3$	58.54	0.72	-	0.164
32	20.8	Beryllium borohydride	H_8B_2Be	38.70	-	-	0.171

* = decomposes

heretofore discarded because of temperature and radiation levels. In addition the possibility of using compounds of heavy elements and hydrogen, which would provide both neutron and gamma shielding, is enhanced.

A literature survey was made to determine potentially useful materials. Table III-11 summarizes the search for neutron-shielding materials with a higher hydrogen content than LiH. Calculation of $\Sigma r/\rho$ values (the ratio of macroscopic neutron-removal cross-section to density) which is a more realistic measure of performance than weight % hydrogen, revealed that only lithium borohydride, beryllium borohydride, and ammonium hydrotriborate possessed $\Sigma r/\rho$'s greater than LiH. Evaluation of the characteristics of these three materials essentially eliminated them from future consideration; for example, ammonium hydrotriborate has a low melting point ($<200^\circ\text{F}$), and beryllium borohydride decomposes at 250°F under 5 atm pressure.

Lithium borohydride was concluded to be the only one of the three materials which warranted a more careful evaluation. Table III-12 summarizes its known properties. LiBH_4 has been cold-pressed or cast into small laboratory specimens.

TABLE III-12
PHYSICAL PROPERTIES OF LITHIUM BOROHYDRIDE (LiBH_4)

Molecular weight	21.78
Color and form	White to grayish lumps and powder
Crystal system	Orthorhombic $a = 6.83 \text{ \AA}$ $b = 4.44 \text{ \AA}$ $c = 7.19 \text{ \AA}$
Melting point	$\sim 280^\circ\text{C}$ (536°F)
Decomposition at melting point	Forms Li, B, and H_2
Density	0.66 gm/cc
Vapor pressure	negligible below melting point
Heat capacity, C_p	19.73 cal/mol - $^\circ\text{C}$ at 25°C
Heat of formation, ΔH_f	-46.44 kcal/mol at 25°C
Heat of combustion, ΔH_c	-290.9 mcal/mol at 25°C

and it is believed that larger shapes could be fabricated. Its 530°F decomposition temperature would restrict its use to the lower shield. In this application it could provide a weight savings but at a high cost, since the current price of LiBH_4 is \$150 to 200/lb. Another factor which detracts from its usefulness is a complete lack of irradiation stability data.

It was generally concluded that the literature survey did not reveal any neutron-shielding materials which present any significant advantages over LiH . The nature of the materials possessing higher $\Sigma r/\rho$'s than LiH , the general lack of information, and high cost make it economically infeasible to embark on a long development program to bring any of the compounds to the present technological level of LiH .

The second portion of the literature survey was concerned with identifying compounds for combined neutron-gamma shielding. The screening criteria were a density of about 3 gm/ml and a weight percent hydrogen of about 3 or better. Table III-13 gives the compounds found during the survey. Evaluation

TABLE III-13
NEUTRON-GAMMA SHIELDING COMPOUNDS

Compound	Formula	Molecular Weight	Density (gm/cc)	Weight Percent	Melting Point (°F)	Comments
Cobalt hydride	$\text{Co}(\text{LiH})_4$	90.72	-	4.45	-	Preparation difficult.
Iridium hydride	$\text{Ir}(\text{LiH})_4$	223.99	~2.9	1.80	>1560	
Rhodium hydride	$\text{Rh}(\text{LiH})_4$	102.90	~3.15	3.00	1065	Dissociation pressure ~25 mm at melting point.
Iridium hydride hydride	$\text{Ir}(\text{LiH})_4\text{H}$	225.00	-	2.24	-	
Rhodium hydride hydride	$\text{Rh}(\text{LiH})_4\text{H}$	135.70	-	3.75	~1065	
Thorium borohydride	$\text{Th}(\text{BH}_4)_4$	291.41	2.59	5.49	~395	Decomposes at 320°F.
Potassium rhenohydride hydrate	$\text{K}_2\text{ReH}_4 \cdot 2\text{H}_2\text{O}$	304.46	-	2.65	-	H_2 evolution at ~520°F.
Barium rhenohydride hydrate	$\text{BaReH}_4 \cdot 4\text{H}_2\text{O}$	399.64	-	3.03	-	H_2 evolution at ~520°F.
Potassium rhenium hydride	K_2ReH_9	273.48	3.07	3.32	-	Difficult preparation and purification. Dissociates irreversibly at red heat.
Lithium barium hydride	LiBaH_3	147.30	3.76	2.05	-	
Lithium strontium hydride	LiSrH_3	97.58	2.88	3.02	-	
Lithium europium hydride	LiEuH_3	161.92	3.38	1.87	-	
Lithium gold hydride	LiAuH_4	207.94	-	1.94	-	
Lithium gallium hydride	LiGaH_4	80.69	-	5.00	-	Decomposes slowly at room temperature.
Tungsten hydride	WH_2	185.86	-	1.08	-	Gas, volatile at 40 to 120°F

of these materials led to the significant conclusion that in most cases the heavy elements which would provide the gamma shielding either were rare and expensive or required tedious and costly preparation and purification of the compound. The exception was lithium barium hydride. Laboratory samples of this material will be used to determine some of its basic physical and chemical properties which might make it worth further evaluation as a combined neutron-gamma shielding material.

G. NaK-PUMP DEVELOPMENT

The NaK-pump effort is focused on a multiloop dc conduction pump applicable to the Reactor Compact Thermoelectric Power System.

1. Pump Design Analysis

The recent reactor system studies had resulted in significant reduction in the primary-loop pressure drop. To determine the effects of the pressure reduction on the pump design, a parametric study of a two-pass pump system was made using the following criteria:

- a. Pump power supply with the performance shown on Figure III-9.
- b. Primary-loop flow of 3.25 lb/sec per quadrant at 1044°F.
- c. Secondary-loop (heat-rejection-loop) flow of 3.1 lb/sec per quadrant at 663°F.
- d. Identical pump throat for both loops.
- e. Bus length of 24 in.

Pumps of various geometries, current levels, and magnetic-flux densities were compared on basis of minimum pump-system weight per output pressure rise. The results of the study, as shown on Figure III-10, indicate that the weight increases only slightly with secondary-loop output pressure rise up to approximately 2.2 psi. The rapid weight increase at higher pressure rises is attributed to the required increase in the pump efficiency. The increase in the pump efficiency is necessary to offset the performance limitation of the power supply.

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Figure III-9. Pump Power Supply Characteristics (Three modules connected in parallel per pump)

Figure III-10. Pump System Weight, Two-Loop System

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Because the total system weight is minimum at about 2.2 psi, this output pressure rise was selected as a new reference design point. The 55-lb pump-system weight compares to 110 lb for the three-pass pump described previously.⁽¹⁾ The parameters of the new reference pump are listed on Table III-14.

TABLE III-14
REFERENCE PUMP PARAMETERS

Channel width	1.0 in.
Channel height	0.5 in.
Channel length	2.5 in.
Magnetic-flux density	16,000 lines/in. ²
Pump weight	11.0 lb/quadrant
Converter weight	30.0 lb/quadrant
Bus weight	14.3 lb/quadrant
Pump-system weight	55.3 lb/quadrant
Current	1800 amp
Secondary pressure rise	2.20 psi
Secondary flow	3.1 lb/sec
Primary pressure rise	2.49 psi
Primary flow	3.25 lb/sec
Pump efficiency	28.2%
Bus efficiency	73.2%
Converter efficiency	3.13%
Pump-system efficiency	0.65%
Heat input	9.54 kw/quadrant
Hydraulic power	61.7 watt/quadrant

The schematic of the electrical and hydraulic design is shown on Figure III-11. The layout design of the two-pass pump, with a cross-section as shown on Figure III-12, was started.

2. Pump Stress Analysis

Preliminary stress analysis on the three critical areas of the pump was made. While the actual analysis was based on the design of the original three-pass pump, the results are basically applicable to the new two-pass design. The critical areas are the pump throat, bus bar, and the magnet.

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Figure III-11. Two-Pass Pump Schematic

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Figure III-12. Cross-Section of Two-Pass Pump

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The highest load on the pump throat occurs during the launch due to the internal pressurization of the NaK. Without external support the 316 stainless-steel throat will be stressed above the yield strength. It is planned to use mica insert between the throat and the magnet pole pieces to provide the necessary external support. The negative pressure in the system which occurs during the NaK-loading process will collapse the throat unless internal supports are provided. Internal pins, electron-beam welded between the walls, should adequately support the throat without significantly affecting the pump performance.

The estimated 500°F differential temperature within the copper bus bar between the primary and secondary throats induces bending stresses in the bus. These stresses are, in turn, transmitted to the bus-throat braze joints and the throats. The resultant stresses can be limited to an allowable level by mounting the throats to permit relative motion between them.

The thermally induced stresses in the magnet due to the maximum temperature differential across it are below the allowable limits of the Alnico 5-7 material.

3. Pump Test Loop

The test facility for the pump test including the NaK loops was designed. The pump test fixture consists basically of a vacuum chamber in which the pump will be mounted and two NaK loops, one for each throat. Each loop is independent except for the common dump tank. The loop is provided with a flowmeter, a flow control valve, cold trap, and heaters. The calculated pressure drop in each loop is 0.7 psi for a NaK flowrate of 40 gpm (3.8 lb/sec at 1100°F). An electrically heated shield surrounding the pump provides the temperature control.

The supporting framework for the test facility has been fabricated and installed. The fabrication of the two NaK-test loops and the dump tank is in progress.

H. ZrH REFLECTOR DEVELOPMENT

Evaluation of the BeO literature resulted in the selection of cold-pressed Brush Beryllium Thermolox 995A for preliminary reflector drum tests. Delivery of materials is expected at the end of October.

Pellets of 60 wt % rare earth oxides (REO) in a continuous nickel matrix were successfully vacuum hot-pressed at 2280°F. Figure III-13 is a photomicrograph of a pellet. Hot-pressing produced the greatest compact density (88.4% of theoretical density) compared to both isostatic pressing and cold-pressing followed by sintering. Most of the voids in the compact were in the REO particles. A graphite die was designed and fabricated to produce full-size absorber crescents. The first attempt to produce a full-size crescent resulted in incomplete sintering. The hot-press induction heater (15 kw) lacked sufficient power to raise the die temperature to 2280°F. A larger (50-kw) induction heater is being installed.

The control-drum bearing friction torque was calculated based on torque data generated by the high-temperature bearing test (Section III-C). These values were: maximum friction torque of 25.2 in.-lb at ambient and 11.6 at 1350°F.

60 wt % RARE EARTH OXIDE IN Ni.
88.4% THEORETICAL DENSITY.
HOT-PRESSED AT 2280°F.
REO PARTICLE SIZE, 40 + 60 MESH.

Figure III-13. Photomicrograph of Absorber Pellet

A reference control-drum design, Figure III-14, was prepared. The following materials were tentatively selected for the drum:

TABLE III-15
MATERIALS FOR REFERENCE CONTROL DRUM

Component	Material
Neutron absorber	Eu_2O_3 in Ni
Neutron reflector	BeO
Drum primary structure	Nb-10W-10 Hf-0.1Y (WC129Y)
Cladding	Nb-1 Zr
Shafts	Ta-10W
BeO mounting bolts	WC129Y
Nickel-columbium diffusion barrier	Molybdenum foil
Absorber mounting bolt nuts	Molybdenum TZM

A parametric study was completed to determine the maximum permissible radial "g"-load on the drum as a function of strongback thickness, strongback density, and margin of safety, and for different specific strongback materials. The study included a fatigue analysis of the stresses around the strongback holes, which indicated that the fatigue life was probably adequate to withstand the present launch-design parameters. This study showed that if the strongback is made of 1/4-in. columbium alloy Nb752 and if the material around the strongback holes is allowed to yield (thus compensating for stress riser effects), the drum can sustain a static radial "g" load of 68 g's with a margin of safety of 0.50 against yield of the gross cross-section of the strongback.

Preliminary analysis of the thermal stresses in the BeO and irradiation growth of the BeO were completed. The thermal stress in the BeO was found to be approximately 13,400 psi at steady-state conditions of 600 kwt, 1100°F inlet temperature and emissivity 0.8. The growth of the BeO due to irradiation was calculated to be 0.75% based on a power level of 1200 kwt, a flux level of $2.5 (10)^{13} \text{ nv}/(\text{E} \geq 1 \text{ mev})$, and a drum temperature of 1800°F.

A more refined thermal-stress analysis is continuing.

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Figure III-14. Control-Drum Design, Reference ZrH Reactor

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IV. SNAP 8 DEVELOPMENTAL REACTOR

A. INTRODUCTION AND SUMMARY

1. Core

Hydrogen-distribution calculations for the fuel rods have shown that the fuel-to-clad gap is adequate to accommodate the predicted fuel growth over the entire core. By additionally specifying fuel-element loading on an individual basis according to diametral gas gap the core fuel element survival margin was optimized.

The accuracy of the critical-loading calculations was confirmed by the actual dry-critical loading tests performed during this quarter. Criticality was predicted with 165 elements (out of 211) and was achieved with 169.

2. Assembly and Acceptance Testing

Acceptance testing of the reflector assembly was completed. Some malfunctions occurred during testing of the control assembly and were identified and corrected. To minimize possibility of operational scrams due to malfunction a special safety network was incorporated into the manual controllers. The controllers were also redesigned and modified to increase their current output capacity without overloading any components.

3. Facilities and Test Preparation

NaK circulation at operating temperature was checked, and after correction of two problems proper circulation was achieved. The first problem was elimination of a blockage, probably paper, in the primary loop, and the second was leakage at the mouth of the primary electromagnetic (EM) pump throat. New replacement pump throats are being fabricated for both the primary and secondary pumps.

4. Test Operations

After completion of the non-nuclear checkout test final leakrate tests were made. The nuclear instrumentation detectors were installed and the instrumentation and control checkout performed. Fuel elements were then loaded into the core on a one-for-one basis with the stainless-steel dummy elements, and

Figure IV-1. Calculated Average Ring-IV Hydrogen Redistribution in S8DR

criticality was achieved on the 10th loading step. Dry critical physics tests were conducted, and control-drum calibrations were obtained.

B. DESIGN AND ANALYSIS

1. Hydrogen Redistribution

The hydrogen redistribution within the S8DR fuel rods has been calculated with the FUSAK computer code⁽⁵⁾ as a function of core position and reactor power level. The redistribution is of interest because of its effect on the fuel growth in addition to the reactivity loss which results as a consequence of it. While the growth is not as well understood as the loss effect, it must nevertheless be evaluated in light of the S8ER fuel-growth data and the fuel-irradiation experiments.

The data in Figure IV-1 show the calculated axial hydrogen redistribution for a fuel element in Ring IV of the core at power levels of 600 and 1000 kw. The figure also indicates the radial redistribution as a band about the average values at the core midplane. The redistribution in Ring IV is typical of the redistribution throughout the core. In addition the figure indicates the axial location of an average fuel temperature of 1230°F for both power levels. This is of interest because of possible Type-III (double-boundary) fuel growth which could occur.^{(1,6)*} Such growth would cause localized areas exhibiting fuel growth larger than the adjacent "non-boundary" areas.

This study indicated that such growth was possible over the power portion of the rod from Z/L (normalized height from the bottom of the core) of 0.3 to 0.45. However, the available fuel-to-clad gap at those element elevations is sufficiently large to accommodate such growth.

The hydrogen-redistribution time constant⁽⁷⁾ is projected to be 10 ± 5 hr for operation at 600 kw 1300°F outlet, and 25 ± 10 hr for operation at 1000 kw 1100°F outlet.

*Type-III double-boundary growth was initially designed Type I or IS-27 type double-boundary growth in Reference 6, pp. 127, 128.

2. Specified Fuel-Element Loading

The fuel-element loading for S8DR was specified on an individual element basis, i. e., each core location had a specific element assigned to it. This was done to increase the probability of maximum fuel-element survival. The fuel elements with the largest diametral gas gaps were placed in the core locations with the maximum predicted fuel growth (Ring IV), and elements with the smallest gaps were placed in the area of lowest predicted fuel growth (Ring IX), with intermediate-gap elements placed at locations of corresponding intermediate fuel growth. The fuel elements which had been designated as the ones for detailed postirradiation measurements were placed in core locations that had outlet coolant channel thermocouples on at least two of the six adjacent coolant channels.

3. Prediction of S8DR Critical Fuel Loading

The calculations of the S8DR critical fuel loading were performed using the ANISN transport-theory code, and the results normalized to the S8ER critical fuel loadings. Measured critical fuel loadings for three reflector shim configurations and the excess reactivity of the fully loaded fully shimmed S8ER were used as the normalization points. Since S8ER and S8DR are very similar in core design and size, it was assumed that the normalization obtained from the S8ER calculations could be applied to the S8DR calculations. It was necessary, however, to account for the difference in the effective reflector thickness and for the difference in the dummy fuel elements. The critical loading in S8DR was performed by replacing stainless-steel dummy elements with enriched fuel elements while lucite dummy fuel elements were used in S8ER.

The calculated critical fuel loading for S8DR was 161 ± 7 enriched elements with the remaining elements in the core being stainless-steel dummy elements. The reflector was assumed to be the nominal 5-in.-thick reference configuration with all drums in the most reactive position (fully inserted). The plus or minus value on the number to critical was the $1-\Sigma$ uncertainty. The actual critical loading of 164.5 elements was well within the quoted uncertainty band.

4. Structural Analysis of Primary-Loop Components

A review of the components of the primary coolant loop to verify component performance margin had indicated that several areas required further analysis. These were the NaK-pump headers, the inlet nozzle in the surge tank, and the primary inlet on the intermediate heat exchangers (IHX). A more detailed analysis of these components during the past quarter has shown that the NaK-pump headers are satisfactory as designed, and the modified surge-tank inlet with a thermal-shock barrier is also satisfactory and meets the requirements of the ASME code. The programmed system startup combined with potential "cold slugs" did create a marginal design condition for the IHX inlet. This was readily avoidable, however, by administrative control.

Stresses in the NaK-pump header were determined for gradients resulting from a start transient in which the NaK-temperature increased from 600 to 1100°F at 2.5°F/sec. The maximum calculated secondary-stress intensity was 8850 psi compared to an ASME code allowable of 39,900 at 1100°F for this type of stress. The maximum thermal stresses in the pump header occur around the end cap and vortex plug which are relatively slow in responding to rapid changes in NaK temperature.

The inlet nozzle to the surge tank was analyzed using a finite-element stress-analysis code and incorporating a relatively detailed thermal analysis of two critical transients. They were the scram on high power and the loss of primary flow followed by a scram on low flow. The results indicated that with the thermal shield and automatic flow setback in the primary loop, the critical stresses are localized in nature and are within the allowables of Section III of the ASME Code for Nuclear Vessels.

The critical section of the IHX is the joint between the inlet nozzle and the massive upper head plate. The analysis of this area was complicated by the unsymmetrical nature of the inlet geometry and the resultant complexities in the thermal gradients. As for the inlet nozzles, a finite-element stress-analysis code was used. As a check an independent analysis was performed on one case using a different finite-element code. The results were in agreement.

Initially the stresses at the joint were obtained for thermal gradients resulting from an assumed startup transient in which the NaK temperature increased from 600 to 1200°F at 2.5°F/sec. The resulting stresses exceeded the allowable maximum secondary-stress intensity from the ASME code. The next startup transient considered assumed a 300°F increase in NaK temperature at 1.5°F/sec. The calculated stresses were still a factor of almost 2 greater than the code allowable.

A system response study was then used to define more accurately the expected thermal response of the IHX to the programmed startup. It was found that a reactivity insertion rate of 0.5 β /min resulted in a maximum NaK-temperature rate-change of about 0.35°F/sec over a temperature change of <150°F. The resulting stresses were within code allowables.

The maximum secondary-stress intensity range occurs at the inside surface of the nozzle adjacent to the weld and is created by a startup and postulated cold-restart combination. With this combination of 25,800-psi secondary stress due to the 0.35°F/sec start ramp and a 31,000-psi stress due to a 150°F cold-restart slug, a stress range of 56,800 psi is obtained which is above the secondary allowable of about 36,000 psi. If the cold-restart slug is 50°F instead of 150 the secondary stress (due to cold restart) would be 10,330 psi. The stress range for the start and cold-restart combination would then be 36,100 psi, which is acceptable. The cold slug at the IHX primary inlet is a consequence of restarting the primary-loop flow with nonisothermal NaK conditions. Since this is subject to administrative control using existing instrumentation no problem is foreseen in reducing the allowable cold slug to 50°F.

5. Cladding and Structural Metal Fracture Processes

In support of the continuing evaluation of fracture processes as related to fuel and cladding structural integrity, the problem of grain-boundary void and/or bubble growth has been studied analytically. The treatment assumes that the concentration and shape of the void and/or bubble nuclei (neutron-induced helium bubbles) at the beginning of the growth stage can be measured experimentally. Numerical results have been obtained using the following parameters as variables: average grain-boundary void dimension and concentration, applied tensile

stress, temperature, grain-boundary and surface-diffusion coefficients, grain-boundary and surface energies, average grain size, and initial-void internal pressure. Some of the most important findings are the following: The initial grain-boundary void concentration is very important, as shown in Figure IV-2. Here failure is defined as the condition when void-tip separation approaches zero.

The initial-void dimension does not appear to be very critical. For example, increasing the initial-void diameter (parallel to the grain boundary) from 70 to 500 Å reduces the time-to-failure by only 10%.

The absolute values of the diffusion coefficients (which are related to temperatures exponentially through an activation energy term) are important. For a given grain-boundary diffusion coefficient, which determines the rate of vacancy flux to the void tip, the influence of the surface-diffusion coefficient on time-to-failure becomes relatively more important when it is less than the grain-boundary diffusion coefficient rather than when it is greater. With other parameters held constant, increased applied tensile stress produces a decreased time-to-failure as shown in Figure IV-3.

Since the steady-state creep rate of a metal generally follows a hyperbolic sine function of the applied tensile stress, a minimum (experimentally observed) in the total creep-strain-to-fracture is expected in the fracture strain vs applied stress (or steady-state creep rate) plots.

C. ASSEMBLY AND ACCEPTANCE TESTING

1. In-Core Instrumentation Assembly

Splicing of the 1/8-in.-diam in-core instrumentation thermocouples to the 1/16-in.-diam extension thermocouples using the tungsten-inert-gas (TIG) welding process was successfully completed. All tests of the completed vessel-head assembly with the in-core instrumentation were also completed. After fuel loading, the head assembly was welded to the core vessel.

2. Reflector and Control Assembly

The acceptance testing (Figure IV-4) of the S8DR reflector assembly was completed and the assembly delivered to the ground test facility at Santa Susana. Because of several incidents during the test some rework and retesting were required.

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Figure IV-2. Grain-Boundary Void Density

Figure IV-3. Failure Time vs Stress

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Figure IV-4. S8DR Reflector and Assembly Acceptance Testing

Temporary modification of the manual controller to increase its current output to the actuator for rewinding the scram spring at room condition was previously described.⁽¹⁾ Following the partial repeat of the acceptance test the ambient test procedures were again started. At this time one control-drum actuator did not operate properly because of damage to the pinion gear. The damaged actuator was replaced with a spare unit. Careful removal and/or inspection of all the other actuators showed no other damage.

From the evaluation of the events before the actuator pinion failure it was concluded that the damage resulted from an inadvertent nonoperational test procedure. At the time the reflector assembly was being inspected for the scram-spring rewind problem this drum was scrambled in a manner which subjected

the pinion teeth to the inertia of the actuator rotor spinning at ~ 850 rpm. Resultant loading on the pinion was high enough to fracture the tooth.

During the brief thermal-vacuum retest following the start of the ambient, the brake of another actuator malfunctioned. The malfunction was traced to a short within the pigtail cable assembly near where it is clamped to the brake housing. The fiberglass insulation on the cable conductor had become frayed, allowing contact between the two conductors. It appeared that the cable had been flexed excessively, probably during the removal and inspection of the actuator for pinion-gear damage. Inspection of the brake-cable assembly on the other actuators revealed no evidence of excessive flexing.

The defective brake assembly was replaced and after another series of retests under thermal-vacuum conditions the balance of the ambient phase of the acceptance test was completed.

It was determined that a similar situation to the pinion failure could result operationally if the phase circuit of the actuator opened during stepping. Malfunctions of the controller, actuator, cable harness, facility wiring, or relays could cause an open circuit. To minimize the probability of this type of scram a special safety network (Figure IV-5) was incorporated into the manual controllers. This network will sense an open or low-current condition in the

Figure IV-5. Safety Circuit for Manual Controller

actuator phase circuit and immediately shut the controller off, precluding the occurrence of other events necessary to cause this type of scram. The controllers were also redesigned and modified to increase their current output capacity to 6.0 amp without overloading any components. The output currents to the actuator phases and brake, which are now adjustable, were set at 5.4 and 2.2 amp.

D. FACILITIES AND TEST PREPARATION

NaK loading of both the primary and secondary heat-transfer systems was completed at the end of the previous report period. Following completion of the loading, NaK circulation was started. Satisfactory flowrate and pressure drops were readily established in the secondary system but could not be done in the primary loop. A detailed step-by-step sequence was started to determine the cause of the problem. This investigation established that a flow blockage was located in the vacuum vessel in either the reactor inlet or outlet line. The NaK was then drained from the primary system and x-rays of the suspect areas taken. A flow blockage was observed in the reactor inlet line in a pipe-to-tubing reducer. Several possible methods for the plug removal were developed and reviewed. It was decided that the portion of piping containing the flow blockage would be cut out, the plug removed from this portion, and the pipe section then reinstalled into the system. A detailed step-by-step procedure for this operation was prepared and a complete dry run performed using the S8DR remote-handling mockup. The actual operation was carried out inside the S8DR vacuum vessel under an argon blanket to prevent NaK oxidation. The plug material was not positively identified because it burned upon removal from the pipe section. On the basis of visual observations the plug material appeared to be a paper substance.

Following removal of the flow blockage satisfactory flow and pressure drops were achieved in the primary loop. System heatup was then begun in accordance with the approved test procedure. At a system temperature of $\sim 1100^{\circ}\text{F}$ the primary electromagnetic pump failed when NaK leakage from the throat seeped into the EM pump stator coil insulation causing a short circuit. The system was immediately cooled down and the NaK drained. Upon initial visual examination

several small cracks were found in the pump throat. Based on detailed metallographic examination, the failure was attributed to intergranular stress-accelerated corrosion. It was also noted in the metallographic examination that the pump-throat material was in a highly stressed condition and had not been properly annealed following the tube-forming process. It was concluded from this that the secondary pump throat might also fail from the same cause since it was of the same manufacturing technique and had seen a similar checkout and storage history. Therefore it was decided to fabricate replacement pump throats for both the primary and secondary systems. The manufacturing process for the new pump throats was designed to eliminate the possibility of any high residual stresses following fabrication.

To enable continuation of the non-nuclear checkout test while the new pump throats were being fabricated the failed EM pump was replaced with a spare from the S8ER program. Following completion of this replacement the initial system heatup to 1300°F was continued to permit a complete "at-temperature" functional checkout and to perform the remote primary-NaK sampling procedure. The heatup was successfully completed with all systems performing satisfactorily. A summary of non-nuclear test results is given by Table IV-1.

TABLE IV-1
NON-NUCLEAR TEST RESULTS

Parameter	Measured Value	Required or Expected
Vault leakrate, Hg/hr at 4 psig	0.07	≤ 0.12
Vacuum system capability, torr	1×10^{-6}	$\leq 1 \times 10^{-5}$
EM pump capacity, gpm		
Primary	170	170
Secondary	148*	170
Loop pressure drop, psi		
Primary	16	20
Secondary	26	30
NaK sample at 1300°F	Obtained	Yes

*Partial power

The NaK-sampling procedure was also successfully completed. An 80-gm sample was obtained using remote-handling techniques and equipment designed for on-line sampling during nuclear operations.

E. TEST OPERATIONS

Upon completion of the initial phase of the non-nuclear checkout test, the vault-sealing and leakrate testing were started. Considerable effort was required in this area for searching out leakage paths and making permanent repairs. A major effort was required to repair the vault liner in the region of the vacuum-pumping duct and cooling-water line penetrations. The results of the final leakrate tests are given in Table IV-1. In preparation for fuel loading, the S8DR reactor and reflector assembly was then installed on the Building 059 vault floor, and the nuclear instrumentation detectors around the assembly. A complete reactor-instrumentation and control-system checkout was performed.

The S8DR core was initially loaded with 211 stainless-steel dummy fuel elements. Fuel elements were then loaded into the core on a one-for-one interchange basis with the stainless-steel dummies. Since the core location for the dummy elements had previously been identified, the interchange procedure eliminated any possibility of loading a fuel element into the wrong core position. The loading sequence was started at the center of the core and progressed outward by hexagonal rings. Criticality was achieved on the 10th loading step with 169 elements in the core. The extrapolated critical loading was 165 elements. (A critical loading of 161 ± 7 had been analytically calculated.) During the fuel-loading operation five site-power outages or fluctuations were experienced. In every case the uninterruptible power system performed and enabled testing to continue unperturbed.

Dry-critical physics tests were made with the fully loaded core to determine drum calibrations, shutdown margin, and excess reactivity. Table IV-2 summarizes the results of these experiments and shows the analytically predicted values.

Drum calibrations were obtained for various configurations so that the effects of shadowing could be determined. These data are currently being

analyzed to obtain a single curve for each drum with varying amplitude factors to account for the shadowing effects. A typical experimental curve is shown in Figure IV-6, along with the analytically predicted curve.

TABLE IV-2
DRY CRITICAL TEST RESULTS

Parameter	Measured	Predicted
Fuel rods required for criticality	165	161 ± 7
Control drum worth, \$		
Neighbors full-in	4.40	3.90
Neighbors full-out	2.60	3.12
Fully loaded excess reactivity, \$	14.10	13.90
Shutdown margin, \$	6.90	7.16
Element worth in outer ring, ¢	<20	25

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Figure IV-6. S8DR Control-Drum Reactivity
(Drum No. 6)

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Figure V-1. Performance Map, Reference ZrH Reactor

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V. REFERENCE ZIRCONIUM-HYDRIDE REACTOR STUDIES

The conceptual design phase of the reference ZrH reactor was completed. The reactor contains 295 S8DR-type fuel elements and can provide 600 kw of thermal power for 26,000 hr, with an outlet coolant temperature of 1300°F. The overall performance envelope is shown in Figure V-1 and the reactor layout in Figures V-2, -3, and -4.

The zirconium-uranium alloy fuel material is hydrided to a hydrogen content (N_H) of 6.3×10^{22} at./cm³, which compares with a nominal content of 6.05 for S8DR. Other differences from S8DR include an increase in the Hastelloy-N cladding thickness from 0.011 to 0.015 in. and larger clearances between the fuel pin and cladding to accommodate fuel growth.

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

The major change from previous zirconium-hydride reactors lies in the design of the reflector control system. The SNAP 10A and 8 reactors were designed to operate in a shadow-shielded configuration. The control drums were semicylinders of beryllium around the core which could be rotated to increase or decrease the neutron leakage from the system. The reference ZrH reactor control system is designed to operate inside the "4 π " shield which completely surrounds the reactor. Ten cylindrical reflector-poison control drums are mounted in "dry wells" cooled by the inlet NaK. The control drums will operate at relatively high temperatures (1500 to 1800°F) because the nuclear heat generated in the drums must be transferred by radiation to the

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Figure V-2. Reference ZrH Reactor Design, Side View

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Figure V-3. Reference ZrH Reactor Design, Core at Upper Gridplate

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Figure V-4. Reference ZrH Reactor Design, Core at Lower Gridplate

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walls of the dry well, which are cooled by 1100°F NaK. Beryllium oxide is used as the reflector material in this high-temperature environment. The selected poison material is europium oxide dispersed in nickel, and high-strength niobium alloys are planned for structural use.

The control drum rotates in a set of ball-and-socket bearings. As in S8DR an aluminum-oxide-coated shaft rotates in a carbon-graphite ball mounted in a socket coated with Al_2O_3 . The maximum bearing temperature is expected to be below 1350°F, and testing of this type of bearing has been conducted at temperatures up to 1500°F. The control drum will be rotated by an S8DR actuator which has been modified slightly for this application.

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

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

Some of the more significant design parameters for the systems are given in Table V-1. In general the design involves very few deviations from the state-of-the-art, as exemplified by the S8DR. In areas where conditions are more severe than S8DR, development programs are underway to establish the required capabilities. For example, high-density experimental Eu_2O_3 -Ni

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dispersions have been fabricated, prototype bearings have been tested at 1500°F for 1000 hr, experimental shock and vibration suppressors for the fuel have been fabricated, and molten-Pb specimens have been tested for 10,000 hr at 1500°F. The engineering design and development of the reference ZrH reactor will now proceed from the conceptual design phase into detailed preliminary design.

TABLE V-1
REFERENCE ZrH REACTOR DESIGN PARAMETERS
(Sheet 1 of 3)

1. General Performance

Design thermal power	600 kwt
Outlet temperature	1300°F
Temperature rise	200°F
Design lifetime	26,000 hr
Coolant flowrate	49,000 lb/hr
Pressure drop	0.65 psi
Operating pressure	35 psia
Peak fuel temperature	1420°F
Peak cladding temperature	1320°F
Short-term reactivity losses	- \$2.16
Long-term reactivity losses	- \$11.59
Burnable-poison reactivity gain	+ \$10.00
Reactivity margin	- \$1.50
Required initial excess reactivity	\$5.25
Peak control-drum BeO temperature	1500°F
Peak control-drum poison temperature	1570°F
Peak control-drum cladding temperature	1510°F
Control-drum bearing temperature	1270°F
Control-drum actuator temperature	950°F

2. Fuel and Core Internals

Number of fuel elements	295
Type of fuel elements	Modified S8DR
Hydrogen content, N_H	6.3×10^{22} atoms/cm ³
Wt % uranium	10.5

TABLE V-1
REFERENCE ZrH REACTOR DESIGN PARAMETERS
(Sheet 2 of 3)

Hydrogen-barrier material	SCB-1
Fuel-clad gap	Optimized
Cladding material (reference) (alternate)	Hastelloy-N Incoloy-800
Cladding thickness	0.015 in.
Burnable poison material	Gd-155
Active fuel length	16.8 in.
Overall (flat-to-flat) element length	17.5 in.
Fuel-element OD	0.570 in.
Shock and vibration suppressor	To be determined
Fuel-element spacing:	(Variable for flow control)
Maximum	0.045 in.
Nominal	0.030 in.
Minimum	0.025 in.
Internal reflector material	BeO
Internal reflector cladding	316 stainless
Gridplate material	316 stainless
Core-shell ID	11.4 in.
Core-shell thickness	0.030 in.
Core-shell material	316 stainless
Reactor coolant material	NaK-78
3. Reactor Vessel	
Vessel OD	22.0 in.
Vessel wall thickness	3/16 in.
Dry-well ID	4.62 in.
Dry-well wall thickness	0.065 in.
Overall vessel height	29.3 in.
Number of inlet and outlet nozzles	4 ea.
Nozzle size	1.5 in.

TABLE V-1
REFERENCE ZrH REACTOR DESIGN PARAMETERS
(Sheet 3 of 3)

4. Reflector Control System

Control-drum type	Reflector-poison
Number of control drums	10
Reflector material	BeO
Poison material	60% Eu_2O_3 in Ni
Candidate structure material	Nb-10W
Candidate cladding material	Nb-1 Zr
Drum diameter	4.5 in.
Drum length	19.0 in.
Structural strongback thickness	0.25 in.
Cladding thickness	0.030 in.
Drum bearing-shaft material	Ta-10W
Shaft diameter	0.75 in.
Shaft plating	Al_2O_3
Bearing ball material	Carbon-graphite
Ball diameter	1.375 in.
Bearing socket material	Ta-10W
Socket plating	Al_2O_3
Control-drum actuator type	Modified S8DR

5. Shield

High-temperature core-gamma shield	Molten Pb
Lead container material	Croloy 2-1/4
Gallery gamma shield	U
Neutron shield material	LiH
LiH container material	316 stainless

VI. THERMOELECTRIC SYSTEM STUDIES

This section describes the progress made in the conceptual design of a reactor-thermoelectric power system which utilizes the reference ZrH reactor and the lead-telluride tubular thermoelectric module which is being developed for the AEC by the Westinghouse Astro-Nuclear Laboratory. The objective of this project is to establish a conceptual design of a "reference" system suitable for adaptation and a minimum lifetime of 20,000 hr. This effort includes preparation of parametric performance data covering a suitable power range above and below 25 kwe to illustrate the trade-offs available between power, weight, area, etc.

A summary of the progress towards achieving the above objective follows.

A. REFERENCE SYSTEM

The reference 25-kwe reactor-thermoelectric system design developed in this study consists of the zirconium-hydride reactor coupled by the primary-coolant loop to four parallel power-conversion sections (see Figure VI-1).

Figure VI-1. Reference 25-kwe System Schematic

Each of the four power-conversion sections includes a thermoelectric converter assembly, a NaK pump, and a heat-rejection loop. Each converter assembly contains 24 thermoelectric tubular modules structurally supported within a rectangular box and interconnected with coolant manifolds for proper flow distribution of the hot-primary and the cold heat-rejection loops. The tubular modules are arranged in parallel "4-packs", each of which provides slightly more than 1 kwe at 56 volts dc. Blocking diodes are used for each 4-pack to isolate possible short-circuits to ground.

The NaK pumps are of the dc-conduction, electromagnetic type. Each pump has 3 throats, of which two pump 1/4th the total primary-loop flow and the third pumps one of the heat-rejection loops. A separate thermoelectric power pack consisting of 3 high-current modules connected in parallel provides the dc current to each pump.

Each heat-rejection loop consists of a radiator section, an expansion compensator, and interconnecting piping and manifolds. The radiator design selected as a reference is of the conventional fin-and-tube type, with stainless-steel NaK tubes diffusion-bonded to aluminum fins and meteoroid armor. Alternate radiator materials and concepts with potential performance improvements were also identified. The possible use of heat pipes for the heat-rejection side of the power-conversion system was also studied, but is not recommended now due to inexperience with a suitable working fluid for the temperature range of interest.

A reference startup-and-control scheme was defined which uses reactor outlet temperature as the control parameter during startup and electrical current output during normal operation. No valves or moving parts other than the reactor control drums are required. A shunt-type voltage regulator which dissipates excess power to space through high-temperature resistance elements was selected for voltage and load control.

The major design and performance characteristics for the reference reactor-thermoelectric system are summarized in Table VI-1. The reactor thermal power and coolant outlet temperature are well within the capabilities of the reference ZrH reactor, and provide adequate margins for reliability, power growth, and/or compensation for possible system degradation. The

TABLE VI-1
REFERENCE REACTOR - THERMOELECTRIC SYSTEM DESIGN
AND PERFORMANCE CHARACTERISTICS

Electric power, kwe	
Gross	25.2
Regulated	24.7
Voltage, volts dc	56
Reactor thermal power, kwt	583
Reactor outlet temperature, °F	1246
Average radiator temperature, °F	563
Net efficiency (system), %	4.24
Unshielded weight, lb	6830
Specific weight, lb/kwe	271
Radiator area, ft ²	1400
Specific area, ft ² /kwe	56
Design lifetime, hr	20,000
System degradation allowance, %	10

unshielded system weight and radiator area correspond to a near-minimum weight design; the area can be reduced at the expense of increased weight. The specific weight of approximately 271 lb/kwe and specific area of 56 ft²/kwe represent a significant improvement over previous designs.

Parametric studies were performed to determine the system characteristics at higher and lower power levels, using a computer code developed for this purpose during the study. From 15 to 35 kwe the unshielded system specific weight ranges from approximately 320 to 250 lb/kwe, respectively. The specific area for near-minimum weight remains in the range of 56 to 60 ft²/kwe, and the reactor thermal power ranges from 350 to 800 kwt.

B. POWERPLANT FOR ORBITAL WORKSHOP

In adapting the reference system described above for specific space missions major emphasis was placed on establishing a conceptual powerplant design suitable for integration with the Saturn-V Orbital Workshop (OWS) concept, currently being studied by NASA under the Advanced Apollo Program.

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The power system design is similar to the reference system except that the radiator area has been reduced slightly at the expense of a small weight penalty, and radiation-shield designs tailored for this mission have been included.

C. LUNAR BASE AND MANNED ORBITING RESEARCH LABORATORY (MORL) POWERPLANTS

Two reactor-thermoelectric powerplant designs evolved in previous joint NASA/AEC studies were updated to incorporate the design and performance changes available with the reference system. In both cases significant improvements were made in system simplification, specific power, and specific weight. For the Lunar Base powerplant, for example, it now appears possible to increase the net power from 20.4 to 35.5 kwe, using the same constraints and criteria established in the previous joint study. For the MORL powerplant the weight and area reductions were 5 and 26% respectively, the total number of coolant loops was reduced from 22 to 5, and the reactor outlet temperature was reduced approximately 50°F.

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VII. AEROSPACE SAFETY

A. INTRINSIC SUBCRITICALITY EXPERIMENTS

The review of the Safety Analysis Report was continued by the AEC San Francisco office. In response to their request, calculations additional to the SAR were made to establish the range of permissible reactivity insertion for various temperature coefficients, using a peak fuel temperature of 1700°F as the limiting criterion. These calculations showed that prompt critical transients could be used for temperature coefficients up to $-0.047 \text{ } \beta / ^\circ\text{F}$ and that delayed critical transients should be used for less negative coefficients. Similar calculations were performed for transients starting at temperatures above ambient. The San Francisco office of the AEC has determined that the program can be approved and they are awaiting the results of the final pre-operational checkout of the reactor.

The electrical and mechanical installation of the SNAPTRAN reactor in the SNAP Engineering Test Facility (SETF) has been completed. The pre-operational checkout of the reactor is in progress. Before the reactor was connected to the facility control system it was checked out on the maintenance console used previously at NRTS. The reactor control system was operated in steady-state, scram, and transient modes and performed satisfactorily. With the checkout of the controls completed, the interface to the nuclear and transient controls is being checked out.

The test vault of the SETF has been returned to an operational status. The shield plugs were installed and cover plates were placed over the openings and sealed. The new cable penetrations were potted with silicone rubber. The inflatable-seal door between the transfer lock and the cell was serviced and checked out. A 5,000-gal. liquid-nitrogen tank was connected to the gaseous-nitrogen system using the pressure-building coils as the evaporator. The vault atmosphere control system was checked out and preparations for the vault leakrate measurements were completed.

An operating manual for the SNAPTRAN reactor has been written and will be reviewed by the Reactors Committee of the Reactor Safeguards Review Panel. Test procedures for the experimental program have been approved,

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and the initial experiments will begin as soon as the checkout and approval requirements have been completed.

A specification for the data-recording unit has been prepared. This unit will prepare digital magnetic tapes of the transient data for processing by a computer code to calculate the parameters of the transient. Rewriting of the COMPOST code for this analysis is in progress.

B. GADOLINIUM FOIL DEVELOPMENT

Fifty feet each of Ni and Ni-2.49 wt% Gd_2O_3 foil were received and met specifications, thus demonstrating the supplier's ability to fabricate the required foil. The Ni foil from this shipment was used to make 38 sleeves for the experimental program, which were tested on SNAPTRAN fuel elements and proved to be satisfactory. As a result fabrication of the enriched Gd sleeves may begin.

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