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SNAP REACTOR,
SNAP PROGRAM
M-3679 (61st Ed.)

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SNAP REACTOR PROGRAMS

PROGRESS REPORT

MAY - JULY 1969

(Title Unclassified)

By

J. G. ASQUITH

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The previous two reports in this series were AI-AEC-12773, "SNAP Reactor Programs Progress Report, November 1968-January 1969," and AI-AEC-12819, "SNAP Reactor Programs Progress Report, February-April 1969."

I. INTRODUCTION

A. PROGRAM SCOPE AND OBJECTIVES

The SNAP Program activities covered in this report include the following major tasks:

- 1) SNAP 8 developmental reactor (S8DR) test
- 2) Reference zirconium hydride reactor engineering
- 3) Reference reactor - thermoelectric system engineering
- 4) Compact thermoelectric converter development
- 5) Component development
- 6) Fuel development
- 7) Aerospace nuclear safety

The S8DR reactor test effort includes all activities necessary for the installation, checkout, operation, and test support of the S8DR. GFY 1969 objectives include reactor installation, zero-power testing, initial-performance mapping, simulating power conversion system (PCS) power operation, 1-Mwt operation, and the start of endurance testing at the 600-kwt 1300°F design point.

The reference ZrH reactor engineering effort provides preliminary engineering for a reference space reactor capable of a broad range of performance, including operation at 600 kwt 1300°F for a minimum of 20,000 hr. The design must also be suitable for use with thermoelectric, mercury-Rankine, Brayton, or organic cycle power-conversion systems, and must be capable of operation in either a shadow-shielded or a 4 π -shielded configuration. The preliminary design of the reference reactor subsystem is to be completed by the end of GFY 1969.

The reference reactor - thermoelectric system engineering effort includes preliminary system definition and related application and integration studies. System conceptual design is scheduled for completion by the end of GFY 1969.

The compact thermoelectric converter technology program is being conducted by the Astronuclear Laboratory of Westinghouse Electric Corporation under subcontract to Atomics International. This program is intended to provide the thermoelectric power-conversion subsystem for use with a ZrH reactor

thermoelectric space power system. The objectives of the activities of this program for GFY 1969 include: the analysis, design, fabrication, development, and test of thermoelectric tubular modules and converter assemblies; engineering and supporting activities for integration of the converter into the overall reactor thermoelectric system and definition of the resulting converter design; and adaptation of the basic thermoelectric technology to a low-voltage high-current thermoelectric module to be used for powering a dc conduction-type pump.

The component development tasks include the development and testing of reflector components, shielding, and liquid-metal components in support of the S8DR, the reference ZrH reactor, and the reactor - thermoelectric system development programs.

The fuel-development tasks provide for basic understanding, substantiation, and improvements in the technology of ZrH-type fuel-moderator elements in support of the S8DR and reference ZrH reactor programs.

The GFY 1969 aerospace nuclear safety program work includes the development of basic experimental information for intrinsic water subcriticality and analytical support needed for incorporating this engineering safeguard into the SNAP system design. The critical experiments will be made on a ZrH reactor core, completed, and analyzed during the fiscal year.

The progress made during the past quarter in these activities is presented in the order in which the activities are listed above.

B. SUMMARY OF PROGRESS

Operation of the S8DR long-term endurance run at 600 kw_t continued throughout the quarter. An increase in fission-product activity level was noted in both the coolant and surge-tank cover-gas. A summary of operation through the end of the period is given below:

Reactor Critical	4,121 hr
Reactor at Power (>525 kw _t)	5,345 hr
Energy Release	2,453,800 kw _h

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Evaluation of the effects due to changing the ZrH reference design reactor burnable poison to a Sm^{149} - Eu^{151} mixture is underway. This change will eliminate possible undesirable temperature-coefficient effects.

The system analysis effort aimed at refining the system component and subsystem weights was continued. Refinements providing a prestart NaK-pumping capability and redistributing radiator panel interconnections to equalize insulation were incorporated.

The thermoelectric-converter-module (four-panel) SN-1 design was completed and is being reviewed for fabrication approval. Tubular modules (TEM-13G) for the converter module SN-1 are on pre-installation checkout test and are producing rated power. Nine prototype 5.25-volt tubular modules are on test in the temperature range between 1025 and 1225°F. A pump power module has passed 6000 hr of operation at 1125°F hot-side average temperature with essentially no degradation.

The prototype high-temperature control-drum bearing is performing satisfactorily after 3500 hr at 10^{-8} torr and 1500°F. A developmental poison-backed control-drum assembly has been fabricated and is being prepared for thermal and mechanical testing.

Twelve Incoloy-800-clad developmental fuel elements have accumulated 5000 hr of thermal-endurance testing at 1400°F. Performance is satisfactory except for four elements which were inadvertently over-temperated due to thermal cycling.

Examination of the NAA-120-4 is continuing. Dimensional and density measurements of the central fuel sleeve and support rod have been completed. These tend to confirm that the fuel growth is essentially isotropic.

The NAA-121-1 experiment (four S8DR-type fuel elements) has been discharged from the test reactor.

II. SNAP 8 DEVELOPMENTAL REACTOR

A. INTRODUCTION

The long-term endurance run at 600 kwt was continued throughout the report period. Following a reactor scram on June 1, 1969, operation resumed with an outlet temperature of 1200°F rather than 300.

A summary of the S8DR operating statistics up to July 31, 1969 is as follows:

Reactor critical	4,121.5 hr
Reactor at power (>525 kwt, >1280°F)	1,670 hr
Reactor at power (>525 kwt, >1000°F)	3,675.5 hr
Energy release	2,453,800 kwh

B. OPERATIONS

Following completion of the 1-Mwt and 1150°F-outlet-temperature operation the reactor was returned to 600 kwt and 1300°F.* The reactor power was reduced from 1 Mwt in a stepwise manner (870 and 730 kwt) similar to the increase. During the 730-kwt dwell a reactor scram occurred. The reactor was returned to the 600-kwt and 1300°F operating condition following the scram.

Reactor operation at 600 kwt, 1300°F was continued until June 1, 1969, when a reactor scram occurred. The reactor was then returned to 600 kwt, 1200°F outlet temperature and has remained at these operating conditions except for two brief scram interruptions since that time.

A summary of scrams experienced by the reactor recently is given in Table II-1. As can be seen from the table none of the scrams was set off by reactor performance conditions. Procedures have been established which will reduce the number of scrams caused by failure of the primary pump power supply.

*See previous progress report, AI-AEC-12819, for discussion of 1-Mwt operation.

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TABLE II-1
S8DR SCRAM SUMMARY FOR REPORT PERIOD

Date	Cause	Remarks
4/29/69	High core-outlet Temperature	Instrument malfunction
6/1/69	Low flow	Primary-pump power-supply fuse blew.
6/24/69	Low flow	Instrument accidentally bumped while maintenance was being performed.
7/9/69	Low flow	Primary-pump power-supply failure.

C. FISSION-PRODUCT MONITORING

The extensive and close monitoring of fission products in the NaK coolant and the primary surge-tank cover gas was continued throughout the report period. This monitoring program provides data on the integrity of the fuel-element cladding during reactor operations. Included in the monitoring program are:

- 1) Radiochemical analysis of periodic samples of the primary-system coolant for Cs^{137} .
- 2) Radiochemical analysis of periodic samples of the surge-tank cover gas for Xe^{133} and Xe^{135} , and
- 3) On-line monitoring of the surge-tank cover-gas gross gamma activity and xenon activity.

The frequency of obtaining cover-gas samples for radiochemical analysis was reduced from weekly to approximately once per month. The weekly sampling greatly perturbed the on-line monitoring capability because of the 3- to 4-day recovery time for Xe^{135} to reach equilibrium. It was found to be more important to have unperturbed on-line data rather than frequent radiochemical analysis. The frequency of NaK-coolant sampling has been increased to provide an independent means of checking on the on-line monitor.

An increase in fission-product activity level in both the NaK coolant and surge-tank cover gas was observed in both the radiochemical analysis (Figure II-1) and the on-line monitoring (Figure II-2). The Cs^{137} shows an increase

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Figure II-1. S8DR Fraction of Core Fission-Product Inventory

at about 25 in the fraction of core fission-product inventory released to the coolant stream from April to late June. From early May to Mid-July the Xe^{135} activity showed an increase of approximately 10, and the Xe^{133} data a factor of about 20. Both Xe^{135} and Xe^{133} give increases less than that of Cs^{137} because of the effect of holdup in the fuel and coolant. Both of these isotopes have relatively short mean lives when compared to Cs^{137} , and would be expected to be affected by holdup.

D. EVALUATION AND CONCLUSIONS

From January through April 1969 the observed long-term reactivity behavior of S8DR agreed quite well with the predicted reactivity behavior. Upon returning to 600 kw, 1300°F outlet operation following the April 29 scram, the reactivity lossrate was observed to be greater than predicted (-0.6% per day observed vs -0.2% predicted). The difference between predicted and observed reactivity lossrate became progressively greater throughout May and June. As may be seen from Figure II-3, the reactivity lossrate increased to about 2.2% /day at the end of the report period.

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Figure II-2. On-Line Xe^{135}
Cover-Gas Activity

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Figure II-3. S8DR Reactivity Lossrate vs Time, End of 1-Mwt Operation to Date

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The increase in reactivity lossrate together with the increase in the fission-product activity level both indicate that some fuel elements with cladding defects exist within the reactor core. Interpretation of the data to determine the number of defective elements is difficult because of: (1) the uncertainties in the model used to describe the release of fission products from the fuel to the liquid-metal coolant; and (2) the fact that it is not possible to completely separate hydrogen lossrate effects on elements due to defective cladding and to defective barrier.

A determination of the number of defective elements from the reactivity lossrate data is even more difficult. However, it is possible to obtain some indication as to when the fuel-element degradation occurred. This is done by empirically obtaining a hydrogen-barrier defect constant which matches the analytically determined lossrate data to the experimentally observed data. The defect constant is used as a means of describing the quality of the hydrogen barrier in the equation used to analytically calculate reactivity lossrate. The results of this analysis are shown in Figure II-4. From this it can be concluded that the two reactor scrams experienced in June contributed to the degradation

Figure II-4. Whole-Core K_D Multiplier Which Matches Observed S8DR Behavior

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of the fuel-element cladding integrity. Some degradation was also experienced during the April 29 scram because the defect constant increased from a "new" core value of about 1 before the scram to about 3 immediately afterwards. Generally little increase in degradation has been observed during periods between scrams. It must also be pointed out that because of simplifying assumptions made in the model the magnitude of the change in hydrogen-barrier defect constant is not a good quantitative measure of element degradation, but merely gives an indication of changes related to reactor scrams.

Close monitoring and analysis of all experimental data is continuing to further the understanding of the extent of fuel-element degradation. This analysis is now being focused on a detailed understanding of all operational data so that it may be interpreted properly during the postoperation fuel examination.

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III. REFERENCE ZIRCONIUM HYDRIDE REACTOR ENGINEERING

A. REACTOR DESIGN

A review of the preliminary design of the reference ZrH reactor was made. As a result of this review three areas are being additionally evaluated to establish methods for incorporating design improvements. These are:

- 1) Effects of changing the reference burnable poison from Gd¹⁵⁵ to a mixture of Sm¹⁴⁹ and Eu¹⁵¹.
- 2) Fuel-element and core geometries designed to prevent the occurrence of system oscillations as were observed in the S8DR.
- 3) Control-drum modifications aimed at reducing the weight and drum imbalance.

Additional design possibilities are being evaluated to determine if improvements are feasible. These include: alternate inlet nozzle designs for reducing the inlet pressure drop; emissivity coatings in the interior of the drywells to reduce control-drum temperatures; and replacing the flexible coupling in the control drive train with a torque-arm coupling.

B. NUCLEAR ANALYSIS

A number of candidate burnable poison materials have been examined to determine an optimum prepoison selection (material and loading) for the 600-kwt 20,000-hr reference reactor design. The optimum loading is defined as the one which will provide the maximum lifetime while maintaining adequate cold shut-down capability. Initial calculations indicated an optimum cross-section (effective one-group) of 400 to 600 barns. While Gd¹⁵⁵ has an effective cross-section in this range, its adverse effect on the negative fuel-temperature coefficient makes its use questionable. It was found that a loading of \$6.00 of a 270-barn poison, plus \$3.00 of a 1700-barn poison, provided a nearly optimum two-poison system. The larger cross-section corresponds to Sm¹⁴⁹ and the lower cross-section material corresponds to the first estimate for Eu¹⁵¹. The low cross-section of the Eu¹⁵¹ results in a greater residual poison than with Gd¹⁵⁵ at the end-of-life (EOL). A slight reduction in the reactivity lifetime is expected.

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FIXED ENDS
(A)

WIRE WRAP
(B)

INTEGRAL FINS
(C)

SLOTTED TUBE
(D)

TWISTED RIBBON
(E)

COOKIE CUTTER
(F)

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Figure III-1. Fuel-Element Spacer Concepts

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A study was made to determine the effect of changes in drum composition on control-drum worth. A design modification which consisted of holes drilled in the reference poison material (Ta-10W) and filled with a lower density poison was evaluated. It is desirable to minimize drum weight without compromising drum worth.

For the study a hole fraction of 0.41 was assumed, with a minimum of 100 mils Ta-10W remaining between holes to assure structural soundness, ease of fabrication, and adequate heat transfer. Where boron was included as the low-density poison, helium evolution was considered and found not to represent a major problem.

The calculations of worth were performed using one-dimensional analysis; although significant detail was given to the analytic model, high accuracy was not possible. The effects of the changes in control-drum design, including the worth of a solid Ta-10W poison reference drum, is shown in Table III-1.

TABLE III-1
WORTH CALCULATIONS

Poison Material	Worth (\$)
Ta-10W	9.00 (reference)
Natural Hf	9.20
Ta-10W/Eu ₂ O ₃	9.70
Ta-10W/B ₄ C (natural)	10.70
Ta-10W/B ₄ C (40% B ¹⁰)	11.30
Ta-10W/B ₄ C (100% B ¹⁰)	11.80

C. FUEL-ELEMENT SPACERS

A program was started to evaluate various methods for controlling the fuel-element spacing in the core assembly to minimize fuel-element circumferential thermal gradients and their associated cladding stresses, and to reduce hot-channel effects of fuel-element clustering. The use of controlled fuel-element spacing as a means of distributing the core coolant flow is also being studied.

Of candidate spacer concepts screened, the following are being considered in a trade study (see Figure III-1):

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- 1) Fixed-end fuel elements,
- 2) Wire wrap,
- 3) Ribbon wrap or integral fins,
- 4) Slotted tube in the tricuspid flow channel,
- 5) Twisted ribbon in the flow channel, and
- 6) Grid spacer (cookie cutter).

The general criteria for spacer selection are:

- 1) Minimum thermal bowing.
- 2) Low pressure drop.
- 3) Low thermal impedance.
- 4) Maintenance of clad stresses at safe level.
- 5) Ease of core assembly.
- 6) Minimum perturbation to core reactivity, and
- 7) Structural integrity compatible with system operational requirements.

Hydraulic, thermal, and stress analyses of the various spacer configurations were begun. Optimum designs of each spacer configuration are being prepared and the trade-offs studied. Pressure-drop and heat-transfer correlations for rod bundles and spacer configurations for varying spacing-to-diameter ratios are being reviewed for application in this study.

The core friction losses were calculated for the six fuel-element-spacing configurations previously cited. Table III-2 contains the friction losses in ascending order based upon a tricuspid flow channel with minimum fuel-element separation of 0.035 in. The fixed-end fuel-element configuration is equivalent to bare rods; the pressure-drop contribution of the internal type spacers is obtained by subtracting the bare rod value, as shown in the table.

Based upon the nominal primary-system pressure drop of 2.0 psi, none of the spacer types shown in Table III-2 can be ruled out upon the basis of excessive pressure drop.

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TABLE III-2
CORE FRICTION LOSSES OF SPACED FUEL
ELEMENTS FOR A 295-ELEMENT CORE
(Nominal spacing: 0.035 in.)

Type of Spacer	Element Pressure Drop (psi)	Pressure Drop Due to Spacer (psi)
Fixed Ends	0.093	0
Grid (Cookie Cutter)	0.145	0.052
Integral Fins	0.152	0.059
Wire Wrap	0.164	0.071
Twisted Ribbon	0.226	0.133
Slotted Tube	0.256	0.163

D. SHIELD ANALYSIS

A reference 4π shield design concept was selected as a basis for future analysis, design, and development. The dimensions for this shield concept are listed in Table III-3.

TABLE III-3
 4π SHIELD DESIGN DIMENSIONS

Region	Material	Thickness (in.)
First Axial Gamma Shield	Lead	6.5
First Axial Neutron Shield	Lithium Hydride	17.1
Gallery	-	24.0
Second Axial Gamma Shield	Depleted Uranium	3.1
Second Axial Neutron Shield	Lithium Hydride	15.7
Radial Gamma Shield	Tungsten	0.5
Radial Neutron Shield	Lithium Hydride	8.5

Discussions were held with Oak Ridge National Laboratory personnel to review the results of two-dimensional transport-theory (DOT Code) calculations on the reference shield. The calculations indicate a dose rate of about 0.8 mr/hr.

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ORNL concurred with AI that substituting W for the Pb shield will result in a several-hundred-pound weight penalty.

In the DOT calculation the Ta drum poison was homogenized radially, which tends to underestimate the low-energy photon and neutron leakage. This has a greater effect on the internal heat generation in the shield than on the external dose rate. ORNL is investigating the effect of streaming which results from partial drum insertion. The results of this study will be used to determine the heating in the shield.

E. SYSTEM SAFETY ANALYSIS

Power histories were generated for various potential modes of loss of active control of the reactor system. Various combinations of Sm^{149} and Eu^{151} prepoisons, along with various initial and ultimate loadings of He^3 for the tritium-helium EOL shutdown mechanism, were evaluated for their effects on the system driftdown characteristics. The LOAFER computer code was modified to accept the unique requirements of the reactor and was the principal calculational tool employed.

A typical case is one in which initial loadings are \$3.00 of Sm^{149} , \$6.00 of Eu^{151} , and \$17 ultimate He^3 . Following loss of active control after 24 months at 600 kwt, the reactor shuts down in 10 yr. Subsequent calculations showed an ultimate loading of at least \$18.50 He^3 to be necessary for the eventual attainment of subcriticality with water flooding. An initial loading of \$21 He^3 , for example, would provide the required level of core poisoning within 370 months following shutdown. Without He^3 for EOL shutdown, the reactor would still operate at 100 watts 800 yr after loss of active control.

Cases were also evaluated in which loss of coolant or flow occurs following 24-mo operation at 600 kwt. Heat-transfer calculations indicate that hydrogen loss would provide sufficient cooling to maintain the integrity of the fuel elements and that reactor power would rapidly drop to approximately 10 kwt, from which point shutdown would occur in less than two yr.

Table III-4 gives the principal shutdown characteristics of the various cases involved.

TABLE III-4
SHUTDOWN CHARACTERISTICS

Total Prepoison Loading (\$)	Helium-3 Loading (\$)		Loss of Active Control Following 24 months at 600 kw	Loss of Active Control Due to Depletion of Excess Reactivity (All Drums in Most- Reactive Position)		Loss of Active Control With Loss of Coolant or Coolant Flow Following 24 months at 600 kw
	Initial	Ultimate	Time to Shutdown (mo)	Time at Power (mo)	Time to Shutdown (mo)	Time to Shutdown (mo)
0	0	0	(100 watts at 9000 mo)	26	(100 watts at 9000 mo)	(100 watts at 8500 mo)
0	0.33	15.00	70	20	70	12
0	1.60	15.00	75	22	80	12
0	0.37	17.00	65	18	60	12
0	1.81	17.00	70	20	70	12
9.00	0	0	(100 watts at 10,000 mo)	78	(100 watts at 10,000 mo)	(100 watts at 9500 mo)
9.00	0.33	15.00	110	60	85	20
9.00	1.60	15.00	115	54	90	20
9.00	0.37	17.00	105	54	85	20
9.00	1.81	17.00	110	50	90	20

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IV. REACTOR - THERMOELECTRIC SYSTEM ENGINEERING

A. GENERAL

Design refinements were made for the reference 25-kwe power system. The new system weights resulting from these refinements are shown in Table IV-1.

TABLE IV-1
REFERENCE 25-kwe ZIRCONIUM HYDRIDE
REACTOR - THERMOELECTRIC POWER
SYSTEM WEIGHT SUMMARY

	Weight (lb)
Reactor	1,830
Thermoelectric converter	1,350
Pumps	410
NaK accumulators	210
Piping	950
Radiator	1,430
Structure	1,210
Electrical wiring	830
Shielding	14,010
Total	22,230

A method of providing NaK flow during prestart operating periods was selected. The selected method provides electrical current to taps on the main pump. During the prestart period sufficient current is supplied across the heat-rejection pump throat to provide 5% flow, while current shunting through the pump converter to the primary-loop throat provides 2% flow in the reverse direction. The current is switched to a tap on the primary-loop throat during the approach-to-sensible-heat position of the startup period and the situation is reversed. A total of 330 watts is required at 1.2 volts.

A change in the heat-rejection loop piping was made to improve the heat distribution during prestart operation. With the change, half the coolant in each

loop quadrant is cooled by radiator panels at the top of the radiator and the other half is cooled by panels on the opposite side at the bottom. Thus each quadrant receives an equal amount of solar energy.

Calculations were made to determine the end-of-life power decay of the power system. The calculation determined the electrical power output characteristics after the control drums are fully rotated, but insufficient reactivity is available for full power. These results are given in Figure IV-1. It can be noted that more than one year is required before the electrical power is reduced by a factor of two (25 to 12.5 kwe).

B. SYSTEM SAFETY STUDIES

System safety analyses have been made to analytically verify the critical assumptions used in each analysis. Specifically an evaluation of the mechanical and thermal forces and vehicle configuration during re-entry of the reference 4π shielded reactor concluded that re-entry dispersion of fission products is highly unlikely. Based on the impact velocities derived in this study, an analysis of the survival capability of the reactor core vessel (i.e. ability of the

Figure IV-1. End-of-Life Power Decay

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vessel to contain the fuel elements) was made. Based on an energy-dissipation analysis it was concluded that intact survival was probable in mediums ranging from soil to water. The survival capability following hard rock or concrete impact requires additional evaluation.

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V. THERMOELECTRIC CONVERTER DEVELOPMENT (Prepared by Westinghouse Astronuclear Laboratory)

A. REACTOR TUBULAR POWER MODULES

1. Performance of Preprototype Modules

The first four tubular modules fabricated and tested at simulated two-loop reactor system temperatures of $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$ were designated as preprototype modules TEM-9U SN-7, -8, -9, and -10. They were fabricated with dimensions essentially identical to a group developed under the Compact Converter Tubular Module Technology Program. However, these were the first to use solid tungsten inner conductor rings designed to eliminate an iron/lead-telluride reaction occurring at the couple hot junctions. This feature was incorporated to enhance the operational stability at temperatures above $\bar{T}_H = 1000^\circ\text{F}$.

During the past quarter one of the modules, TEM-9U SN-10, completed 5166 hr of testing in a two-loop liquid-metal system at Atomic International. Performance trends for this remained unchanged from those given in previous reports. Electrical-power-output data for TEM-9U SN-10 along with data from SN-7, -8, and -9 are given in Figure V-1. The figure shows that the thermal-cycling tests made on SN-9 and -10 had essentially no effect on subsequent performance for either module. SN-10 was operated at two-loop reactor system temperatures at WANL for 1989 hr before being shipped to AI. Life testing of this module has been completed.

Although test data for the four indicate a higher level of operational stability than that exhibited by any previously built module incorporating TEGS-2P PbTe and tested at temperatures above 1000°F , additional work has been done to further reduce degradation. The success of TEM-9U SN-19, which will be discussed below, attests to this fact. No adverse effects as a result of any thermal-cycle testing have been observed in any preprototype reactor modules.

2. First-Generation Prototype Modules

Eight reactor power prototype modules were tested in static test stands at WANL during the past quarter. TEM-13F SN-3 completed 500 hr of static testing at WANL and was shipped to AI where further testing was carried out in

Figure V-1. Electrical Power Output of Preprototype Reactor Modules

Figure V-2. Comparison of Electrical Power Output From Solid Conductor Ring Prototype Modules

a liquid-metal loop. In addition, thermal-cycling tests were made on four modules. All modules were endurance-tested at average clad temperatures simulating the nominal temperatures of the two-loop reactor system, $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$.

a. Solid Tungsten Conductor-Ring Modules

Power-output data recorded from TEM-13A SN-1 and -2, the first two reactor power modules fabricated and tested under the Compact Converter System Technology Program, are shown in Figure V-2. Similar data from TEM-13B SN-2 and TEM-13C SN-2 are also shown. It should be emphasized that all of these modules were fabricated before the development of the two-piece inner conductor ring. Initial degradation rates observed in these modules are assumed to be primarily controlled by the contamination of N-type material by P-type as a result of PbTe diffusion through and around the mica-insulator/inner-conductor-ring butt-joint.

Initial degradation rates of the TEM-13B and TEM-13C modules are substantially higher than those of the TEM-13A series, as shown in Figure V-2. This relationship is expected since the TEM-13B and TEM-13C series modules were fabricated with thinner thermoelectric (TE) washers than the TEM-13A series (0.050 and 0.033 in. vs 0.090, respectively).

Also shown in Figure V-2 are the data taken after a series of ten thermal cycles on TEM-13A SN-2, TEM-13B SN-2, and TEM-13C SN-2. All of these thermal-cyclic tests were performed in accordance with the test plans for these modules ($20^\circ\text{F}/\text{min}$ ramp rates). Although in each case reductions in power occurred after thermal cycling, it is difficult to separate the effects of thermal cycling from those of degradation caused by the materials interactions in these modules. TEM-13B SN-2, which has been removed from static test, will be destructively examined to verify the degradation mode present in these solid W conductor-ring modules, as well as to determine the effects of thermal cycling on their various internal components. An additional one thermally cycled during the past quarter, TEM-13D SN-2, affords a much better evaluation of the effects of thermal cycling because its rate of degradation is much lower than those of previous ones.

b. Modules Incorporating Stepped Inner Conductor Rings

Four prototype power modules with two-piece (stepped) inner conductor rings were tested during the past quarter. The two-piece conductor ring was designed to entrap the large mica insulator eliminating the possibility of assembly gaps that allow contact on adjacent PbTe washers. As a results of the design improvement, reduced degradation rates have been achieved.

Figure V-3 shows adjusted power-output data from TEM-13D SN-2 and three TEM-13F-series modules. For each data point the experimentally measured power-output data have been corrected to account for minor deviations in experimentally recorded operating temperatures from the design operating temperatures, $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$. This adjustment is made to remove the effects of minor temperature fluctuations on experimental data which, in an analysis of data from modules with short testing times, could mask the effects of degradation.

Figure V-3. Comparison of Electrical Power Output From Stepped Conductor Ring Prototype Modules

TABLE V-1
SUMMARY OF TEM-13G SERIES MODULES

Module, SN	Purpose	Status
-1	Obtain degradation-rate data at design reactor condition.	On static test at indicated design conditions. No measurable degradation after 583 hr of testing.
-2	Obtain performance data in a two-loop reactor system environment.	Static-tested for 415 hr at WANL, installed in shroud assembly and shipped to AI for subsequent testing.
-3	Determine cyclic test capabilities during static testing at WANL.	On static test for 773 hr at WANL. Cyclic tests scheduled after 2000 hr of steady-state testing.
-4	Obtain degradation-rate data at reduced operating temperature levels.	On static test for 744 hr with no measurable degradation.
-5	Obtain degradation-rate data at elevated operating temperature levels.	On test, no measurable degradation.
-6 through -9	To be installed in Power-Converter Module SN-1 and tested in liquid-metal loop at AI.	All modules on static test for evaluation at WANL.

TABLE V-2
TEM-13G BEGINNING-OF-LIFE DATA AT 1025°F/538°F,
SN-1 THROUGH -4

Module	$\bar{T}_H$ (°F)	$\bar{T}_C$ (°F)	Total Power Input (watts)	Open- Circuit Voltage (volts)	Circuit Resistance (milliohm)	Power Output (watts)	Overall Efficiency (%)	Load Voltage (volts)	Load Current (amp)	Load Resistance (milliohm)
Calculated	1025	538	4865	9.14	93.8	223	4.58	4.57	48.7	93.8
TEM-13G-SN-1	1025	544	5153	9.51	97.3	232	4.50	4.83	48.1	100.4
-2	1027	542	5151	9.52	95.3	238	4.61	4.70	50.6	92.8
-3	1024	538	5037	9.22	95.8	222	4.42	4.68	47.5	98.6
-4	1035	540	5338	9.70	94.0	250	4.69	4.82	51.9	92.8

TEM-13D SN-2 incorporates TEGS-3P material whereas TEM-13F-series modules use TEGS-2P. All of these were fabricated with 0.033-in. PbTe washers (identical in size to those of the TEM-13C series discussed above) and 0.003-in. mica insulators. Figure V-3 exhibits the decrease in beginning-of-life (BOL) power level (approximately 20%) resulting from the use of TEGS-3P PbTe in TEM-13D-series modules.

The advancements made in the reduction of degradation rates during the past year are evident in a comparison of TEM-13F-series test data, shown in Figure V-3, with similar data from TEM-13C SN-2 series modules shown in Figure V-2. All of these modules incorporate identical-sized PbTe washers. The TEM-13F series, however, were fabricated with 0.003-in.-thick mica insulators, whereas those of the TEM-13C series were 0.0015 in. In addition the stepped-conductor-ring concept was used in the TEM-13F series. Although TEM-13F modules do not exhibit the low degradation rates projected for future ones, orders of magnitude reduction in initial degradation rates resulted from the incorporation of two-piece (stepped) inner conductor rings into the design.

A series of ten thermal cycles was completed on TEM-13D SN-2 during May. The time at which the cyclic test was conducted is indicated in Figure V-3.

3. First-Generation 5.25-Volt Prototype Modules

Fabrication of nine TEM-13G-series prototype modules was completed during the past quarter. (Design concepts and details of this series were given in WANL-PR(EEE)-010.) The purpose and present status of each of these "first-generation prototype reactor modules" are given in Table V-1. Each has been placed on static test at WANL during the past quarter and, with the exception of SN-2, was still on test at the writing of this report. TEM-13G SN-2 has been installed in a shroud assembly and shipped to AI for testing in a two-loop liquid-metal system.

To establish the level of performance reproducibility, each of the nine modules was operated at nominal indicated clad temperatures $\bar{T}_H = 1025^\circ\text{F}$ and $\bar{T}_C = 538^\circ\text{F}$ long enough to obtain stable performance data. Table V-2 gives the recorded performance parameters for TEM-13G SN-1 through -4 at these operating conditions, as well as calculated parameters for operation at these temperatures. The TEM-13G modules are being tested with single-piece electrical

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heaters. The heater is approximately 0.75 in. longer than the circuit. The heat input listed in Table V-2 is the total electrical heat input to each module.

The spread in the power-output data which ranges from 222 to 250 watts can be seen to correspond with the spread in heat-input data. The lowest output power, 222 watts from TEM-13G SN-3, is very close to that calculated for the indicated operating temperatures. Operation of a group of identical TE modules at a given heat-input level implies that each should have the same radial-temperature drop. This is true because thermal impedance (defined as the ratio of the module radial-temperature drop to the circuit heat-input rate) is essentially constant over a wide range of temperature conditions near the design operating point. Over sets of operating conditions varying by more than 200°F and 80°F respectively from the design average hot- and cold-clad operating conditions, circuit thermal-impedance data vary by less than 2.1% from the data recorded at test conditions. This phenomenon has been observed in all modules tested at WANL and is predicted by TEMOD, the analytical module performance computer code.

A series of tests were performed on TEM-13G SN-1 through -4 to determine if module-to-module performance variations were actually caused by inaccuracies in determining the average clad temperatures. SN-1 and -2 data were used as a baseline for these experiments because of the reproducibility of power input, power output, and temperatures. BOL conditions for SN-1 through -4 are re-compiled in Table V-3.

In the first portion of the experiment, heater levels of TEM-13G SN-3 and -4 were set at about the level of SN-1 and -2. The indicated average hot-clad temperature levels were also adjusted, as close as experimentally possible, to the TEM-13G SN-1, -2 level (1026°F). The data set recorded for each module after achieving equilibrium conditions is listed in Section II of Table V-3. With the exception of the indicated average cold-clad temperature, which was adjusted to achieve the input-power and hot-clad temperature conditions, all parameters agree very well. In particular, the power-output data from all three modules agree to within 3 watts of the norm (235 watts).

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TABLE V-3
RESULTS OF OPERATING-CONDITION PERTURBATION TESTS
ON TEM-13G, SN-3 AND -4

	TEM-13G	Heat Input (watts)	$\bar{T}_H$ (°F)	$\bar{T}_C$ (°F)	Open- Circuit Voltage (volts)	Circuit Resistance (milliohm)	Power Output (watts)
I. Beginning-of-Life Conditions	SN-1	5153	1025	544	9.51	97.25	232
	-2	5151	1027	542	9.52	95.3	238
	-3	5037	1024	538	9.22	95.8	222
	-4	5338	1035	540	9.70	94.0	250
II. Q_{IN} , $\bar{T}_H$ Adjusted to TEM-13G SN-1, -2	SN-1	5153	1025	544	9.51	97.25	232
	-2	5151	1027	542	9.52	95.3	238
	-3	5173	1022	522	9.40	95.5	232
	-4	5150	1026	550	9.55	97.8	233
III. Q_{IN} , $\bar{T}_C$ Adjusted to TEM-13G SN-1, -2	SN-1	5153	1025	544	9.51	97.25	232
	-2	5151	1027	542	9.52	95.3	238
	-3	5164	1045	541	9.59	100.0	232
	-4	5132	1018	544	9.46	95.9	233

In the next phase of the experiment the power-input levels of SN-3 and -4 were maintained at the 5150-watt level and the indicated average cold-clad temperatures were adjusted as close as experimentally possible to the baseline level of 543°F. Experimental data recorded under these operating conditions are listed in Section III of Table V-3. Once again the correlation of experimental parameters was significantly better than that achieved under the initial operating conditions. Power-output data for all modules was 235 ± 3 watts.

Electrical-power output is a strong function of module radial-temperature difference and, as in the case of thermal impedance, is a weak function of average module temperature near design operating conditions. For this reason identical modules operating with the same radial-temperature drop should produce nearly equal electrical-power outputs. Based on this reasoning it is thought that those tested as shown in Table V-3 should have had reproducible radial-temperature differences. That is, with constant power input, constant hot-clad

temperatures should have produced similar cold-clad temperatures and vice-versa. On the basis of the data shown in Sections II and III of Table V-3, it is concluded that the basic cause for original module-to-module performance nonreproducibility was a result of inaccurate average temperature determinations. Accordingly new hot- and cold-side thermocouple harnesses containing 21 thermocouples on each side were designed and used in the testing of all subsequent TEM-13G modules.

As seen in Table V-4 a much higher degree of module-to-module performance reproducibility has been achieved in the TEM-13G modules using the new thermocouple harnesses. In addition, a much better correlation between calculated and experimental results has been achieved.

No degradation has been observed in any of the modules during the limited testing to date.

B. PUMP TUBULAR POWER MODULES

Three tubular modules designed to deliver power to an electromagnetic pump are on test. These prototype pump modules, designated TEM-14 SN-1, -2, and -3, are being tested at average clad temperatures simulating those of the zirconium hydride two-loop reactor system, $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$.

TABLE V-4
TEM-13G BEGINNING-OF-LIFE DATA AT $1025^\circ\text{F}/538^\circ\text{F}$,
SN-5 THROUGH -9

Module	$\bar{T}_H$ ($^\circ\text{F}$)	$\bar{T}_C$ ($^\circ\text{F}$)	Total Power Input (watts)	Open- Circuit Voltage (volts)	Circuit Resistance (milliohm)	Power Output (watts)	Overall Efficiency (%)	Load Voltage (volts)	Load Current (amp)	Load Resistance (milliohm)
Calculated	1025	538	4865	9.14	93.8	223	4.58	4.57	48.7	93.8
TEM-13G-SN-5	1026	538	4898	9.26	95.9	224	4.57	4.62	48.4	95.3
-6	1027	536	4967	9.212	94.6	224	4.51	4.64	48.3	96.0
-7	1027	536	4889	9.085	94.6	218	4.48	4.55	47.9	95.7
-8	1025	539	4896	9.06	92.1	223	4.57	4.51	49.4	91.4
-9	1027	537	4891	9.09	96.2	215	4.41	4.57	47.0	97.2

Electrical-current and electrical-power-output data recorded from the TEM-14A series modules as functions of time are shown in Figures V-4 and -5, respectively. The electrical-current data from SN-2 and -3 have been adjusted to account for minor deviations in the load resistor from the design load of 0.44 milliohms. The BOL data from both SN-2 and -3, mathematically adjusted to the design load resistance, agree to within 1% of the design levels. SN-1 performance parameters are somewhat lower than the design levels because the electrical resistance of the Ni power pins is larger than calculated in the original design study, which assumed Mo pins. The power-pin electrical resistance of SN-2 and -3, however, was reduced by incorporating twelve rather than six power pins at each end.

Excellent performance stability has been realized in the TEM-14-series pump modules. No measurable degradation of performance has been observed for TEM-14A SN-1 during the first 6002 hr of operation at design operating conditions, nor on either SN-2 or -3 after 2390 and 444 hr of testing, respectively.

C. POWER-CONVERTER MODULE (PCM)

1. PCM Metal Mockup

Fabrication of a full-scale converter module mockup (Figure V-6) was completed during the past quarter. This unit is similar in design to PCM SN-1 which is to be delivered for liquid-metal-loop testing later this year.

The objective of this effort was to establish design feasibility, from the fabrication point-of-view, and gain experience in welding and inspection operations. Figure V-7 shows individual component parts of the metal mockup.

2. Hydraulic Test Program

The hydraulic testing of full-scale models of the converter module is complete and partial results were reported previously.* During this reporting period analysis of the data generated by these tests was completed and the flow distribution within a converter module model was investigated.

*"Compact Converter System Technology, Quarterly Progress Report," WANL-PR(EEE)-010, April 1969

Figure V-4. Electrical-Current Data From Three Prototype Pump Power Modules

Figure V-5. Power-Output Data From Three Prototype Pump Power Modules

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Figure V-6. Converter Module Metal Mockup

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Figure V-7. Heat-Rejection Manifold and Converter Module Subassembly

To determine if flow variations exist among the four parallel flowpaths, a special series of hydraulic tests was conducted to measure flow distribution. The tests were conducted with water flowing through a full-scale plastic mockup of the primary- and secondary-flow channels and associated manifolds. Flow-rates were measured in each of the four primary and secondary channels and the ratios of the highest to the lowest were used to compute the flow distribution factor, defined as follows:

$$\text{Flow distribution (percent)} = \left[\frac{\text{Maximum Flow}}{\text{Minimum Flow}} - 1.0 \right] \times 100$$

The distribution factor was plotted versus the mean Reynolds Number of the flow passage, as shown in Figure V-8.

The module-to-module flow distribution for the primary side is on the order of 1 to 2% in the normal operating range of the module, while the secondary flow distribution varies by approximately 8%. The secondary flow drops into the laminar regime at about 50% of design flow. When this occurs an

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Figure V-8. Converter Module Flow Distribution
Based on Water-Flow Tests

abrupt change in flow distribution develops, with the flow mismatch increasing with decreasing flowrate. As the transition regime is traversed, the location of the peak flow channel also changes abruptly with minor flow variations.

Figure V-9 shows the results of the flow distribution tests scaled to NaK-78 design-flow conditions for both the primary and secondary flow channels. The percentage of the total flowrate (assumed to be 0.52 lb/sec in each loop) is shown in each tubular module channel.

An analysis of the influence that the measured distribution of flow might be expected to have on module performance was completed using the TEFLOW digital computer code. The TEM-13G tubular module design was used and four operating conditions studied, corresponding to the four pairs of off-design flowrates noted in Figure V-9. The results of the study are summarized in Table V-5. The TE performance of each off-design module (numbered 1 through 4 corresponding to the Modules in Figure V-9) is indicated. Also the net performance of the four tubular modules connected electrically in series

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Figure V-9. Predicted Converter Module NaK-Flow Distribution
Scaled From Water-Flow Tests

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TABLE V-5

SUMMARY OF EFFECTS OF MEASURED-FLOW VARIATIONS ON TEM-13G PERFORMANCE
(Primary-Inlet Temperature = 1245°F, Secondary-Inlet = 460°F)

Parameter	Off-Design Module No.				Four-Pack	
	1	2	3	4	Reference	With Flow Variations
Primary Flowrate, lb/sec	0.1290	0.1297	0.1308	0.1305	0.52	0.52
Secondary Flowrate, lb/sec	0.13	0.1243	0.1316	0.1342	0.52	0.52
Electrical-Power Output, watts	263.7	260.2	265.3	266.8	1056	1056
Matched-Load Voltage, volts	5.34	5.31	5.36	5.38	21.40	21.39
Heat Throughput, watts	5466	5446	5485	5494	21,892	21,891
Circuit Efficiency, %	4.82	4.78	4.84	4.86	4.82	4.82
Primary-NaK ΔT , °F	199.6	197.7	197.5	198.2	198.3	198.25
Secondary-NaK ΔT , °F	185.7	193.7	184.1	180.7	186.0	185.9
Mean Primary-Clad Temperature, °F	1126	1128	1128	1127	1127	1127.2
Mean Secondary-Clad Temperature, °F	556	560	555	554	556	556.2

Figure V-10. TEM-13G Power/
Efficiency/NaK-Temperature
Performance Map

Figure V-11. TEM-13G Power/Load-
Voltage/NaK-Temperature
Performance Map

Figure V-12. TEM-13G
Power/Efficiency NaK-
Temperature Per-
formance Map

Figure V-13. TEM-13G Predicted
Axial NaK-Temperature
Performance Map

is tabulated and is compared to the equivalent performance of a converter module with perfect flow distribution (defined in Table V-5 as the Reference Four-Pack). It is concluded that the change in performance which can be expected from flow maldistribution will be negligible under normal operating conditions. Note that the Reference Four-Pack calculation is concerned with the influence of flow maldistribution on module performance, and as such does not reflect actual operating temperatures.

3. Systems-Oriented Performance Analysis

The performance analysis of converter modules with the TEM-13G tubular module design was continued during the reporting period. In WANL-PR(EEE)-010 a set of performance maps illustrated the performance of a single tubular module for a fixed set of inlet-NaK temperatures while the primary and secondary flowrates were varied over a useful range. This effort was extended to cover the complementary situation where the flowrates are fixed and the temperatures permitted to vary.

The primary- and secondary-loop flowrates were each fixed at 0.13 pps, a convenient value quite close to the nominal design values. The primary-NaK-inlet temperature was varied from 1045 to 1345°F in steps of 100°F, and the secondary-inlet temperature was varied from 400 to 500°F in steps of 25°F. Figure V-10 is the predicted power/efficiency/temperature map for the TEM-13G module. Each operating point on the map is a matched-load point. Three additional maps were generated. Figure V-11 shows the power/temperature/load-voltage map. Figure V-12 gives the mean clad surface temperatures on the primary and secondary sides as dependent variables and Figure V-13 illustrates the axial-temperature drop in the NaK loops as a function of inlet-NaK temperature. The latter was not quite as amenable to the four-parameter plot treatment as the other data. Due to the lack of dispersion in the data, the secondary-inlet-NaK isotherm lines were not connected.

Figure V-14 shows the radial-temperature distribution within the TEM-13G module at three axial stations. The upper curve is associated with the primary-inlet end, the middle curve with the geometric center (axially) of the module, and the lower curve with the secondary-inlet end. The boundary conditions were that the primary and secondary flowrates were both 0.13 pps and the inlet-NaK temperatures were 1245 and 460°F.

Figure V-14. Predicted TEM-13G Radial-Temperature Distribution at Nominal Design Operating Conditions

Figure V-15 shows the axial-temperature distribution within the module, over the active circuit length. The zig-zag effect is due to the coarseness of the finite element model used. Twenty axial slices were selected. Although selection of a larger number of slices de-emphasizes the effect, the overall performance results are not significantly changed and the running time for the problem increases considerably.

4. PCM SN-1 Design

The detail design and fabrication specifications of PCM SN-1, Figure V-16, were completed and submitted for approval during the past quarter. The fabrication of component parts is on schedule.

A stress analysis was performed to generate safe load limits for the attachment of the module to the NaK loop. Loads, torques, and bending moments will be generated by the expansion and contraction of the loop piping. It is estimated that the unit may safely be loaded at the four piping connection points up to loads of 50 lb and 100 in.-lb of torque, in any direction and any combination. The central support plate was assumed to be fixed at one end and constrained at the other to permit motion only in a direction perpendicular to the tubular modules and along the axis of the plate. The analysis indicates that the reaction loads on the plate must not individually exceed 50 lb in any direction. The design of the converter assembly provides for angle supports which will attach to the extremities of the converter modules, thus providing additional strengthening for the support plates. When the converter module is considered by itself it is evident that additional structural support must be provided on the ends of the support plate. The design of SN-1 has been modified slightly in this fashion to provide the required support.

D. POWER-CONVERTER ASSEMBLY

The preliminary design layout for the power-converter assembly (PCA SN-1) has been started from the conceptual design shown on Figure V-17. The design will include an improved power-converter module design, SN-2, in the converter assembly.

TEM-13G NAK INLET TEMPERATURE MAPPING

A. BULK PRIMARY FLUID	E. INNER CONDUCTOR CENTER	I. OUTER CONDUCTOR CENTER
B. INNER CLAD SURFACE	F. HOT JUNCTION	J. OUTER BORON NITRIDE RING CENTER
C. INNER CLAD CENTER	G. THERMOELECTRIC MATERIAL	K. OUTER CLAD CENTER
D. INNER BORON NITRIDE RING CENTER	H. COLD JUNCTION	L. OUTER CLAD SURFACE
		M. BULK SECONDARY FLUID

Figure V-15. Predicted TEM-13G Axial-Temperature Distribution at Nominal Design Operating Conditions

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CRACKS, FATS AND UNUSED AREA TO 10X MAGNIFICATION.

TO MAINTAIN 23.75 DIM.

[illegible]

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Figure V-17. Converter Assembly

E. SUPPORTING TECHNOLOGY DEVELOPMENT

A primary cause for degradation in tubular modules has been identified as an interdiffusion problem between the N- and P-type lead telluride. The N-type material was found to be contaminated by the P-type lead telluride with the contamination concentration near the butt-joint where the mica insulator meets the hot conductor ring. However, the P-type material experiences little or no change in modules that use solid W rings for the hot-side electrical conductor. Experiments have been planned to investigate the diffusion phenomenon and the behavior of mica under different processing and operating conditions.

1. Sublength Module Design

TEM-15 designates a series of modules which will be fabricated to evaluate the effects of processing techniques, assembly clearances, and mica insulation on performance.

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A letter suffix has been assigned to distinguish the type of insulating barriers used to separate the N- and P-type washers. TEM-15A utilizes the conventional one-piece 0.005-in. mica insulators. TEM-15B has a layer of 0.0005-in. W sandwiched between two layers of mica with total thickness of 0.005 in. TEM-15C has two mica insulators with a thin-film W coating applied to each of the 0.0025-in. -thick mica mating surfaces.

A serial number applied to the end of each of the above types identifies the fabrication process and the assembly clearance incorporated for the module. Figure V-18 depicts the matrix of modules to be investigated and identifies the modules to be fabricated initially.

The TEM-15A series is designed to investigate the influence of mica radial clearance on mica deformation during processing, the influence of weld shrinkage on circuit alignment and mica deformation, and the influence of axial preload on mica deformation and circuit alignment. As shown in Figure V-18, four TEM-15A modules have been selected for the initial evaluation.

The TEM-15B series has been selected to provide a baseline for the evaluation of insulator barriers between the N- and P-type washers. The initial performance of the 15B modules will be reduced because of the large radial-shunt heat losses through the 0.0005-in. -thick W barrier. However, the experiment

Figure V-18. TEM-15 Design Matrix

will provide a means of evaluating the magnitude of the performance improvement which can be anticipated once an acceptably designed barrier has been incorporated. A single 15B module has been selected for initial evaluation.

The barrier being developed to provide complete separation of P- and N-type materials with negligible shunt heat losses will be investigated in the TEM-15C series. These modules will be similar to TEM-15A modules except the mica insulator rings will be made of two pieces of 0.0025-in. mica with a thin-film metallic deposit on mating surfaces.

2. Module Examination

Three modules were removed from test; destructive examination was started to evaluate degradation modes.

TEM-9Q SN-3 testing was terminated after 6686 hr at $\bar{T}_H = 1150^\circ\text{F}$ (621°C), $\bar{T}_C = 392^\circ\text{F}$ (200°C). The module had PbSnTe (TEGS-3P) as the P-type washer of the TE couple, and single-phase N-type PbTe (GE-N) in the other washer. During the life test the module exhibited a decrease in power output, from 108.2 watts at BOL to 98 watts after 6686 hr at test conditions. This power change resulted from a decrease in voltage and a slight increase in resistance. Since the PbSnTe was considered a possible substitute for P-type PbTe, some understanding of what caused module degradation was desirable.

The module was sectioned transversely at its center and one-half mounted and sectioned axially. Visual examination of the TE circuit revealed that the mica washers separating the TE materials appeared to be relatively free of inclusions as compared to modules containing TEGS-2P PbTe in the P-washer. This result may be caused by the mechanical property differences that exist between the two P-type materials, i.e. the PbTe (or some material present) flows more readily into the mica during testing.

A sample was removed from the center portion of the module where the hot-side temperature was 1184°F (640°C). The section was ground axially to expose the large-diameter mica washer covering a P-type washer. The former is the one which forms a butt-joint with the inner conductor ring. The mica was stripped away to expose the immediately adjacent surface of the P-type washer and the portion of P-type material nearest the N-type washer. Seebeck

Figure V-19.
Seebeck Coefficient vs Distance
from the Inner Conductor Ring
for a P-Type Washer from
TEM-9Q SN-3, $T_H = 1184^\circ\text{F}$
(640°C)

Figure V-20.
Seebeck Coefficient vs Dis-
tance from the Inner Con-
ductor Ring for an N-Type
Washer from TEM-9U
SN-3, $T_H = 1184^\circ\text{F}$
(640°C)

probe traces were taken on this surface, starting at the inner conductor ring and proceeding in a radial direction to the outer conductor ring. Figure V-19 is a plot of the Seebeck coefficient vs distance from the inner conductor ring for the probe trace taken on the P-type washer of TEM-9Q SN-3. It should be noted that the Seebeck probe trace exhibited a phenomenon never observed on probe traces taken on module washers of P-type PbTe. Such traces either exhibit no change when W contacts are used or show a rise in the Seebeck coefficient near the hot inner conductor ring when Fe is used. On the PbSnTe washer examined the Seebeck coefficient showed an initial low value near the hot-conductor-ring/washer interface, a rise to near normal value at approximately 0.040 in. from the hot conductor ring, and a decreased Seebeck coefficient in the remaining portion of the washer. The washer was further ground until a position approximately 0.030 in. from the butt-joint mica was reached and the Seebeck probe traces were repeated. The resulting Seebeck coefficient trace was identical to that observed at the butt-joint interface. This observation seems to indicate that the Seebeck behavior of the material is a bulk property change due to long-time testing in a temperature gradient rather than a reaction with other components in the module.

The section of TEM-9Q SN-3 was then ground to expose the large mica washer adjacent to an N-type washer, and after the mica was removed a similar Seebeck probe trace was taken on the N-type surface. Figure V-20 is a plot of the Seebeck coefficient vs distance from the inner conductor ring for the N-type washer. The trace showed that the washer had been contaminated within approximately 0.150 in. from the inner conductor ring. The extent of contamination is considerably less than that observed when P-type PbTe is adjacent to the N-type washer. However, it is apparent that some interdiffusion has occurred and PbSnTe can contaminate N-type PbTe.

TEM-9U SN-15 was also removed from test after 3500 hr at $\bar{T}_H = 1037^\circ\text{F}$ (558°C), $\bar{T}_C = 393^\circ\text{F}$ (200°C). This module had been processed at a higher temperature, 1382°F (750°C), and pressure, 30,000 psi, than the normal processing conditions of 1202°F (650°C) and 20,000 psi. The positive end of the module containing approximately 3.5 in. of the TE circuit was mounted and cut axially to examine the circuit configuration. Visual examination of the sample revealed the circuit was not significantly different from other TEM-9 modules processed at the normal temperature and pressure.

Another section was removed from the center of the module where the hot-side test temperature was approximately 1067°F (575°C) and polished to expose a large mica washer covering an N-type washer. The mica was stripped away and Seebeck probe traces were taken every 90° around the circumference of the washer starting at the inner conductor ring and proceeding in a radial direction to the outer conductor ring. Figure V-21 is an average of four traces of the Seebeck coefficient vs distance from the inner conductor ring. This figure shows that the washer is contaminated to a distance approximately 0.100 in. from the inner ring and possibly converted to P-type near the inner conductor ring. The sample surface was ground away until 0.005 in. of material was removed, and the probe traces were repeated. The traces were taken at the same positions as those originally taken on the surface next to the butt-joint mica. Examination of the data showed that both the degree and radial depth of the contamination were less than before. Continuing examination of this module is in process, especially in the region of the N-washer adjacent to the small mica washer.

Figure V-21. Seebeck Coefficient vs Distance from the Inner Conductor Ring for an N-Type Washer from TEM-9U SN-15, $T_H = 1067^\circ\text{F}$ (575°C)

In other modules examined, particularly TEM-9U SN-7, the contamination level near the butt-joint mica was much greater than observed in TEM-9U SN-15. The SN-7 was tested about the same length of time (3500 hr) as SN-15. Although SN-7 was tested at a different set of temperature conditions, $\bar{T}_H = 1128^\circ\text{F}$ (607°C) and $\bar{T}_C = 572^\circ\text{F}$ (300°C), a comparison between the two modules is possible. A section was cut from SN-7 at a position where the hot-side test temperature was 1022°F (550°C), approximately 45°F (25°C) lower than that of the sample from SN-15. Seebeck probe traces taken on the sample from the SN-7 showed the N-type washer was converted to P-type for approximately one-half its radial thickness. This degree of conversion and contamination was greater than encountered in the SN-15, even though the SN-15 hot-side temperature was slightly higher. From the above data there seems to be a possibility that the higher processing parameters of the TEM-9U SN-15 were responsible for a reduced level of intercouple contamination. It should be emphasized, however, that complete elimination of interaction between couples is not likely to be achieved by processing parameter changes. Other data from capsule tests indicate that some impermeable barrier, such as a very thin layer of W or Mo, is necessary to halt contamination through the mica.

TEM-13B SN-2 was removed from test after 1187 hr at $\bar{T}_H = 1125^\circ\text{F}$ (608°C) and $\bar{T}_C = 572^\circ\text{F}$ (300°C). Before removal from test it had exhibited degradation in power output. Over its test life it was subjected to five thermal cycles due to instrumentation failures and ten controlled temperature-ramp cycles.

A section was removed from the center of the module where the hot-side temperature was approximately 1158°F (625°C) and ground to expose a large mica washer over an N-type washer, and the mica stripped away to expose the N-type PbTe surface. Seebeck probe traces were taken every 90° around the module circumference, starting at the inner conductor ring and proceeding in a radial line to the outer conductor ring.

Figure V-22 is a typical plot of the Seebeck coefficient vs distance from the inner conductor ring for the N-washer. It shows that the washer is converted to P-type over one-half its radial dimension and contaminated over

Figure V-22. Seebeck Coefficient vs Distance from the Inner Conductor Ring for an N-Type Washer from TEM-13B SN-2,
 $T_H = 1158^\circ\text{F} (625^\circ\text{C})$

almost its entire surface. The plot further indicates that transport, presumably of Te as discussed below, has affected the N-type material to a considerable extent. The depth of contamination in an axial direction has not yet been determined.

3. Mica Evaluation

Studies were begun to investigate the feasibility of using alternate electrical insulators in place of mica. At the present time only two materials were found to be possible candidates. Sheets of borate-bonded mica flakes (BBM) were considered because the softening effect at the processing temperature could conceivably eliminate mica cracking. India Ruby mica (IRM) was also considered because of its reported denser structure.

Test capsules containing these candidate alternate insulator materials were fabricated and processed as shown in Table V-6. The cold-compaction cycle was eliminated for one of each type of insulator test to observe the effect on deformation response. Both N- and P-type TE materials were included to test compatibility with these micas. The capsules were tested for 1000 hr at

TABLE V-6
INSULATOR MATERIAL PROCESSING

Capsule Serial Number	Type Materials		Processing
	PbTe	Mica	
EXC-75	N	borate-bonded	no cold autoclaving
-76	N	India Ruby	hot and cold autoclaving
-79	P	borate-bonded	hot and cold autoclaving
-80	P	India Ruby	no cold autoclaving

650°C (1202°F). After testing, the capsules were section, examined metallographically, and probed for Seebeck coefficient values. Visual examination showed that the mica in EXC-75, -76, and -79 was not sheared or cracked and was free of any traces of TE material in the lamellae. The mica from EXC-80 was fractured and TE material was observed in these fractures. Electrical measurements showed that both micas were compatible with N-type material. However, both the BBM and the IRM reacted with the P-type PbTe at the interfaces so that neither one could be considered suitable for use in power modules. The two micas, therefore, have been eliminated as possible choices.

Capsule and pressure tests designed to test the integrity of unprocessed mica yielded results which indicated that vapor contamination was a possible extraneous factor which obscured the basic purpose of the experiment. Specifically, split-sleeved test capsules containing N- and P-type PbTe were fabricated but not autoclaved. Destructive examination of these noncompacted capsules after 1000 hr at 1202°F (650°C) showed that the pattern of contamination in the N-type material was probably caused by vapor diffusion around the mica via the void volume produced by assembly tolerances. Compacted capsules which eliminate the void volume yielded different results as described below.

In another experiment a sheet of 0.005-in. hot-treated unprocessed mica was pressed between N- and P-type PbTe. The electrical resistance of the resulting circuit was measured as a function of temperature from room temperature to 1382°F (750°C). The entire resistance of the circuit could be

attributed to the mica since the resistances of the other components are negligible. A resistivity plot showed that the resistivity of heat-treated mica (the same mica used in modules) was still 10^8 ohm-cm at 1382°F (750°C). The large value for the circuit indicated very little, if any, interaction between the PbTe washers through the mica. The N-type/mica/P-type configuration was held for 165 hr at 1202°F (650°C) to determine if any resistance change occurred with time at temperature. No significant change in the resistance was noted over the life of the test, indicating that unprocessed mica for these test conditions was an effective electrical insulator and diffusion barrier between N- and P-type PbTe. The test was then raised to 1382°F (750°C).

After 100 hr at 1382°F (750°C) the test was halted and the PbTe pellets were removed and examined with the Seebeck probe. The pellets were probed on the face which was pressed against the mica insulator. The P-type pellet remained unaffected but the N-type PbTe exhibited high Seebeck values across its entire face. The relative value of the Seebeck coefficient was higher than any values measured on module washers, even for those that had been partially converted. The pellet was ground away in 0.005-in. steps and the Seebeck probe trace repeated after each removal of material. The Seebeck coefficient remained essentially unchanged, even after 0.020 in. of the pellet was ground away. Other experiments and the high value of the Seebeck coefficient, coupled with its even distribution throughout the pellet and with the constant high value of the resistivity of the circuit (10^8 ohm-cm), leads to the conclusion that the contamination of the N-type pellet was probably due to vapor diffusion.

Two sleeved EXC test capsules were fabricated using a 1-mil W foil sandwiched between two 1.5-mil mica discs. After standard processing these capsules were tested for 1000 hr at 1202°F and EXC-82 was removed for destructive examination. The temperature of EXC-81 was then raised to 1382°F and testing continued for another 1000 hr. Examination of EXC-82 showed that the foil was successful not only as a diffusion barrier but also as a strengthening member for the mica. Seebeck probe across the N-type pellet face in contact with the mica showed that no reaction had occurred in this critical area. Previous tests which did not use the barrier invariably yielded data which showed considerable contamination had occurred in the N-type material as a result of

P-type diffusion. However, in this compacted barrier-type capsule no contamination was observed. Figure V-23 shows that even when the mica was deformed into a node of the type which causes cracking and delamination in other capsules, the W appears to have prevented this undesirable behavior. All the data obtained from EXC-82 showed that the interface as well as the entire pellet for both N- and P-type material was completely unaffected; hence, no interdiffusion occurred. EXC-81 tested at 1382°F (750°C) is currently being examined for evaluation of effects at the higher test temperature.

Four capsules were fabricated and placed on test at 1202°F (650°C) to evaluate how well W-coated mica acts as a diffusion barrier. The mica, 5 mils thick, was coated with approximately 1000 Å of W evaporated by means of an electron beam. Two of these discs were assembled with the W coatings facing and then positioned between N- and P-type PbTe in a standard sleeved EXC capsule. The four capsules were designated EXC-A5 through -A8. Two of the four contained three W-coated disc assemblies and one Mo-coated assembly. The other two contained all W-coated disc assemblies, one of each type according to standard procedures, with two omitting the cold autoclaving. The circuit resistance has been measured to be in the megohm range over the test time of 320 hr. The test will continue for 1000 hr after which destructive examination will be carried out.

Two capsules, EXC-83 and -84, are currently on test at 1202°F (650°C). These have W-coated mica discs with undersized diameters. The experiment is designed to evaluate mica deformation modes when additional clearances and W coatings are used, and to study diffusion patterns.

4. Lead Telluride Evaluation

An evaluation to establish what was diffusing from the P-type material and contaminating the N-type PbTe led to the conjecture that the excess Te in the P-type should be fairly mobile. To establish the effects of Te diffusion on the electrical properties of N-type PbTe, three separate experiments were begun.

Varying amounts of Te (0.25, 0.50, and 1.0 wt %) were mixed with N-type PbTe powder which was pressed into a series of electrical measurement bars. The bars were processed using the standard hydrogen heat treatment. The

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Figure V-23. Mica/Tungsten/Mica
Tested in Capsule EXC-82 for
1000 hr at 1202°F (650°C)

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electrical characteristics of one bar from each series and of a control bar containing no excess Te were measured as a function of temperature. Figure V-24 shows the relative Seebeck coefficient vs temperature for these bars.

It was apparent that as the amount of Te increased, the material changed from N- to P-type at room temperature with increasingly higher values for the P-type Seebeck coefficient. With increasing temperature the material eventually changed back to N-type material. The data clearly showed that as the excess Te concentration increased, the crossover from P- to N-type occurred at higher temperatures. Accompanying the changes in Seebeck coefficient was an increase in the electrical resistivity over the temperature range of the measurements.

Figure V-24. Seebeck Coefficient vs Average Temperature for Heat-Treated N-Type Bars Containing Excess Tellurium

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Figure V-25. Seebeck Coefficient and Resistivity vs Average Temperature for Center N-Type Pellet from EXC-52, Tested for 2800 hr at 1202 $^\circ\text{F}$ (650 $^\circ\text{C}$)

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It is of interest to compare the results obtained on the N-type bars containing excess Te with those measurements taken on a contaminated pellet. The pellet was N-type and occupied the center position between two P-types in Capsule EXC-52. The capsule had been heated to 1202°F (650°C) and held for 2800 hr during which time the circuit resistance increased in value. Upon removal from test the capsule was sectioned axially and probed electrically. The center pellet was completely contaminated and converted to P-type. Electrical-resistivity and Seebeck-coefficient measurements were made as a function of temperature, Figure V-25. The resulting Seebeck-coefficient curve is quite similar to those obtained on bars containing excess Te. The material starts as P-type at room temperature and converts to N-type at about 600°K (622°F). Complete agreement of the curves does not occur, which is to be expected because of the difference in the way the specimens arrived at the final state. The bar had a homogeneous distribution of excess Te whereas the pellet probably had a concentration gradient of contaminant from the ends to the center.

It is significant that in all cases observed the electrical resistivity has been considerably increased above the normal values for N-type material. In addition, the Seebeck voltage over any module operating temperature range, $S.V. = \int \alpha dT$, would be reduced if the washer were completely contaminated. However, the contamination is not homogeneous for a washer and accordingly the results would not be completely comparable. In any case the data show that N-type material can be contaminated by excess Te (without any sodium present) so that the electrical properties are similar to those obtained on a pellet that was contaminated by contact with P-type PbTe.

To further define the effects of excess Te as a possible diffusing contaminant, P-type material was mixed with additional amounts of excess Te (0.25, 0.50, and 1.0 wt %), melted in quartz tubes to assure homogeneity of the batch, then processed in the same manner as normal P-type material. Pellets were pressed from each batch of material, along with control pellets from normal P-type PbTe, and one from each batch (0.25, 0.50, and 1.0 wt % excess Te) was inserted into a graphite die along with a pellet of normal N-type material. This arrangement formed a diffusion couple, with the amount of excess Te varying from the amount present in normal P-type PbTe to, in the extreme

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Figure V-26.
Seebeck Coefficient vs Distance
from P/N Interface

Figure V-27.
Electrical Properties of N-Type
Lead Telluride After Sublimation
Experiment

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case, 1.0 wt % over the normal amount. The dies were inserted into an inert-atmosphere furnace and compacted for 7 hr at 1472°F (800°C) under a load of 5000 psi. Immediately following compaction the dies were sectioned axially and ground and polished for Seebeck-probe examination. The Seebeck-probe traces were taken in an axial direction, starting in the P-type PbTe and proceeding into the N-type material.

It was postulated that if Te diffusion was responsible for contamination of the N-type material, the affected portion of the N-type pellet should become greater as the Te concentration increased. Figure V-26 is a plot of the Seebeck coefficient vs distance from the interface of the P-/N-type couple for the three alloys considered. The figure shows that as the Te concentration of the P-type increased, the affected distance and magnitude of the contamination also increased.

The results of this die experiment supports the possibility that Te diffusion through or around the mica insulators in TEM modules is probably responsible for contamination of the N-type PbTe and the subsequent degradation of module performance during life testing.

An experiment was set up to evaluate whether vapor transport from the P-type PbTe could produce changes in the N-type material at a lower temperature under isothermal conditions. Accordingly, a bar of P-type lead telluride was sealed into a Vycor vial along with a bar of N-type material. The bars were kept separated by means of an indentation in the wall of the vial. The vial was backfilled with one-half an atmosphere of high-purity argon, sealed off, and inserted into a furnace at 1100°F (593°C). After 1100 hr at test conditions the vial was removed from test and the electrical properties of each bar measured. Figure V-27 is a plot of the resistance and Seebeck coefficient vs the average temperature of the N-type bar. From these data it is evident that the bar exhibited P-type electrical characteristics to 720°K (838°F) where it experienced thermal conversion back to N-type. The data are quite similar to the results obtained by solid-state contamination of N-type PbTe caused by direct contact with P-type material or by direction additions of Te to N-type.

Similar measurements carried out on the P-type bar showed that no changes occurred in the electrical properties. Previous experiments have shown that the sublimates produced by heating P-type PbTe are PbTe and Te. Hence the

sublimation experiments also indicate that Te vapor can affect N-type material at 1100°F, (593°C) even when no physical contact is made. However, the results of compacted capsule tests described above show that in the absence of a void volume (the situation existing in a processed module) vapor transport does not contribute as a diffusion mode.

5. Material Studies

For some time questions have been raised concerning oxidation rates of Ni and Fe pins which have been used as electrical leads from the internal Te circuit to external circuits.

Accordingly a set of pins, two Ni and two Fe, were inserted into a furnace and heated to 975°F (525°C) in air. Four stainless-steel wires were attached to each pin and resistance measurements were taken via the four-print-probe method using a Keithly milliohmmeter.

After 5000 hr at 975°F (525°C) the Fe pins exhibited an 11% rise in resistance while the Ni pins exhibited no change (slope = 0) on the resistance-vs-time plot. The conclusion drawn here is that Ni is a completely acceptable material for use as conductor pins in reactor modules, at least up to 5000 hr. It should be pointed out also that the temperature of the test under discussion is higher than the temperature of pins in any module tested to date or any anticipated operating range in the future.

Another area of inquiry for conductor pins was the interface at the Ni-pin/Mo collector rings. It has been suggested that a possibility exists for inter-metallic compound formation between Ni and Mo, although the temperature at this junction is relatively low, 1100°F (593°C).

An experiment has been started to determine the long-time effect of contact between the two materials at a temperature approximating module conditions. Four TEM-13-type Ni conductor pins were inserted into one end of a Mo rod by shrink-fitting them into drilled holes. The holes and pins were of the same dimensions as those in the Mo collector rings in a module. The entire configuration will be tested in a vacuum by measuring the resistance as a function of time using the same technique as in the oxidation experiment described above.

6. Process Development

An investigation has been underway to define a process for applying thin W coatings to mica insulating washers. The purpose of the coating is to provide a barrier to diffusional intercontamination between adjacent P- and N-type PbTe washers. After screening both chemical-vapor deposition and sputtering as potential coating processes, both with unsatisfactory results, electron-beam evaporation was chosen as the coating process to be developed. At present some coatings 800 to 1600 Å thick have been applied to 0.0025-in.-thick mica stock.

In the first contemplated applications, TEM-15C SN-4 and -8, two 0.0025-in.-thick coated insulating washers (coated faces together) would occupy each insulator position, giving a total barrier thickness of 1600 to 3200 Å. A number of preliminary experiments were conducted, some of which were successful. A tentative process sequence was defined for mica during the various preparation stages. The task remaining is to identify and finalize the process parameters in the coating step itself, and then qualify the process by obtaining reasonable reproducibility over several consecutive experimental runs.

F. TUBULAR MODULE DESIGN CHARACTERISTICS

Table V-7 summarizes design characteristics for the tubular module.

TABLE V-7
SUMMARY OF TUBULAR MODULE DESIGN CHARACTERISTICS

Module	Diameter		Circuit Length (in.)	Washer Thickness (in.)	Mica (in.)	Material		Conductor Ring		Collector Ring
	Inner* (in.)	Outer (in.)				P	N	Inner	Outer	
TEM-9U-7, -8, -9, -10 -9AD-1	0.5	1.66	7.2 15.0	0.090	0.005	TEGS-2P	GE-N	Tungsten Tungsten-foil/ Iron	Tungsten-foil/ Iron	Iron
TEM-13A-1, -2				0.090	0.005	TEGS-2P		Tungsten	Molybdenum	Iron
-13B-1, -2				0.050	0.005	-2P		Tungsten	Molybdenum	Iron
-13C-1, -2				0.030	0.005	-2P		Tungsten	Molybdenum	Iron
-13D-1, -2	0.75	1.64	15.0	0.030	0.003-2 pc	-3P	GE-N	Iron/Iron	Iron	Iron
-13F-1, -3				0.030	0.003-2 pc	-2P		Tungsten/ Iron	Molybdenum	Molybdenum
-13F-2				0.030	0.003-2 pc	-2P		Tungsten/ Iron	Molybdenum	Molybdenum
-13G-1, -2, -3, -4, -5				0.090	0.005	-2P		Tungsten/ Molybdenum	Molybdenum	Molybdenum
TEM-14A-1 [†]	0.75	2.14	5.6	0.700	0.010	TEGS-2P	GE-N	Tungsten-foil/ Molybdenum	Molybdenum	Molybdenum
-14A-2, -3 [§]								Tungsten/ Molybdenum		
TEM-15	0.75	1.64	4.0	0.050	0.005	TEGS-2P	GE-N	Tungsten/ Molybdenum	Molybdenum	Molybdenum

*After processing

[†]Six power leads on each end

[§]Twelve power leads on each end

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VI. COMPONENT DEVELOPMENT

A. INTRODUCTION AND SUMMARY

The current component development effort comprises the design-verification testing of the S8DR components and the technology development applicable to reliable high-performance components for the reference zirconium hydride reactor. The latter include bearings, actuator, 4π shieldable reflector, shielding, and NaK pump.

The design-verification testing of the three S8DR actuators and four sets of the control-drum bearings is nearing the end of the scheduled 12,000 hr. These components continue to perform satisfactorily.

Both the Ni-clad silver and BeO-hardened copper conductor material show stable electrical performance after 12,000 hr at 1200°F and 8700 hr at 1400°F. Although both Hiperco-27 and Armco iron exhibited stable magnetic properties throughout the 12,000-hr test at 1200°F, the former is preferred for flux densities greater than 12,000 gauss.

The prototype high-temperature bearing is performing satisfactorily after 3500 hr at 1500°F and 10^{-8} torr. A molybdenum-sulfide/tantalum compact which is being evaluated as possible backup to the carbon graphite for bearings appears to have an upper temperature limit of 1300°F. The 5,000-hr test at 1600°F indicates that both the Al_2O_3 and TiC bearing material is compatible with the P5 carbon graphite.

The first 4π shieldable control-drum assembly was completed. Iron titanate is being evaluated as possible thermal emittance coating for both the Ta-10W absorber and the niobium cladding for the BeO reflector. The design of the control-drum-drive system was begun.

The fabrication of the development NaK pump is continuing. Problems encountered in the forming of the complex throat section have caused some delay.

Preliminary results indicate the addition of carbon to the lead has no beneficial effect on the decarburization of the liquid-lead shield structure. Design and fabrication of that and the annular LiH test shields are in progress.

B. CONTROL-DRUM ACTUATOR

1. S8DR Design-Verification Test

Three actuators of the modified S8DR design are undergoing a 12,000-hr design-verification test. The test was described in the previous progress report.* SN013 has completed all tests; SN010 and 011 have completed 1920 and 908 hr, respectively, of the 2000-hr dwell test. Performance data are similar to those reported previously,* and are within design requirements. Both units are operating satisfactorily at present.

The investigation of the previous failure of actuator SN011 concluded that the probable cause of failure was the introduction of a particle of foreign material into the magnetic gap. The presence of this particle resulted in a progressive rubbing and buildup of material on the mating parts, which ultimately locked the rotor. This unit has been repaired, reassembled, and returned to test.

2. Zirconium Hydride Actuator

The efforts of this program are directed toward the development of advanced actuators for ZrH reactors.

a. Coil Reliability

Tests are being conducted to evaluate potential high temperature and reliability conductor materials for actuator coils. These tests are being performed on coils wet-wound with Al-T cement on rectangular-shaped alumina bobbins. Three different conductor materials have been tested or are under test as candidate material for coils; nickel-clad silver, Cube alloy, and Oxalloy.

The first test run was designed to produce failure in the conductor materials by thermal cycling to 1300°F. This test was converted into a steady-temperature 1400°F test after more than 400 thermal cycles did not produce a failure. The remaining four coils of this test have shown no significant circuit resistance variation after a total of 406 thermal cycles to 1300°F, nine thermal cycles to 1400°, 339 hr at 1300°, and 8740 hr at 1400°F.

The detail examination of coils subjected to total of 10,082 hr at 1200°F and 411 thermal cycles between 400° and 1200°F has been completed. Both the Cube alloy and Ni-clad Ag conductors were in good condition and are expected to be

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satisfactory for 1200°F application. The Ni-clad Ag is preferred, particularly for rectangular-shaped coils, because of its greater flexibility. The relative room temperature stiffness of the Cube alloy makes it difficult to wind with good repeatability.

A third test is being made with coils wound with Oxalloy conductors on rectangular-shaped alumina bobbins mounted in an S8DR actuator housing, to obtain long-term high-fatigue-loading testing of Oxalloy wire at 1125°F.

Testing was concluded after a total test time of 13,867 hr and a total of 541 thermal cycles. Detail examination of the coils and conductor revealed that elongation of the latter had caused it to extend beyond the original coil envelope in one location. This appears to be caused by a thermal ratcheting mechanism induced by thermal cycling and not by time-at-temperature. In general the coils were in good condition after testing. It is concluded that the Oxalloy conductor should provide reliable long-term service at 1125°F, provided the thermal cycles are limited to 250 in rectangular-coil (S8DR stator-coil) configuration and 500 in round-coil configuration.

Performance test of prototype actuator coils has been started. The coil configuration, bobbin shape and material, conductor size and number of turns, lead-wire termination, and fabrication process are representative of the advanced actuator design. Six coils were wound with Cube alloy and six with Ni-clad Ag. After 40 thermal cycles and 2,276 test hours of which 2,063 were at the test temperature of 1300°F, two Cube-alloy coils failed. The failures were due to brittle fracture of the conductor near the lead-wire termination. This area is subject to mechanical flexing during the coil-fabrication process. The inherent stiffness of the conductor probably results in localized high stresses. This area of high stress is more subject to corrosion by the acidic solvent of encapsulant during the winding and curing cycles. Testing is continuing; the remaining ten coils have accumulated 50 thermal cycles and 3,332 hr at 1300°F in a total test time of 3,613 hr of a scheduled 20,000.

b. Magnetic Materials

Magnetic and impact testing on two magnetic materials, Hiparco-27 and Armco magnetic ingot iron (MII), was completed. (Magnetic testing is described in References VI-1, -2, and -3.) Both materials are stable with time at temperature as determined by testing at 1200°F for 12,080 hr. The magnetic properties

TABLE VI-1
S8DR CONTROL-DRUM BEARING TEST STATUS

Item	Test No.				
	1	2	3	4	5
<u>Bearing Material</u> (Reference: P5 vs Al ₂ O ₃ ; Backup: P5-N vs TiC)	Refer- ence	Refer- ence	Refer- ence	Refer- ence	Backup
Nominal Test Temperature, °F	1150	1150	1250	1150	1150
Test Pressure, torr	10 ⁻⁵	10 ⁻⁵	10 ⁻⁵	10 ⁻⁸	10 ⁻⁸
Maximum Friction Torque, in.-lb					
Initial (ambient temperature)	1.8	1.6	1.1	2.7	3.8
For 30 thermal cycles:					
450 °F	3.2	3.7	2.8	3.1	4.0
Nominal test temperature	1.3	1.7	2.2	2.8	2.4
Operational Testing, in.-lb					
(1 thermal cycle each 100 hr)					
450 °F	5.3	4.5	5.2	3.2	6.5
Nominal test temperature	3.8	3.0	3.8	2.2	4.7
Pre-1000-hr Dwell	1.9	2.4	2.9	1.7	-
Post-1000-hr Dwell	2.7	3.1	3.1	*	-
Total Thermal Cycles	86	81	83	60	33
Equivalent 270° Drum Cycles	274	270	300	232	92
Total Hours: at vacuum	12,755	12,615	12,856	11,014	6,388
at 1150 °F	11,676	11,674	-	9,970	5,180
at 1250 °F	623	623	11,905	672	714

*1000-hr dwell test in progress.

of Hiperc-27 are superior to the magnetic properties of MII at densities above 12,000 gauss or at temperatures above 1200°F. Impact testing indicates the MII is about twice as tough (resistance to impact) as Hiperc-27 at ambient temperature. At elevated temperatures, 600°F and above, the Hiperc-27 magnetic material has the greater resistance to impact. Neither magnetic material has sufficient toughness to be used as a structural material to resist impact.

C. CONTROL-DRUM BEARINGS

1. S8DR Design-Verification Tests

Design-verification testing of the S8DR control-drum bearings continues. Reference design bearings, Al_2O_3 on Inconel-750 against P5 carbon-graphite, are at ground test conditions (two tests at 1150°F, one at 1250°) of 10^{-5} torr, and one test is at simulated space conditions of 10^{-8} torr and 1150°F. A backup design bearing, TiC on Inconel-750 against P5-N carbon-graphite, is also at 10^{-8} torr and 1150°F. The scheduled testing will subject the bearings to 12,000 hr simulated life at vacuum and include 85 thermal cycles and a 1000-hr dwell at temperature. Table VI-1 summarizes current test status and bearing performance. The frictional torque values as shown in the table are within S8DR operational limits.

2. Advanced ZrH Reactor Bearings

The ZrH reactor bearing program is designed to advance bearing technology from SNAP 8 requirements to 20,000 hr life at 1500°F. A life test on bearings with reference materials (P5 carbon-graphite against Al_2O_3 -plated Ta-10W shaft and socket) is being conducted, and additional tests to evaluate bearing materials are in progress.

a. Ultrahigh Vacuum Bearing Test

Simulated life testing of the prototype bearing is continuing at 1500°F and 10^{-8} torr under the nominal 25-lb radial and 75-lb thrust loads. Bearing friction torque has varied between 4.0 and 7.1 in.-lb during the 3500 hr accumulated at temperature. The data are shown in Figure VI-1, with the maximum friction coefficient of 0.25 (equivalent to a torque of 7.1 in.-lb) indicated. Testing will continue to 10,000 hr.

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Figure VI-1. Ultrahigh Vacuum Bearing Test, Friction-Torque
at 1500°F, 10^{-8} torr

b. Sliding Friction

Two specimens of an 80 wt % MoS_2 , 20 wt % Ta compact material (Boeing 108M) are undergoing short-term sliding friction testing to 1600°F at 10^{-8} -torr and 150-psi loading to evaluate its potential as an alternate to the P5 carbon-graphite. One specimen is being tested against bare Ta-10W and the second against Al_2O_3 -coated Ta-10W. Short-term (approximately 72 hr) static dwell tests were made between 1200 and 1600°F. The results (Figure VI-2) show that potential use of the material seems limited to 1300°F or lower due to the friction rise following dwells above 1300°F. While all friction coefficients for temperatures equal to or below 1300°F fall between 0.2 and 0.3, dwells at 1400, 1500, and 1600°F result in proportionately greater friction coefficients. Table VI-2 summarizes the friction data.

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Figure VI-2. 108M Compact Material Sliding Friction Test Data

TABLE VI-2
 MAXIMUM FRICTION COEFFICIENTS OF 108M VS
 ALUMINUM OXIDE AND TANTALUM/10 TUNGSTEN

Temperature (°F)	Bearing Couple	
	108M vs Al_2O_3	108M vs Ta-10W
Ambient	0.39	0.30
400 to 1200	0.25	0.32
1300	0.23	0.24
1300 (post-dwell)	0.25	0.26
1400	0.27	0.26
1400 (post-dwell)	0.34	0.39
1500	0.25	0.26
1500 (post-dwell)	0.35	0.52
1600	0.31	0.34
1600 (post-dwell)	0.61	1.10

After 900 hr at test the 108M against Ta-10W seized at 1600°F; the tests were discontinued and the specimens inspected. Figure VI-3 shows post-test condition of the two couples. Note the heavy material transfer to the bare Ta-10W and the light material transfer to the Al_2O_3 . Neither 108M specimen shows any detrimental effects of the test. The contacting surfaces of both friction couples were cleaned and testing at 1300°F was started. After 250 hr at test, maximum friction coefficients have been 0.47 against Ta-10W and 0.24 against Al_2O_3 .

c. Material Compatibility

The 5000-hr 1600°F material-compatibility tests have been completed and analysis of the test couples is in progress. Preliminary analysis shows the reference P5 carbon-graphite against Al_2O_3 on Ta-10W and the backup P5 against solid TiC (K-162B) couples to be in excellent condition. Figure VI-4a shows a 100X photomicrograph of a typical section of the Al_2O_3 vs P5 specimen. There are no cracks through the Al_2O_3 coating and no signs of coating-substrate separation. There is evidence of some minor grain boundary attack in the Ta-10W

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108M (80 WT % MoS_2 , 20 WT % Ta) AGAINST Ta-10W

108M AGAINST Al_2O_3 PLATED ON Ta-10W

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Figure VI-3. Molybdenum Disulfide Dry Film Lubricant Sliding Friction Studies, 10^{-8} torr to 1600°F

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Ta-10W

8516-7-1

(100X)

(a) P5 CARBON-GRAPHITE AGAINST Al_2O_3 ON Ta-10W

8516-6-1

(100X)

(b) P5 CARBON-GRAPHITE AGAINST SOLID TiC

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Figure VI-4. 5000-hr Material Compatibility Studies at 1600°F and 10^{-8} torr

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beneath the coating and some nitrides and carbides are also present. Although the effect of these appears to be negligible, it will be studied. The P5 carbon-graphite against solid TiC is shown in Figure VI-4b; no material deterioration or transfer appears to exist.

D. SHIELD MATERIAL DEVELOPMENT

1. Liquid-Lead Studies

The program to study the effectiveness of saturating lead with carbon to minimize the decarburization of structural alloys by the molten Pb has been described in detail in an earlier report.^{(VI-4)*}

Metallographic examination of the structural alloys and chemical analysis of the Pb from test capsules exposed for 2500 hr at 1300 and 1500°F is currently in progress.

It has been observed in these tests that the addition of C to Pb did not have the desired effect of reducing or eliminating decarburization of the 1020 steel or the Croloy 2-1/4 steel. For example, earlier tests on Pb vs Croloy 2-1/4 steel tested at 1500°F for 2500 hr revealed that decarburization of the steel had occurred to a depth of 0.0025 in. The addition of the zirconium and magnesium corrosion inhibitors to the Pb reduced the decarburization to about 0.001 in.

Two groups of Croloy 2-1/4 test capsules in the current study contain Pb with C additions, one group with sufficient C (100 ppm) to saturate the Pb at 1500°F, the other with half the amount (50 ppm). The two groups were found to be decarburized to a depth of 0.003 and 0.004 in., respectively.

The addition of the same quantities of corrosion inhibitors (250-ppm Zr and 400-ppm Mg) to the C additives produced a decarburized layer of steel 0.005 in. thick.

These unexpected results, i. e., that the addition of C to the Pb did not suppress the decarburization but may have, in fact, increased the amount of decarburization, are unexplainable with the data on hand.

It must be concluded from these tests that the addition of the Zr (250 ppm) and Mg (400 ppm) is more effective in reducing decarburization of the Croloy 2-1/4 steel than the addition of 50 to 100 ppm C. It is believed that the decarburization is reduced by the formation of a boundary layer of the Zr and Mg additives or some of their compounds between the Pb and the structural alloy.

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

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Figure VI-5. Liquid-Lead Shield

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The effect of adding C to the Zr and Mg in the Pb is to increase the decarburization to an amount equal to or greater than that observed in capsules without additives of any type. It is possible that the C combines with the Zr and/or the Mg and reduces or eliminates the corrosion-inhibition action of the metallic additives. If the added C is combined with the metallic additives, very little or none of the C is available for solution in the lead, and the steel is, therefore, decarburized.

These assumptions made on the basis of the 2500-hr test data will be re-examined when the results from the 5000-hr tests are available within the next two months.

2. Liquid-Lead Shield Test

The test plan for the liquid-lead shield test was completed and a preliminary design of the test shield and handling fixture was made. Figure VI-5 shows the layout of the test shield, instrumentation, etc. A redesign is currently in progress to decrease the layer of Pb to 4.5 in. and to reverse the curvature of the top to place the shield nearer the reactor core.

3. Aggregate Neutron-Gamma Shielding Materials

The de-emphasis of the aggregate neutron-gamma shield materials program reduced the activities under this program to the completion of the previously described compressive-creep-testing of the LiH/ZrH aggregates.

The 1000°F tests on pellets of LiH/ZrH under loads of 25, 50, and 150 psi were completed, the capsules opened, and the pellets examined. The pellets were essentially unchanged in appearance, except for a thin whitish coating which was assumed to be lithium hydroxide. The post-test dimensions of the pellets were the same as the pretest values within the limits of the ability to remeasure the pellet at the exact location as before the test.

If the total creep in 1000 hr for the LiH/ZrH aggregate specimens is compared to the creep of LiH measured on the same equipment, significant differences are noted, as shown on Figure VI-6. At room temperature the creep of LiH and the LiH/ZrH aggregate are essentially the same. At 350°F that for the aggregate is slightly less than for LiH, and at 750°F the aggregate creep is significantly less than for LiH. At 1000°F the creep of the aggregate is essentially nil, but for LiH at 50 psi it was about 2% and at 150 psi about 9%.

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Figure VI-6. Compressive-Creep-Test Data

It is assumed that the aggregate being a mixture of ZrH powder and granules cemented together by LiH does exhibit a small amount of creep until which time the ZrH becomes a packed column and creep ceases. Examination of the test data reveals that almost the entire movement occurred in the first 100 hr, at which time movement essentially ceased.

The reason for the very small amount of creep in the 1000°F samples is explained by the fact that these pellets were initially tested at 350 and 750°F during which time the allowable creep had occurred. When the temperature of the test was increased to 1000°F, creep could not occur since the ZrH had already reached a packed column condition.

It was concluded that the LiH/ZrH aggregates have excellent load-bearing characteristics at temperatures up to 1000°F, and that this property could be used in future shield design to reduce the weight of the structural support members in the shield system.

4. Annular Lithium Hydride Shield (Experimental No. 3)

The objective of this test is to design an annular LiH shield and to investigate the problems associated with casting this unique configuration.

The test will provide information on the problems associated with the design, fabrication, and casting of a unique annular configuration which simulates in geometry, but not in size, the annular shield surrounding the reactor in the 4π shadow shield system. This test also will provide information on potential radiation streaming problems which might be caused by cracks or voids in the casting.

The test shield was designed using a surplus SNAP 10A shield vessel (Figure VI-7). The top head is removed and rigidized flat plates added to the top and bottom. A 9-in.-diam tube is placed in the center of the vessel to simulate the reactor cavity.

SNAP 2 casting hardware is modified to provide three filler tubes to connect the shield vessel to the reservoir which holds the surplus molten LiH during the casting operation.

Instrumentation consists of: 26 Chromel-Alumel thermocouples which will monitor the temperatures during the melting and freezing cycles of the LiH in the reservoir; the three filler tubes; and the three vertical zones in the shield vessel directly below the filler tubes.

Auxiliary heaters are mounted on the bottom of the reservoir and on the three filler tubes to provide fine control of the temperatures in those regions during the final stages of freezing.

Two standpipes or vent lines are installed on the flat upper head of the vessel to provide exit paths for gases to reduce pocketing under the flat surfaces. One vent line is routed to the outside of the reservoir, and the other extends up through the reservoir in a separate pipe. Each of these vent lines is placed between a pair of filler tubes. The area between the remaining pair of filler tubes is unvented to provide a nonvented area for comparison.

The shield vessel cavity is filled with a stainless-steel honeycomb matrix with the cell direction parallel to the central cavity. The fabrication of the casting assembly is about 80% complete.

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Figure VI-7. Annular Lithium Hydride Shield
(Experimental No. 3)

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E. NaK PUMP

The NaK pump development is focused on a multiloop dc conduction pump applicable to the reference zirconium hydride reactor — compact thermoelectric power system. A mockup assembly of the development pump described previously^(VI-5) is shown in Figure VI-8.

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Figure VI-8. NaK-Pump Assembly Mockup

(60X)

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Figure VI-9. Electron-Beam-Welded Support Plate

The completion of the two development pumps to evaluate performance has been delayed by fabrication problems on the throat assemblies. As can be seen in the figure, the center section of the throat has a rectangular cross-section but the ends are circular. The throats were cold-formed from seamless round tubing by first forming a rectangular tubing then reforming the ends to provide the transition. Minute cracks on the internal surfaces have been observed on most of the throats formed to date. Additional throats utilizing a modified forming process are now being fabricated.

The walls of the rectangular throat section are held to a minimum practical thickness to minimize the electrical resistance and shunt losses. The thin walls require a support plate at the midplane to limit the stress and the deflection imposed by the internal NaK pressure. The design requires that the attachment between the support plate and the throat wall contain no potential stress risers. Two methods for accomplishing the required attachment have been developed.

Figure VI-9 shows the previously described^(VI-3) electron-beam-welded joint with an acceptable "fillet" on either side of the plate. Because the support plate cannot be located visually during the welding, careful fixturing is required to align it with the beam source. To alleviate this difficulty an alternate joint design utilizing a tungsten-inert-gas weld is being evaluated. In this case the throat walls are slotted to receive and position the plate during the welding.

Fabrication of the multiple-strip flexible bus assembly by the diffusion-bonding method was successfully demonstrated. As shown in Figure VI-8 the ends of the multiple-strip were bonded with the spacers to form solid end pieces. The method of brazing this bus assembly and the lower buses to the two throats in a single operation is now being qualified.

F. 4π SHIELDABLE REFLECTOR

The objective of this project is to develop a control reflector system for a 4π -shielded reactor. The current program consists of three tasks: (1) drum development; (2) prototype reflector assembly development; and (3) material studies.

1. Drum Development

Fabrication of the development drum, Figures VI-10 and -11, was completed. This drum was the first of the reference design and consisted of the

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Figure VI-10. Development Drum No. 1

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Figure VI-11. Development Drum No. 1 During Fabrication
Before Installation of Cladding

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following: Ta-10W absorber, UOX + 0.5% MgO grade BeO reflector, Ta-10W mounting bands for the BeO, and Ta-10W cladding.

Recent analysis shows that the Ta-10W cladding significantly reduces the drum control worth. A review of materials indicates that Cb-1Zr alloy should be acceptable for use as cladding.

A method for fastening the BeO retaining band was developed. Blind pull rivets of T222 Ta alloy are used to attach the bands. Pretensioning of the band is accomplished using a dead weight, and the rivet holes are drilled through the band and into the absorber while weighted. This assures dimensional control and alignment. Strain gages placed on the band showed an average loss of only 8% of the pretension when the weight was removed after rivetting, which is an acceptable amount.

2. Prototype Reflector Assembly Development

Two drum and drive-system conceptual designs were completed, Figures VI-12 and -13, one for the shadow-shielded and the second for the 4π -shielded configuration. The shadow-shielded design utilizes the S8DR-type drum actuator mounted on the top of the reactor vessel. The actuator was connected to the pin-and-slot drum-lockout mechanism mentioned in Reference VI-3, which was in turn connected to the drum upper shaft through a flexible coupling. In the 4π -shielded design the actuator and drum-lockout mechanism (same as for the shadow-shielded design) was placed on the exterior of the shield above the reactor and connected to the drum upper shaft through two flexible couplings and a tube. Design of the first development drive train for the 4π -shielded configuration was started.

3. Material Studies

The material study effort consists of three subtasks: (1) BeO property tests; (2) coating evaluation tests; and (3) material compatibility.

a. Beryllium Oxide Materials Properties

Density, grain size, and modulus of rupture tests were conducted on unirradiated Thermalox-995 BeO (Brush Beryllium Co.). This material was used for the shock and vibration samples reported previously.^(VI-3) The test samples

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Figure VI-12. Control-Drive With Shadow Shield

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Figure VI-13. Control-Drive with 4π Shield

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were made from cold-pressed and sintered blocks approximately 4-1/2-in. diam by 6-in. long. Similar tests were also made on cold-pressed and sintered UOX + 0.5% MgO BeO used for the development drum. The test data are tabulated in Table VI-3. It was found that the density of Thermalox-995 BeO was slightly less at the center of the block tested than at the periphery at one end. The grains in the center of the Thermalox-995 BeO block showed a slightly larger average and maximum grain size at the surface.

TABLE VI-3
BERYLLIUM OXIDE MATERIAL PROPERTIES

Property	Thermalox-995	UOX + 0.5% MgO
Density, gm/cc:		
(Number of samples	10	11)
Minimum	2.85	2.86
Maximum	2.87	2.92
Average	2.86	2.88
Grain Size, μ :		
(Number of samples	5	2)
Average	16.0-16.4	8.8-10.7
Maximum	66-73	
Modulus-of-Rupture, psi:		
(Number of samples	9	20)
Room temperature, minimum	31,329	21,967
maximum	37,430	26,442
1500° F, minimum	31,000	-
maximum	37,000	-
1800° F, minimum	25,700	-
maximum	26,800	-

The Thermalox-995 modulus-of-rupture samples were all core-drilled within a 1-in. circle at the center of each block, and the longitudinal axis of each sample was parallel to the longitudinal axis of the block. The UOX + 0.5% MgO samples represent both axial and radial orientation of the reflector block. The modulus-of-rupture values of the axial samples were less scattered than for radial samples.

b. Coating Evaluation Tests

A study resulted in the choice of iron titanate to replace the aluminide coating, which failed as reported previously.^(VI-3) This coating provides thermal emittance control on the absorber and cladding. The use of a W diffusion-barrier between the coating and Cb cladding appears to eliminate any potential materials compatibility problems as well as provide additional protection to the Cb from low partial pressures of the oxygen.

c. Material Compatibility

The base materials are being tested at 1800°F in 1×10^{-6} torr vacuum. The interface combinations are:

BeO/Ta-10W	BeO/Cb ⁷⁵²
Ta-10W/Cb ⁷⁵²	Ta-10W/Hastelloy-N
Cb ⁷⁵² /TZM	
BeO/Hastelloy-N	

Other compatibility samples, including alternate cladding material and iron-titanate-coated samples, are being prepared.

VII. FUEL DEVELOPMENT

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 VII-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.

The second main objective is to understand and, as necessary, upgrade the present ceramic-barrier metal-clad fuel-element technology to satisfy the requirements of the 20,000-hr 600-kwt ZrH 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.

TABLE VII-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%

A. FUEL-ELEMENT MATERIALS EVALUATION

1. Cladding Barriera. SCB Coating - Hastelloy-N Performance

The six 15-mil Hastelloy-N-clad developmental elements (three S8DR type as controls and three double-clad) continued on thermal endurance at 1400°F. These elements have now accrued 8456 hr with no appreciable change in leakrate

performance. The double-clad elements continue to outperform the S8DR-type controls in average leakrate and cumulative percent hydrogen loss, as shown by the computer-generated curves in Figures VII-1 and -2, respectively.

Destructive examinations were performed on two Phase III S8DR performance test elements, No. 359 and 399, which have completed 8000 hr of endurance at 1400°F. Results of the permeation scans, flex tests, and hydrogen losses are summarized in Table VII-2, along with the data from previous Phase I, II, and III elements. (Phases I, II, and III denote representation of the start-of-core, midcore, and end-of-core production, respectively.) Nothing which could be considered abnormal was found in either element. Little if any evidence of the fuel-barrier reaction was found in No. 359, while some iridescent coloring was seen on the SCB coating of Element No. 399.

b. Suppression Device

A dynamic-load suppression device is being developed for use in SNAP fuel elements. The primary function of this device is to minimize or eliminate damage to the hydrogen barrier and fuel rod due to launch loads. A column buckling device which meets all the designated criteria has been designed, fabricated, and successfully tested.

The reference column buckling device (CBD) is shown in Figure VII-3. The end disc is coated with SCB glass to alleviate potential compatibility problems. The 0.052-in.-diam 502-steel column has a buckling load of 100 ± 10 lb with the CBD configuration.

Three Incoloy-800 double-blind-end elements containing CBD's were fabricated, with axial assembly clearances of 2, 6, and 10 mils, for sinusoidal vibration testing. The sinusoidal vibration input used is given in Table VII-3. The device successfully sustained the input loads. This is indicated by post-test radiographs, shown in Figure VII-4.

Each element was tested individually, attached to the lower gridplate of a virtually rigid test fixture. The output responses measured from an accelerometer attached to the upper end pin of each element reflected the input loads up to about 500 cps. Very flat responses matching the 5g input were obtained. Some increases in response loads were measured above ~500 cps; however,

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Figure VII-1. Leakrate History of the Six 15-mil Hastelloy-N-Clad Developmental Elements

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Figure VII-2. Cumulative Hydrogen Loss for the Six 15-mil Hastelloy-N-Clad Developmental Elements

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TABLE VII-2
PERMEATION SCAN AND FLEX SUMMARY, S8DR
PERFORMANCE-TEST ELEMENTS

Test Time (hr)	Phase No.	Element No.	Percent of Total Measured Leakrate*						Coating Adherence Test Number of Flexes*				Percent Hydrogen Loss†
			1-1/4 in. of Closure End		12-1/2 in. Center		1-1/4 in. of Blind End		Closure End		Center of Tube	Blind End	
			Before	After	Before	After	Before	After	Before	After	After	After	
4000	I	32	-	-	-	-	-	-	9	5	-	23	1.83
		35	38	46	36	14	26	40	3	5	5	23	3.37
	II	94	32	43	50	18	18	39	7	1	5	23	5.42
		100	-	-	-	-	-	-	19	7	-	23	2.56
	III	170	-	47	-	14	-	39	17	3	-	23	2.38
		360	22	47	50	6	28	47	23	23	23	23	1.20
8000	I	65	-	51	-	9	-	40	5	3	-	13	5.71
		77	57	51	29	12	14	37	5	5	9	15	7.33
	II	96	23	45	59	13	18	42	9	5	5	25	7.38
		344	-	51	-	4	-	44	9	11	-	15	2.25
	III	359	-	54	-	15	-	31	23	23	-	23	2.27
		399	38	47	24	11	38	42	23	23	23	23	4.34

*Before and after testing.

†Based upon integrated leakrate.

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Figure VII-3. Postvibration Test Condition
of Column Buckling Device

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Figure VII-4. Photograph and Drawing of Column Buckling Device

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TABLE VII-3
SINUSOIDAL VIBRATION INPUT

Frequency	Amplitude
5 to 16 cps	0.368 in double amplitude
16 to 2000 cps	5g peak
1 octave/min	
up and down sweeps	

impact forces generated at these frequencies are not damaging. Hydrogen-permeation rates of the elements were not measured as these tests were only intended to prove the mechanical design of the CBD. Previous tests had demonstrated that the CBD prevented damage to the SCB glass coating.

To obtain a measure of the column buckling load design margin, all three elements were subjected to 15g sinusoidal inputs. Although the calculated impact forces indicated that the devices in elements with 6- and 10-mil axial gaps would buckle, the one with 6-mil gap did not; as expected, those with 2- and 10-mil axial gap did. Sharp increases in the response of the latter elements occurred at about 50 cps on the downsweep. This is believed to be the point at which collapse occurred. The 15g input produced loads close to the failure limit for all the gaps and the apparently contradictory result is probably due to small variations in diameter, material strengths, and eccentricity. Each element was sectioned and examined for damage. Element 9206 with the 6-mil gap and the unbuckled CBD appeared to be free of damage. The primary conclusion from these tests is that the CBD will meet the design requirement with approximately a 50% margin.

c. Alternate Cladding

The twelve 15-mil Incoloy-800-clad developmental elements continued on test. Six have double-blind-end closures and six have cup-plug-type closures. After 5000 hr of 1400°F thermal endurance, the elements were subjected to five "A"-type ramp-heating inputs. Upon permeation testing it was found that the leakrate of four of the elements (two double-blind-end and two cup-plug-type) had increased significantly, while the remaining eight elements had been affected

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AVERAGE LEAKRATE, CC (STP)/HR,
NORMALIZED TO 1400°F AND 2 ATM H₂

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Figure VII-5. Permeation History of 12 Incoloy-800-Clad Elements

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to a lesser degree. A subsequent investigation revealed that in some cases the elements had seen localized temperatures over 1600°F during the ramp-heat inputs. The cause (spacing of the induction heating coils) is being corrected.

During the investigation one low-leaking element and the four high-leaking ones were decanned; the other seven continue on test. A performance summary of the 12 elements including the acceptance, preproduction, and 5000 hr of endurance is shown in Figure VII-5. For comparison, spot data from the 30 S8DR performance-test elements is also shown. At the end of the 5000-hr endurance period, the 12 Incoloy elements averaged about 0.2 cc(STP)/hr less than the S8DR group. After equivalent time at temperature, approximately 6000 hr, the Incoloy elements have lost an average of 2.3% of their hydrogen inventory while the performance-test elements average 3.0%.

The four double-blind-end Incoloy-clad elements containing enriched uranium rods have now accrued 3500 hr of a scheduled 5000-hr endurance run. Total time at temperature including acceptance and preproduction testing is now approximately 4600 hr. Performance summaries of this group of elements are given in Figures VII-6 and -7. That the group performance is not as good as that of the 12 normal uranium elements is due primarily to the influence of one comparatively high-leaking element.

The new heat of Incoloy tubing (HH7531A) was received, inspected, and accepted. A check of the applicability of the oxidation and coating cycle parameters developed for the last heat of material (810535) was made. A 50°F increase in the oxidation temperature was made to obtain essentially the same chromium/iron oxide ratio on the substrate before coating. The subsequent as-coated material showed excellent appearance and adherence.

Tests are being run on 15-mil SCB-coated Incoloy-800 cladding to determine the effects of the coating process and NaK environment on the creep ductility and rupture properties. Testing began with cladding material that has a duplex grain structure and intergranular precipitates at the outside surface. This surface condition was the result of a step in the early ceramic-barrier coating process, as described in the preceding quarterly. This surface condition is believed to reduce the strain-to-failure and rupture time of Incoloy-800 in NaK creep-rupture tests. Additional tests are being started on coated material in which this surface condition has been eliminated.

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Figure VII-6. Average Normalized Leakrate of the Four Enriched Uranium Double-Blind-End Incoloy-Clad Elements

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Figure VII-7. Cumulative Hydrogen Loss for Four Enriched Uranium Double-Blind-End Incoloy-Clad Developmental Elements

The status of the creep-test program is shown in Figure VII-8.

Figure VII-8. NaK and Vacuum 1400°F Biaxial-Creep Tests
on Incoloy-800

2. Fuel

a. Burnable Poison Studies

Included in the preceding progress report were the results of seven of ten thermal-endurance tests of fuel elements which included both nickel-foil-wrapped fuel and fuel with vapor-deposited gadolinium oxide films. The remaining three elements were to be subjected to 5000-hr endurance testing at 1250, 1400, and 1500°F. The 1500°F test was terminated after 3096 hr with a hydrogen content of 1.59 H/Zr to prevent phase changes. This element contained three fuel specimens with Gd_2O_3 films and one foil-wrapped specimen. The results were similar to those previously reported for the 2000-hr test. The Gd_2O_3 film adherence was excellent on the two specimens with the normal ($\leq 32 \mu\text{in.}$) surface finish (A and B), whereas the film on the specimen with the rough finish ($> 100 \mu\text{in.}$) (C) was abraded from the peaks of the machining marks on the fuel. The Ni foil had

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Figure VII-9. NAA-120-4 Experiment Irradiation Data

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diffused into the fuel as was the case in the 2000-hr test. The appearance of the ceramic barrier was similar to that in the 2000-hr test, i.e. very dull green adjacent to the coated portions of the fuel and very dark and porous adjacent to the uncoated ends of the specimens. X-ray diffraction of the dark areas yielded even less SiO_2 than was observed in the 2000-hr test. The remaining tests at 1250 and 1400°F have accumulated 4608 hr at temperature.

The compatibility tests using discs of fuel and SCB-1-coated Hastelloy-N were terminated after 2715 hr to determine whether the fuel had adhered to the ceramic barrier. The stacks of discs readily separated upon removal from the fixtures, indicating no adherence. However, the ceramic barriers in the 1400 and 1500°F tests exhibited dull regions indicative of some reaction between the fuel and barrier. No significant differences were observed in the appearances of barriers adjacent to fuel with or without a Gd_2O_3 film. Specimens from the three tests are being mounted for metallographic examination.

B. IRRADIATION TESTS

1. NAA-120-4

Dimensional measurements on the central fuel sleeve and central portion of the support rod were completed. These measurements were obtained using a profilometer measuring device, dial indicator, and/or vernier calipers, depending upon specimen condition. Some of the fuel-sleeve diameter measurements were deduced from peak cladding measurements because of fuel condition. Also, some of the fuel-sleeve lengths were obtained by measuring the neutron radiographs for the same reasons. Diameter measurements on the support rods and inside diameter (ID) measurements on the fuel sleeves are being obtained on the fuel specimens during metallographic examination.

Density measurements were obtained on the central fuel sleeves and support rod; from these measurements the volume changes were calculated. Figure VII-9 shows the volume changes based on density measurements vs the percent diameter changes on the intact fuel sleeves. The data indicate a near isotropic growth, strongly related to the design operating temperatures.

Hydrogen analyses are nearing completion and the current data are shown in Table VII-4. The analyses thus far tend to agree with neutron radiographs and metallography.

TABLE VII-4
NAA-120-4 EXPERIMENT, H/Zr BASED ON HYDROGEN ANALYSIS*

Specimen No.	Center Sleeve				Center Support Rod			
	Pre-irradiation Measurement	Top	Center	Bottom	Pre-irradiation Measurement	Top	Center	Bottom
E-1	1.85	1.77 [†]	1.75 [†]	1.71 [†]	1.85	1.77 [†]	1.75 [†]	1.71 [†]
E-2	1.72	1.74	1.74	1.72	1.69	1.65	1.67	1.70
E-3	1.72	1.72	1.67		1.71	1.66		
E-4	1.86	Sample Lost	1.82	1.81	1.85	1.78	1.78	1.77
E-5	1.86	1.83	1.86	1.86	1.85	1.84	1.86	1.88
H-1	1.71	1.66	1.66	1.67	1.70	1.51	1.53	1.57
H-2	1.71	1.74	1.58	1.44	1.70	1.68	1.57	1.50
H-3	1.70	1.69	1.70	1.70	1.71	1.62	1.63	1.63
H-4	1.85	1.76	1.74	1.72	1.85	1.68	1.68	1.69
H-5	1.85	1.70	1.76	1.91 [†]	1.83	1.68	1.72	1.91 [†]
M-1	1.71	1.58 [†]	1.30 [†]	1.60	1.70	1.58 [†]	1.30 [†]	1.34
M-2	1.72	1.62	1.59	1.12 [†]	1.71	1.54	1.52	1.12 [†]
M-3	1.70	1.24	1.63 [†]	0.90 [†]	1.71	0.99	1.63 [†]	0.90 [†]
M-4	1.85	1.69	1.69	1.69	1.86	1.52	1.66	1.64
M-5	1.84	1.71	1.72	1.70	1.85	1.66	1.66	1.65
V-1	1.71				1.72			
V-2	1.71				1.71			

*Accuracy ± 0.02 H/Zr

[†]Sleeve and support of specimen analyzed together

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 H₂ content will be used to evaluate the effect of δ -to- β phase transformation after significant fission-product accumulation.

Included in the experiment is a parallel out-of-pile thermal test which will be used to differentiate between thermal and irradiation effects on fuel-element behavior. This test consists of four fuel elements subjected to thermal treatments which simulate as nearly as possible the thermal treatment of the in-pile elements.

During this quarter the irradiation experiment was discharged and the out-of-pile tests terminated. The radiation tests were ended early to provide immediate information on fuel-element behavior which might assist in the evaluation of the S8DR reactor behavior.

The operation of the radiation experiment during this report period was normal with no significant changes in operating parameters or experiment performance. Before discharge, the No. 4 experiment temperature was raised and the phase-change test started. Approximately two weeks of high-temperature operating time were accumulated on the phase-change test before discharge. The shutdown and discharge were performed routinely.

3. NAA-140-1

Work on this experiment was terminated. Effort is being redirected to shorter-term irradiation tests on Incoloy-800-clad fuel elements.

4. NAA-83

The primary objective of the NAA-83 irradiation experiment is to measure the effect of neutron irradiation on the high-temperature stress-rupture ductility of hydrogen-barrier-coated Hastelloy-N and Incoloy-800 cladding materials.

The second NAA-83 experiment, ORR-211, was discharged from the ORR on April 26, after 1178 hr of irradiation at the design temperature.

The third experiment, ORR-212, began irradiation on May 3 and had received 1700 hr of exposure as of July 18. Incoloy-800 specimen A-26 ruptured after 1200 hr at 1300°F and a stress of 9020 psi. Specimen A-47, also scheduled for in-pile rupture, had not yet failed. These data are compared with the literature (see Figure VII-10)^(VII-1, -2) and indicate that the processed Incoloy-800 is stronger than as-received mill-annealed material. Postirradiation testing of the remaining samples from the three experiments will clarify this conclusion.

TABLE VII-5
NAA-83 EXPERIMENT ORR-209 IN-PILE STRESS-RUPTURE RESULTS

Specimen No.	Material	Nominal Wall Thickness (mils)	Hoop Stress* (psi)	Time to Rupture (hr)	Strain-at-Failure (%)	Average Strain Rate (in./in.-hr x 10 ⁻⁵)
A-3	Incoloy-800	15	10,680	300.0	3.9	13
A-9		15	10,680	385.9	4.3	11
A-26		15	9,020	1200		
A-47		15	9,020	1771†		
B-1	Hastelloy-N	11	7,950	1012.8†	1.1†	1.1
B-3		11	7,950	842.3	2.0	2.4
C-6		15	8,750	386.5	0.4	1.1
C-11		15	8,750	508.7	0.9	1.7

* The hoop stresses are based on the wall thicknesses shown. They will be revised based on measurements of the ruptured specimens.

† Did not rupture.

TABLE VII-6
STATUS OF NAA-83 POSTIRRADIATION STRESS-RUPTURE TESTS

Specimen No.	Material*	Test Temperature (°F)	Pressure Differential (psi)	Hoop Stress (psi)	Expected Time to Rupture (hr)
C-1	Hastelloy-N	1400	259	4670	10,000
C-10			259	4670	10,000
C-3			304	5480	5,000
C-21			304	5480	5,000
A-5	Incoloy-800	1300	317	5710	5,000
A-6			317	5710	5,000
A-22			317	5710	5,000
A-30			317	5710	5,000
A-29			385	6930	2,000
A-41			385	6930	2,000

*Nominal wall thickness for all samples is 0.015 in.

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Figure VII-10. Biaxial Stress-Rupture Data for Irradiated As-Processed Incoloy-800 at 1300°F: Comparison of NAA-83 Test Results with Earlier Investigations

Bubble testing of the ruptured Incoloy-800 specimens from ORR-209 confirmed the locations of the ruptures. The actual strains-at-failure are presented in Table VII-5.

Preliminary dosimetry data indicate that both the ORR-209 and -211 were exposed to a thermal flux of $\sim 5 \times 10^{13}$ n/cm²-sec and a fast flux of $\sim 3 \times 10^{12}$ n/cm²-sec (>3 Mev). Detailed flux maps and fluence data are prepared for each experiment.

Both creep-test systems for out-of-pile testing in the Atomic International Hot Laboratory (AIHL) have been completed; one system is operational and one is being calibrated. The nine unirradiated Hastelloy-N control specimens have received 750 hr of the planned 1000-hr thermal soak at 1400°F. (The thermal soak is intended to duplicate the thermal history of the ORR-209 and ORR-211 specimens.) The four irradiated Hastelloy-N specimens from ORR-209 are being tested at 1400°F and six of the Incoloy-800 specimens from ORR-211 are being tested at 1300°F. The specimen numbers, pressures, and expected times-to-rupture are shown in Table VII-6.

Visual examination, gamma scanning, and dimensional profiling of all specimens from the ORR-209 and ORR-211 experiments are complete.

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TIME, HR

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Figure VII-11. S8DR Preproduction Fuel-Element Performance Data

C. FUEL-ELEMENT TESTING

1. Preproduction Element Performance Tests

The endurance testing at 1400°F continued for the six preproduction elements. Because of the limited number of retorts now available, these six fuel elements are clustered in groups of three in two retorts. The fuel-element permeation rates are being computed as a ratio of the total H₂ flow in the retort based on individual permeation rates as measured in the prior 10,000-hr endurance test. These six fuel elements have completed more than 5000 hr in this extended endurance cycle and now have accumulated more than 16,000 total hours at 1400°F (see Figure VII-11).

2. S8DR Element Performance Tests

The non-nuclear performance testing of S8DR normal fuel elements continued. Phase-I fuel elements, representative of start-of-core production elements, have completed more than 11,000 of the scheduled 20,000 hr of endurance testing at 1400°F; Phase-II elements, midcore production, 10,600 hr; and Phase-III, end-of-core production, 8500 hr. Phase-V elements, which are being endurance-tested at 1300°F, have completed 10,000 hr.

Figures VII-12, -13, and -14 compare the performance parameters (i.e. hydrogen-permeation rate, hydrogen-barrier parameter, and hydrogen loss) of the S8DR production elements with the S8DR preproduction and the pre-S8DR "Special Seven" elements. These time-increment graphs show that the overall hydrogen-retention performance of the S8DR elements is in general comparable to that of the preproduction elements and far superior to that of the Special Seven.

The hydrogen-retention parameters for the ten Phase-V elements being endurance-tested at 1300°F are plotted in Figure VII-15. The permeation-rate data are normalized to 1400°F, 2 atm pressure for direct comparison with other S8DR tests. Results to date indicate that the performance during 1300°F endurance is somewhat better than for 1400°F tests, as shown by comparison of permeation rates and the hydrogen-barrier performance parameter.

All of the S8DR fuel elements are subjected to two thermal cycles at 2000-hr intervals in the endurance test program. Periodically, two elements are removed from test for detailed destructive test and evaluation. To date,

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Figure VII-12. S8DR Production Fuel-Element Hydrogen Permeation Compared with "Special 7" and Preproduction Fuel Elements

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TIME, HOURS X 1000

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Figure VII-13. S8DR Production Fuel-Element Hydrogen Barrier Performance Compared with "Special 7" and Preproduction Fuel Elements

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Figure VII-14. S8DR Production Fuel-Element Total Hydrogen Loss Compared with "Special 7" and Preproduction Fuel Elements

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four elements each have been removed from Phase I, II, and III tests; two were removed after 4000 hr and two upon completion of the 8000-hr endurance tests.

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Figure VII-15. S8DR Production Fuel-Element Performance Parameters of Phase-V Endurance Tests at 1300°F Compared with 1400°F Tests

VIII. AEROSPACE SAFETY

The measurement of the prompt temperature coefficient of the SNAP TRAN reactor with varying concentrations of gadolinium-155 has been completed. The configurations studied are listed in Table VIII-1.

TABLE VIII-1
PROMPT TEMPERATURE COEFFICIENT MEASUREMENTS

Configuration	Loading	Gadolinium Σ_a	Transient Numbers
1	Type-I sleeves	0	1 to 18
2	Type-I with void in center and Type-IV around void	-	19 to 23
3	Type-II sleeves	0.073	24 to 33
4	Type-I and -IV mixed	0.118	34 to 39
5	Type-III sleeves	0.150	40 to 46
6	Type-I sleeves	0	47 to 53

Prompt temperature coefficients of approximately -0.05% /°F were obtained in the measurement of Type-III sleeves. The decision was made not to measure Type-IV sleeves as they could be very close to a zero temperature coefficient.

In each configuration a series of transients was conducted. Typically, they consisted of subprompt critical transients of 60 and 80¢ followed by a series of prompt critical transients of \$1.20 at progressively higher temperatures. With the Type-III sleeves it was necessary to limit the transients to the range of 50 to 60¢ because of the small amount of reactivity feedback.

The power and temperature data from nuclear detectors and energy probes were recorded as analog signals on FM magnetic tape during the transients. After the transient the data were digitized and recorded on a digital tape.

The data from the transients were processed using the STAC kinetics analysis code, and yielded a mean temperature coefficient for each transient. These were correlated with the mean temperature, defined as the middle of the temperature range, which produced analyzable data. The data points corresponding to each analyzable transient are shown in Figure VIII-1, together with

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Figure VIII-1. Temperature Coefficient of Reactivity from the Intrinsic Water Subcriticality Experiments

a least-squares fit for each type of sleeve. A curve projected for the Type-IV sleeves by linear extrapolation is also shown, although no transients were run with those sleeves.

The data show a uniformly decreasing temperature coefficient with increasing Gd loading. The rate of decrease is higher at high temperature, so that Type-IV sleeves would have had a zero coefficient at about 1100°F (see Figure VIII-2).

As a check, some of the Type-I transients were done at the beginning of the test series and some at the end. The results were indistinguishable, indicating that no observable trend was present due to degradation of the instrumentation or alteration of the properties of the fuel.

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Figure VIII-2. Temperature Coefficient vs Gadolinium Loading

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