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SNAP REACTOR PROGRAMS

PROGRESS REPORT

NOVEMBER 1968 - JANUARY 1969

(Title Unclassified)

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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

1. S8DR Test

The installation of the fueled reactor and reflector assembly into the vacuum vessel was completed. As a part of the installation, the remote reflector removal and installation operation was successfully demonstrated.

The dry and wet critical tests in the vacuum vessel were performed as were the five automatic reactor startups.

NaK pumps throats in both the primary and secondary systems were replaced. Chloride stress-corrosion problems were encountered with these new pumps and fabrication of replacement pumps was begun.

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A summary of S8DR operating statistics up to January 30, 1969 is as follows:

Reactor critical	-----	504 hr
Reactor at power (>525 kwt, >1280°F)	-----	123 hr
Energy release	-----	107,198 kwh

2. Reference ZrH Reactor Engineering

The preliminary design of the Reference ZrH Reactor continued during the report period with several design modifications made to simplify the configuration.

A detailed study was performed to determine the dynamic loads which would be imposed upon the reactor system by a Saturn launch. Although the results of the study have not been completely evaluated, it appears that the design requirements imposed by the NASA 417-2 Specification are unduly severe.

3. Reactor - Thermoelectric System Engineering

The overall system layout and system performance parameters were updated during the quarter, and additional conceptual design studies were performed.

Optimization studies were conducted to determine what ratio of direct-to-scattered dose in the space station would result in a minimum shield weight. It was found that the minimum-weight shield had inadequate shielding in the transverse direction for protecting the astronauts during rendezvous and for preventing fogging of film in proposed test modules operating near the station. The selected conceptual shield design represents a 500-lb weight penalty over the minimum-weight configuration.

A preliminary safety evaluation of the low-orbit space-station mission was conducted. The methods of analysis used in the study were based on those developed for the SNAP 10A safety analysis. The input parameters used were those peculiar to the space-station mission. This preliminary assessment of the risks presented by this mission showed them to be similar in magnitude to those posed by the SNAP 10A launch.

4. Thermoelectric Converter Development

The development of a thermoelectric (TE) power-conversion system using tubular lead-telluride converter modules is proceeding at Westinghouse Astro-nuclear Laboratory (WANL) under subcontract to AI.

The endurance testing of three preprototype modules continued during the quarter, while a detailed destructive examination was made of a fourth preproto-type module to evaluate the cause of the greater than expected loss in perfor-mance. It was found that the N-type PbTe washers had become contaminated by P-type material due to mass transport around and/or through the mica washers which separate them. The design of the inner-conductor-ring was changed to provide a positive seal by the mica between the N- and P-type PbTe materials. The destructive examination of one module indicated that a positive seal was attained, but thermal-endurance testing will be required to demon-strate the elimination of the loss in performance.

The preliminary design of both the converter module (four-pack) and the converter assembly (six converter modules) was established. The preliminary thermal, hydraulic, and stress analyses indicate that all of the design require-ments can be met.

5. Component Development

The design-verification testing of three S8DR actuators and four sets of bearings continued during the quarter. The testing of position sensors and limit switches was completed and the examination of selected units has been started.

The evaluation-testing of nickel-clad silver and BeO-hardened copper con-ductor materials for actuator coils is continuing with over 8500 hr at 1200°F and 5000 hr at 1400°F. The testing of magnetic materials has passed 10,000 hr.

Parametric load testing of the prototype high-temperature bearing was conducted to establish the effect of shaft misalignment at several temperatures. The bearing friction associated with a 2° misalignment was not significantly different from the friction levels with a straight shaft.

6. Fuel Development

The eight 15-mil Hastelloy-N-clad developmental fuel elements have completed 5000 hr of out-of-pile testing at 1400°F. Five of these elements have metal barriers (304-SS or Ni) between the fuel and coating and are showing about half the leakrate of the three control elements.

Analytical and experimental studies were made to evaluate the effects of launch vibration on alternate fuel-element configurations. These studies indicated that the response of the fuel elements can be significantly reduced if the elements are attached to the gridplates and a spacer is placed inside the cladding tube to prevent relative motion between the fuel rod and the cladding. Additional effort is planned to develop a satisfactory spacer, probably of the column-buckling type.

Twelve Incoloy-800-clad fuel elements have accrued over 2000 hr of endurance testing and continue to have satisfactory performance.

Biaxial creep-rupture tests of Incoloy-800 have been begun in NaK and in vacuum. The present data are somewhat suspect due to oxygen contamination of the NaK, but indicate a fairly close correlation between the rupture times of the as-received material and the tests in NaK. The tests in vacuum are proceeding longer than predicted, based on the as-received data.

The NAA-120 irradiation experiment was shipped to Atomics International, and the hot cell examination was started. The NAA-121 experiment continued to operate normally at Hanford. The detailed design analyses of the NAA-140 experiment continued during the quarter.

The test specimens for the NAA-83 cladding irradiation test were fabricated and shipped to the Oak Ridge National Laboratory (ORNL). The first set of test specimens was inserted in-pile in mid-January and will be irradiated for about 1000 hr.

7. Aerospace Safety

Ni foils containing Gd_2O_3 were obtained for use in the intrinsic subcriticality experiments. Foils containing three different concentrations of Gd-155 were

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fabricated into sleeves which could be slipped over the SNAPTRAN fuel elements (one set was also fabricated without gadolinium). The reactivity worth of the varying amounts of Gd-155 was measured in a water-immersed core. Subsequent experiments will establish the worth in the operational configuration and will evaluate the effect of the Gd on the reactor temperature coefficient.

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II. SNAP 8 DEVELOPMENTAL REACTOR

A. INTRODUCTION

The installation of the fueled reactor and reflector assembly into the vacuum vessel was completed. As a part of the installation, the remote reflector removal and installation operation was successfully demonstrated.

The dry and wet critical tests in the vacuum vessel were performed as were the five automatic reactor startups.

NaK pumps throats in both the primary and secondary systems were replaced. Chloride stress-corrosion problems were encountered with these new pumps and fabrication of replacement pumps was begun.

A summary of S8DR operating statistics up to January 30, 1969 is as follows:

Reactor critical	-----	504 hr
Reactor at power (>525 kwt, >1280°F	-----	123 hr
Energy release	-----	107,198 kwh

B. INSTALLATION OF CORE AND REFLECTOR INTO VACUUM CHAMBER

Following completion of the facility and heat-transfer system checkout test of the last report period, the NaK was drained from both the primary and secondary NaK systems in preparation for installation of the fueled reactor core. The dummy reflector halves, which had been installed on the mockup core vessel to produce a representative thermal environment during checkout, were removed. The S8DR reflector halves were then installed and operationally checked. Also, during this phase a complete checkout of the manual and automatic control systems was performed. The checkout was made at this time so that proper operation could be demonstrated before mounting the reflectors on the fully fueled core. Upon completion of the checkout, the reflector halves were removed from the mockup core installation and temporarily stored.

The mockup core and reactor inlet/outlet piping were removed from the vacuum vessel and cleaned of residual NaK by steam cleaning with de-ionized water. The inlet and outlet lines were removed and then mounted on the fueled

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Figure II-1. Core and Reactor Piping Assembly

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reactor. Figure II-1 shows the core and reactor piping assembly ready for installation into the vacuum vessel.

After completion of the core installation into the vacuum vessel, but before reflector installation, NaK was loaded into the system and circulated at 1300°F for ~150 hr. This test was performed as a final check on the quality of the core vessel and piping seal welds. A visual examination made following completion of the NaK circulation showed these welds to be satisfactory.

The reflector assembly was then installed on the core vessel in the vacuum vessel. Demonstration of the remote-handling techniques for reflector removal and installation was successfully performed during the final fitup of the reflector assembly. Figure II-2 shows the reflector halves mounted on the remote handling fixture as it was being prepared for installation into the vacuum vessel. Final dimensional clearances between the reflector assembly and the core vessel were obtained. The measured values are presented in Table II-1. The final reactor assembly installation inside the vacuum vessel is shown in Figure II-3.

The reflector assembly was checked out and drum scram times measured. All scram times were well within safety requirements. During checkout the No. 5 actuator did not perform satisfactorily. The problem was traced to an improper brake current setting in the drum controller. After adjustment to the correct value the actuator performed satisfactorily.

The vacuum vessel lid was welded closed following completion of the critical testing. A pressure proof test at 42 psig and a leakrate test at 10 psig were performed. These tests were carried out identically to the ASME code certification requirements and verified the Safety Analysis Report containment requirements.

C. VACUUM VESSEL CRITICAL TESTING

A set of dry and wet critical tests was performed to determine reactivity coefficients and control-drum worths in the vacuum vessel. A summary of these measurements is given in Table II-2. The only parameter outside the range of predicted values is the control-drum worth when the neighboring drums are in the "full-out" position. This deviation from expected value is attributed to an underestimate of the drum interaction effect for the S8DR configuration.

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Figure II-2. Reflector Halves Ready for Installation

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TABLE II-1
FINAL REFLECTOR CLEARANCES

Vertical Position	Axis			
	+Y	-Y	+Z	-Z
Top	0.073	0.079	0.080	0.080
Middle	0.073	0.073	0.73	0.080
Bottom	0.072	0.073	0.072	0.074

Figure II-3. Final Reactor Assembly Installation 7759-1218CN

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TABLE II-2

S8DR VACUUM VESSEL CRITICAL TEST

Parameter	Predicted	Measured*
Vault worth, ϵ	$+20 \pm 20$	+35
NaK worth, ϵ	$+30 \pm 30$	+20
Isothermal temperature coefficient, $\epsilon/^{\circ}\text{F}$	-0.19 to -0.22	-0.17 to -0.23
Shutdown margin, \$	-6.40 to ± 2.00	-6.10 ± 1.00
Excess reactivity, \$	14.40 ± 3.00	12.70 ± 1.00
Drum worth, \$:		
Neighbors full-in	3.90 ± 0.60	3.80 ± 0.15
Neighbors full-out	3.10 ± 0.50	2.40 ± 0.10

*Measured results based on preliminary evaluation of control-drum calibrations

TABLE II-3

PRIMARY SYSTEM NaK-SAMPLE RESULTS

Element	Specification Value (ppm)	Sample (12/11/68) (ppm)
Aluminum	20	20
Boron	5	5
Barium, Cesium	-	<5
Cadmium	1	<1
Cobalt	5	<1
Chromium	10	7
Iron	15	5
Lithium	10	<1
Manganese	1	1
Nickel	10	1
Rubidium	150	30
Silicon	10	75
Uranium	-	<1
Zirconium	100	<5

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A NaK-coolant sample was obtained during wet critical to provide a reference level from which future samples can be compared. Table II-3 presents both the specification limit and quantity found in the sample for the key impurities which will be monitored closely throughout operation.

During critical testing a number of problems arose in the operation of the full-out and "clutch-rewound" limit switches. The faulty operation of these switches was traced to either low actuator current or mechanical interference with the snubber when in its full-out position. These difficulties have been corrected by increasing the actuator current to the proper level and by special clutch-rewind procedures.

D. AUTOMATIC STARTUP TO OPERATING TEMPERATURE (Pre-Mercury-Injection Phase)

Five automatic startup tests were conducted on January 10 through 15, 1969. A typical power transient above 1 kw is shown in Figure II-4, while the reactor and intermediate-heat-exchanger coolant temperatures are shown in Figure II-5. A detailed analysis of these transients was started and the following preliminary estimates of basic controlling parameter values have been obtained:

- 1) The magnitude of the fuel and lower-grid temperature coefficients of reactivity are estimated to be higher than predicted from theoretical calculations, while the upper-grid coefficient effects are estimated to be negligible. Partial analysis of test results are shown in Table II-4.

TABLE II-4
COEFFICIENTS OF REACTIVITY

Location	Coefficient ($\beta/^\circ\text{F}$)	
	Current Estimate	Predicted
Fuel	-0.10 to -0.11	-0.07
Lower grid (inlet)	-0.08 to -0.10	-0.06
Upper grid (outlet)	0.0	-0.06

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Figure II-4. S8DR Automatic Startup, Reactor Power

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Figure II-5. S8DR Automatic Startup, Reactor Temperature

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- 2) The fuel-to-coolant heat transfer (UA) for fuel temperatures below 500°F is by radiation in a near-vacuum due to the absence of free hydrogen anticipated in the fuel-to-clad gap from fuel assembly procedures. The overall UA variation over the range of 200 to 1400°F fuel temperature is shown in Figure II-6 on the basis of preliminary test analyses. Theoretical support for values of the UA curve below 500°F has been obtained from computer model calculations showing the dependence of the damped power oscillations shown in Figure II-4 on the values of UA at low fuel temperatures.

The reduction and interpretation of startup test data is continuing and will be used to modify the computer model representing the S8DR system.

E. NaK ELECTROMAGNETIC-PUMP THROATS

New NaK-pump throats were installed in both the primary- and secondary-coolant systems. Spare throats from the S8ER program had previously been installed as an interim measure when a leak developed in one of the throats originally installed for the S8DR test. Lack of stress-relieving of the rectangular tubing used in fabricating the original S8DR pump throats led to a stress-corrosion cracking situation.

The newly installed S8DR pump throats were operated for approximately 200 hr as a part of the overall facility shakedown and checkout test. A visual examination of the primary-pump throat following this operation revealed two localized areas of deposit in the region of the outermost rib which encloses a copper bus bar. Chemical analysis of the deposit revealed the presence of predominantly copper and silver. No evidence of sodium or potassium was found. A visual examination of the area revealed a small crack crosswise in the rib. No deposits were found on the secondary-pump throat upon examination for similar deposits. The suspect throat in the primary system was removed and replaced with a spare of identical design and fabrication history.

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Figure II-6. S8DR Core Heat Transfer as a Function of Fuel Temperature

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III. REFERENCE ZIRCONIUM HYDRIDE REACTOR ENGINEERING

A. REACTOR DESIGN

The reactor preliminary design activity was directed toward the completion of major subsystem assembly drawings. This included assemblies of the internal and external reflector, core vessel, unitized core, and reactor vessel with control drums. The current reactor vessel and control-drum assembly is shown in Figure III-1. This represents a modification of the earlier conceptual design^{(1)*} in the region of the outlet gridplate-to-control-drum drywell region. Reactor startup kinetic studies illustrated the desirability of a negative outlet-coolant temperature coefficient of reactivity to reduce startup time. Experimental data from the S8ER and S8DR indicate that a significant temperature coefficient of reactivity is associated with the expansion of the reflector assembly relative to the core. It was therefore desirable to couple the reflector motion to the outlet-coolant temperature. This was accomplished by using a bolted connection between each drywell and the gridplate. The drywell support-plate is slotted to permit the desired radial motion. Stability of drywells for launch shock and vibration is provided by the close fit between the reflector end-pins in the expansion slots and the fit of support-plate to gridplate.

Another modification made to reduce resistance to radial drywell motion was to eliminate the flange on the core-can by welding the can directly to a machined lip on the outlet gridplate. The final assembly of the unitized core is accomplished by welding the core can to the inlet gridplate after the fuel elements are in place.

The lower-head configuration was changed to an ASME standard flanged and dished configuration. The divider between inlet and outlet plenums is of spun light-gauge metal and is welded to the lower head. The flanged connection at the gridplate is slotted to reduce thermal stresses.

The control-drum actuator and upper-bearing configuration shown in Figure III-1 is that associated with shadow-shielded operation. The integration problems associated with this configuration were considered to be the more severe and are being investigated before the 4π -shielded case. Preliminary studies are in progress to establish the actuator, shaft, and bearing configuration for a 4π -shielded system.

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

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Figure III-1. ZrH Reactor Model II Reference Design

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ATOMIC INTERNATIONAL
ZrH REACTOR - MODEL II
REFERENCE DESIGN
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B. REACTOR VESSEL CONFIGURATION

To improve stress levels and facilitate easier and lower cost assembly and to effect a reduction in the overall reactor dimensions, six variations of the lower-plenum inlet and outlet areas were reviewed (see Figure III-2).

Each of the configurations L, M, N, O, P, and Q had varying stress levels. Each was analyzed on the basis of 35-psi internal pressure and temperatures of 1300°F for the center-outlet plenum, 1100°F for the outer-inlet plenum, and 1200°F for the common wall. The complete reactor vessel was used in the analysis.

The best features of P and Q were combined with some further modifications to evolve the final configuration shown in Figure III-3. The close approximation of the evolved design to a standard ASME flanged and dished head permitted a standard configuration to be selected. Maximum thermal stress is at the temperature gradient near the junction of the flow divider and the head, and is approximately 22,000 psi. Maximum pressure stress is at the small radius of the torispherical head, and is approximately 9000 psi. The flow divider has been thinned to offer little restraint to the thermal stresses. It will provide some lateral support to the core but all axial core restraint will be provided by the drywells.

C. LAUNCH ENVIRONMENT ANALYSIS

To take advantage of North American Rockwell Space Division's (SD) experience in the field of launch dynamics gained from the design, fabrication, and launch of the Apollo system, Space Division was requested to conduct a realistic launch environment analysis for typical mission configurations. The resulting analysis included launch vibration and acceleration loads for four typical configurations of the ZrH system launched on the Saturn-IB or Saturn-V. Previously the reactor system dynamic loads were based on the launch environmental design criteria provided in the NASA 417-2 specification, Revision A, and issued by the NASA Lewis Laboratory. This specification was reviewed by Atomics International and SD personnel and was not considered representative of anticipated launch conditions for the ZrH system.

Table III-1 shows the launch loads specified in the NASA 417-2 specification as compared to the SD results.

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Figure III-2. Reactor Vessel
Evolution

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TABLE III-1
COMPARISON OF LAUNCH ENVIRONMENTAL REQUIREMENTS,
INPUTS TO THE NUCLEAR POWER SYSTEM

	NASA 417-2	Saturn-V	Saturn-IB
Shock:			
Magnitude, g	18	-	-
Half-sine duration, msec	8	-	-
Number	18	-	-
Vibration:			
Magnitude, g peak	5	(b)	(b)
Sweep, cps	16-2000	-	-
Acoustic Noise:			
Magnitude, db peak	138	135	135
Duration, sec	300	7	7
Peak shroud response, g _{rms}	-	17.6	(g)
Frequency at peak, cps	-	400	(g)
Acceleration:			
Axial positive, peak g	7	5	4.4
Duration, sec	300	(c)	(c)
Axial positive, average g	-	1.5	2.0
Total burn time, sec	-	900	625
Axial negative, g	3	1	0.8
Duration, sec	300	(d)	(d)
Transverse, g	4.5	1	1.25
Duration, sec	300	(d)	(d)
Random Vibration:			
Peak spectral density, g ² /cps	0.64 ^(e)	0.0085 ^(f)	(g)
Frequency band at peak, cps	200-700	80-160	(g)

(b) No requirement for sinusoidal vibration tests.

(c) Peak acceleration occurs at end of first-stage burn; very short duration.

(d) Occurs during staging; less than 5-sec duration.

(e) Specified for component tests; no system requirement.

(f) Reactor-shield response for Saturn-V launch.

(g) Probably same as Saturn-V.

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Figure III-3. Lower-Plenum Reactor
Vessel

With the limited structural component and functional data available for the ZrH system, the SD results must be considered preliminary. The actual environment is a function of the final design.

The dynamic loads, both acoustic and mechanical, depend on the system and missile geometry and flight plan. Since the SD study required a prediction of the booster engine noise at liftoff and the aerodynamic noise around the shroud during the atmospheric phase of flight, the shroud geometry and noise-reduction characteristics were assumed based on the Apollo SLA and the OGO shroud.

IV. REACTOR - THERMOELECTRIC SYSTEM ENGINEERING

Effort during this reporting period has been directed toward establishing the conceptual design of the Reactor-Thermoelectric System for a space station. Evaluation and design tasks were performed in the areas of system design and performance, shield weight optimization, radiator design, system control, and safety.

A. SYSTEM DESIGN AND PERFORMANCE

A layout of the current reference design reactor - TE power system is shown in Figure IV-1. The system, designed for 25 kwe, consists of a 4π -shielded U-ZrH NaK-cooled reactor with compact lead-telluride TE power conversion. The power-conversion system (PCS) components are located in a gallery between a lower and upper shadow shield. Heat is rejected by the radiator having an effective area of 1280 ft². During launch and nonoperating periods the power system is located within a heat-shield shroud. During startup and full-power operation the system is extended exposing the radiator to space.

A system schematic of the powerplant is shown in Figure IV-2. It is a two-loop system, each loop having four parallel flow-circuits. The PCS components for each flow-circuit are located in the gallery, and consist of the main power converter and a pump converter which furnishes electrical power to two series-connected EM pumps. A weight breakdown for the system is given in Table IV-1.

B. SHIELD-WEIGHT OPTIMIZATION

Shield-weight optimization calculations were made using the following criteria:

- 1) Separation distance = 117 ft.
- 2) Doseplane diameter at 117 ft = 46 ft.
- 3) The doserate from the reactor should be less than 25 rem/yr (or 2.85 mrem/hr).
- 4) Side-doserate <200 rem/hr at 100 ft.
- 5) Reactor thermal power = 600 kw.
- 6) One Command Service Module (CSM) docked in the front docking area (see Figure IV-3).

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. URANIUM ZIRCONIUM HYDRIDE, NaK-COOLED REACTOR	
. LEAD TELLURIDE THERMOELECTRIC CONVERTER	
. POWER OUTPUT	25kwe
. SYSTEM WEIGHT	19,880 lbs 795 lbs/kwe
. EFFECTIVE RADIATOR AREA	1,280 ft ² 51 ft ² /kwe

RADIATOR CROSS-SECTION

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Figure IV-1. Reactor - Thermoelectric Power System

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Figure IV-2. ZrH Reactor - Thermoelectric System Schematic
(System Efficiency = 4.3%)

TABLE IV-1
REACTOR — THERMOELECTRIC POWER SYSTEM
WEIGHT SUMMARY

Item	Weight (lb)
Reactor	1,680
Thermoelectric converters	1,140
Pumps	280
Expansion compensators	210
Piping	670
Radiator	2,210
Structure	1,120
Electrical wiring	280
Radiation shield	<u>12,290</u>
Total	19,880

Figure IV-3. Space Station Radiation Environment Due to Nuclear Reactor

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The geometric criteria relate to the space-station configuration shown in Figure IV-3, and the dose criteria come from an evaluation of dose to astronauts and film during the time on the space station and during rendezvous maneuvers. In addition the side-doserate is limited by the allowable radiation to film on remote experimental modules operating 2 to 10 miles from the station.

The doserate in the space station is due to direct radiation, and from neutrons and gamma rays leaking from the side shield and scattering off sections of the CSM not within the shadow shield cone. The scatter-doserate is thus a function of the side-doserate. The shield weight was optimized with respect to the fraction of doserate from direct radiation, keeping the total doserate fixed at 2.85 mrem/hr. A minimum weight is obtained when the side-doserate is 350 rem/hr at 100 ft. To meet the criteria of 200 rem/hr at 100 ft, the side shield thickness was increased, resulting in a 500-lb weight penalty from the minimum-weight case.

Figure IV-4 shows the shield configuration and the materials used for the various sections of the shield. This shield was optimized to give a direct-doserate of 1.82 mrem/hr at 117 ft and 200 rem/hr at 100 ft on the side. The CSM scattered-doserate was calculated to be 1.03 mrem/hr, resulting in a total doserate of 2.85 mrem/hr in the space station.

Figure IV-4. 4π Shield with Structure

C. RADIATOR DESIGN

Design studies were made to evaluate potential radiator materials. Three fin-armor materials were initially considered: aluminum, Lockalloy (Be-Al alloy), and expanded pyrolytic-graphite. The use of aluminum results in the heaviest radiator; however, fabrication techniques are well known and little development is expected. Lockalloy has the potential of reducing the radiator weight by about 500 to 600 lb over aluminum, primarily through the utilization of its structural capabilities. The material is a relatively new development; consequently, fabrication process development can be expected. Expanded pyrolytic-graphite was found to provide only a marginal weight savings and, in addition, had poor strength and fabrication characteristics. Consequently, it is no longer considered a prime material candidate.

Conceptual designs of an aluminum and Lockalloy radiator were performed. The aluminum radiator design, for example, uses a cylindrical structure of titanium alloy, stiffened by bulkheads and stringers, as the primary structure. Each length of fin is separately supported and an anchor at the midlength of the fin provides axial location. Flexible clips permit thermal expansion of the fin relative to the structure. The structure also contains guide rails for deployment of the power system.

D. SYSTEM STARTUP STUDIES

A digital computer simulation for the reference ZrH reactor was prepared to study the dynamics of system startup. The model included the complete primary loop with the TE converters and pumps. This simulation was used to calculate transients during the initial phases of system startup before powerplant extension with several sets of reactivity coefficients, reactivity rates, and initial NaK-flowrates. The results showed the feasibility of a "bootstrap" operation of the TE pumps from prestart flowrates in the 5 to 10% range. It was determined that a negative core-outlet temperature coefficient could significantly reduce the required startup time. This prompted a redesign of the core-outlet-plenum configuration to accomplish the negative-reactivity feedback.

The fuel-to-coolant heat-transfer coefficient used in the analytical model was subsequently revised to include the fuel-to-cladding heat-transfer coefficient variation observed in the experiments with S8DR. The effects of this heat-

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transfer coefficient variation on the startup transients for the reference ZrH system is currently being evaluated.

E. CONTROL-SYSTEM DESIGN

A reference control-system concept was established which would provide the logic and timing functions to perform reactor startup, long-term control, and reactor shutdown. The control-system output provides the power to drive the ten actuator motors which control the position of the ten reactor control drums. Only one drum is stepped at any one time, with all ten stepped in sequence to maintain equal drum position. The three basic control parameters are: (1) the Operate/Shutdown Command, (2) the powerplant extended or retracted position, and (3) the plant electrical power output and/or reactor temperature. Preset signals determine the timing rate at which the reactor is brought to subcritical, through critical, and up to full power.

The block diagram of the control system is shown in Figure IV-5. Also shown are the estimated electrical power requirements, volume, display area, and weight. The main features of the system are its built-in redundancy, self-checking circuits, and maintainability by replacement of five basic modules.

The complete system consists of nine replaceable modules (five different types):

- 1) Two logic and timing units, one being active and the other a wired-in standby unit.
- 2) One selector to select one of the above two units.
- 3) Three independent power-switch units, each being capable of handling the full-power load to the drum actuators.
- 4) Two drum-position-indicator units, one being an active unit and the other a wired-in standby unit.
- 5) One checkout unit, used to determine which of the other units is faulty.

● REQUIRED POWER

- CONTROL SYSTEM
- ACTUATORS
- PUMPS

ELECTRONIC PACKAGE

- VOLUME
- DISPLAY AREA
- WEIGHT

PRESTART STARTUP OPERATION

-	85 w	85 w
-	400 w-hr	-0 w
550 w	550 w	-
	800 in. ³	
	80 in. ²	
	40 lb	

● FEATURES INCLUDED

- BUILT-IN REDUNDANCY
- SELF-CHECKING CIRCUITS
- MAINTAINABLE BY REPLACEMENT OF MODULES (5 TYPES)

8-N26-154-21

Figure IV-5. Control System

Features of the control system are:

- 1) The power switches are self-checking on every drum step and indicate their redundancy status.
- 2) The active logic and timing module has self-checking of all major circuits with "GO," "NO GO," and "REDUNDANCY LOSS" indications. The standby unit may be completely checked out, on command, for all input signals. The standby unit may be substituted for the active unit on command.
- 3) The standby drum-position-indication unit may be, on command, substituted for the active unit.

Also, in conjunction with the control-system design a circuit which computes TE open-circuit voltage by measuring the TE load current and load voltage was breadboarded. Since TE open-circuit voltage relates to temperature, this parameter could be used for reactor temperature control.

V. THERMOELECTRIC CONVERTER DEVELOPMENT

(Prepared by Westinghouse Astronuclear Laboratory
under Subcontract N854-0051)

A. REACTOR POWER MODULE

1. Performance of Preprototype Modulesa. TEM-9AD SN-1 Static Test

A 15-in. long 9U-series module fabricated with TEGS-2P P-type lead-telluride and GE N-type lead-telluride was placed on test during the quarter. This module, designated TEM-9AD SN-1, uses W-foil barriers between the TE material and the Fe hot and cold conductor rings. After 4044 hr of life testing at $\bar{T}_H = 1000^\circ\text{F}$ and $\bar{T}_C = 392^\circ\text{F}$ TEM-9AD SN-1 showed a circuit efficiency of 5.67% compared to 6.26% at BOL. Internal resistance has increased 31 m Ω since the module was placed on test (from 161 to 192), while the open-circuit voltage increased slightly (12.2 to 12.4 volts).

It was first pointed out in the previous progress report⁽²⁾ that effective Seebeck data for TEM-9AD SN-1 were higher than predicted. Recently a better temperature-averaging method was included in DAHA, the data handling code, and it was discovered that for certain distorted axial temperature profiles, past average temperatures (hence effective Seebeck) were slightly in error. Using the new method of determining average temperatures, effective Seebeck coefficient data for TEM-9AD SN-1 are in agreement with predicted values within experimental accuracy.

TEM-9AD SN-1 was removed from test in December to allow rewiring of the stand and has not been tested during the past month. This module is scheduled to be placed back on test in February.

b. TEM-9U SN-7, -8, -9, -10 Endurance Test

Testing of three TEM-9U-series preprototype modules (SN-8, -9, and -10) continued during the past quarter. Power-output data from these three modules along with data from TEM-9U SN-7, which was previously removed from test and destructively examined, are given in Figure V-1. The results of the SN-7 destructive examination are discussed in another part of this section.

Figure V-1. A Comparison of Electrical Power Output from Four Preprototype Modules Operated Between $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$ (nominal)

TEM-9U SN-8 was operated at simulated two-loop reactor system temperatures, $\bar{T}_H = 1125^\circ\text{F}$ (607°C) and $\bar{T}_C = 570^\circ$ (299°C), for 569 hr on a WANL static-test stand before removal for shipment to AI. An additional 3384 hr of testing has been accrued in the two-loop liquid-metal system. Similarly TEM-9U SN-10 was operated at two-loop temperatures at WANL for 1989 hr before being shipped to AI, where it has accrued an additional 1032 hr in the two-loop liquid-metal system.

The reduction of power output measured for TEM-9U SN-10 immediately after being placed on loop test (from 65.7 watts to 57.5) was caused primarily by two factors relating to module instrumentation. An increase of seven $\text{m}\Omega$ (from 98.0 to 104.3) in internal resistance was observed, caused by a change in electrical wiring. The electrical power pins were included as part of the load-resistance during static testing and are at present considered part of the circuit-resistance on loop test. An investigation has been undertaken to determine the cause of the high electrical-pin resistance. The bulk electrical-resistance of the electrical pins in the circuit should be approximately one $\text{m}\Omega$ at BOL. In

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addition TEM-9U SN-10 was not operated under matched-load resistance immediately after being placed on loop test. This situation has been remedied, causing the increase in power output of TEM-9U SN-10, as seen in Figure V-1.

TEM-9U SN-9 has remained on static test at WANL. Figure V-1 shows that the thermal cycling performed after 2369 hr of testing has had essentially no effect on subsequent module performance.

c. TEM-9U SN-9 Thermal-Cycle Tests

Qualification of a TE generator for the reference 25-kwe reactor - TE system design requires that the module withstand 50 thermal cycles. Before the thermal-cycling of prototype tubular modules two preprototype tubular modules have begun a series of thermal-cycle tests.

The first such module to be cycled, TEM-9U SN-9, was tested in a static test stand at WANL. In each cycle the main heater power was reduced, cooling the module from standard hot-side test conditions, $\bar{T}_H = 1125^\circ\text{F}$, down to $\bar{T}_H = 500^\circ\text{F}$ at a rate of $20^\circ\text{F}/\text{min}$. After a 58-min soak period the module was reheated to standard conditions, again with a ramp rate of $20^\circ\text{F}/\text{min}$. During the tests the guard heater power and test-stand cooling-fan speed were held at constant levels. Including a 58-min soak period at standard operating conditions ($\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$), the duration of each complete thermal cycle was three hr.

A first set of ten consecutive thermal cycles was run after the module had accrued 2368 hr at life-test conditions, followed by a second of eleven cycles after the module had accrued a total of 3383 hr on test. Table V-1 is a compilation of experimental parameters measured at BOL and before and after each set of thermal cycles.

These results show the module performance to be unaffected by the 21 thermal cycles. Within the accuracy to which clad temperatures can be controlled, there was no measurable change in module parameters during either set of thermal cycles.

Figure V-1 gives power-output data from all TEM-9U series preprototype modules plotted as a function of time at life-test conditions. The time at which each set of TEM-9U SN-9 thermal cycles was conducted is marked on the figure.

TABLE V-1
TEM-9U SN-9 THERMAL-CYCLE TEST RESULTS

Time on Test (hr)	Circuit Heat Input Rate (watts)	V_{∞} , ML (volts)	R_g (Ω)	P_e (watts)	η (%)
Beginning-of-life (zero hr)	1277	4.93	0.0948	64.0	5.01
2368 (before thermal cycles)	1258	4.82	0.0998	58.7	4.67
2417 (after 10 cycles)	1269	4.81	0.0988	58.6	4.62
3378 (before second cycle set)	1247	4.74	0.1049	53.6	4.30
3425 (after 21 cycles)	1252	4.77	0.1055	53.9	4.30

Figure V-2. Electrical Power Output Obtained from Two
TEM-9Q Series Modules (Containing TEGS-3P Material)
Operated Between $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$
(nominal)

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There is no apparent discontinuity in the level of the power-output data or in the rate of degradation as a result of either set of thermal cycles. The data exhibit the ability of the tubular module to withstand repeated thermal cycles with no observable effect on module performance.

A similar set of thermal-cycle tests was started on TEM-9U SN-10 in the AI two-loop reactor system. These tests, however, have been conducted with a one-day soak period between cycles. Ten cycles have been completed on TEM-9U SN-10 and the time at which the cycling occurred is indicated on Figure V-1. Once again there has been no measurable degradation resulting from the thermal-cycling of this module.

In addition to the thermal cycling of preprototype modules, similar tests have been started on TEM-13A SN-2, a prototype-size module. These tests, the results of which are discussed in a later part of this section, have indicated that the large modules are also extremely insensitive to rapid thermal cycling.

d. TEM-9Q SN-5 and SN-6 Endurance Tests

The 9Q tubular module series has been identified to investigate the high-temperature capability of TEGS-3P material. Although modules fabricated with TEM-9Q-type TE circuits exhibit lower BOL circuit efficiencies than modules using TEGS-2P, the alternate P-type material is being examined because of its relatively high performance stability at elevated temperatures.

Two 9Q modules were placed on test during the last quarter at simulated two-loop reactor temperatures. The 9Q SN-5 module has accrued 1719 hr on test and the SN-6 has been tested for 1365 hr. The power output of these modules is shown in Figure V-2 as function of time on test. These data may be compared with the preprototype-module (TEM-9U SN-7, -8, -9, and -10) life-data presented earlier in Figure V-1. It is evident that the TEGS-3P modules show no measurable degradation to date, while the preprototype modules after a similar time on test already revealed a finite reduction in power.

The rate of change of internal resistance is generally a more subtle indicator of degradation trends than the power-output alone. Both 9Q modules in question have exhibited internal resistance increases of approximately 2 m Ω . All previous TEM-9Q series modules tested at hot-clad temperatures ranging from 1000 to 1200°F have exhibited decreases in internal resistance during initial

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testing. Instrumentation of the 9Q SN-5 and SN-6 modules is unique in that the internal resistance is measured between access pins. All previous modules were so instrumented that internal resistance was measured only across the TE circuit. Internal-resistance data from the SN-5 and SN-6 modules suggest that a deterioration of the access-pin circuit may exist at elevated operating temperatures.

2. Performance of Prototype Modules

a. TEM-13 Series Module Static Tests

Five reactor power prototype tubular modules were placed on test during the previous quarter. Two (TEM-13B SN-1 and TEM-13C SN-1) were destroyed due to testing equipment malfunctions which occurred during their heatup cycles. One module was removed from testing for heater modifications, and two continued on test during the quarter at constant temperature and matched-load conditions.

The two continuing on test, TEM-13A SN-1 and SN-2, are being tested at average-temperature conditions simulating operation of the reactor system. Power-output data recorded for these are reported in Figure V-3 along with

Figure V-3. Comparison of Electrical Power Output Data from Three Prototype TEM-13-Series Modules at Life-Test Conditions

data from the TEM-13B SN-2 module. The latter was tested for 422 hr at the same temperature conditions before a failure of the electrical heater. This module is scheduled to be placed back on test in February, at which time it will be subjected to a series of thermal cycles.

It is clear from Figure V-3 that a significant degradation mechanism is present in the TEM-13-series modules. The cause of this has been identified as poisoning of the N-type TE material by the P-type as a result of diffusion around and/or through the mica washers. The poisoning effect will be proportionally more severe in the higher-voltage modules which use a large number of very thin washers of TE material. The larger degradation rate of TEM-13B SN-1 shows this indeed to be the case.

A redesign of the inner-conductor-ring has been undertaken to eliminate the possibility of contact between P- and N-type PbTe; the new design is discussed in another section of this report.

b. TEM-13A SN-2 Thermal-Cycle Test

A series of ten thermal cycles was completed on the subject prototype module during December. The test profiles are essentially identical to those used for the TEM-9U SN-9 discussed previously. The cycling of the module was started after 2303 hr at life-test conditions. The results obtained from these tests show no discernible change in module performance due to thermal cycling. As shown in Figure V-3 there is no apparent discontinuity in the level of the power output or in the rate of degradation as a result of these cycles.

3. Engineering

a. Inner-Conductor-Ring Improvements

Destructive examination of TEM-9U SN-7, as discussed later in this section, indicates that based on present evidence there has been contamination of the N-type washers due to material transport from the P-type PbTe around and/or through the mica washers which separate the N- and P-type PbTe. Furthermore, the amount of contamination appears to be greatest in that region of the N-type material immediately adjacent to the large mica washer near the inner-conductor-ring where the mica washer forms a butt joint with it. This indicates that part (or all) of this problem may be due to an imperfect seal at this butt joint which is formed during module compaction.

Figure V-4. Stepped Inner-Conductor-Ring Schematic

Figure V-5. Sectioned View of TEM-9U SN-21
(See Figure V-4 for identification of individual components.)

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Figure V-4 schematically illustrates a new inner-conductor-ring design which provides a positive mica barrier between the P- and N-type PbTe washers. The inner-conductor assembly is fabricated in two pieces. The large mica is sandwiched between the two inner-conductor-ring components at assembly and therefore eliminates the possibility of an imperfect seal between the mica and the inner-conductor ring. The two-piece inner-conductor-ring design concept also provides flexibility in material selection which can be used to minimize chemical reactions and therefore module degradation at the PbTe washer/conductor-ring interface. Therefore, a different conductor-ring material can be used adjacent to the N-type material than that adjacent to the P-type.

The new inner-conductor-ring design was used in the TEM-9U SN-19, -21, and -22 modules.* After processing TEM-9U SN-21, the module was destructively examined. Figure V-5 shows an 8X magnification of a section of TEM-9U SN-21. The destructive examination indicated that a positive seal between the N- and P-type lead-telluride materials was established at the inner-conductor-ring interface and that good axial alignment and radial contact were established between the two inner-conductor-ring components.

b. End-Closure Improvements

In an effort to decrease the axial weld shrinkage and reduce the mica distortion observed in the first few couples of each end in prototype-size modules which have been destructively examined, it was decided to reduce the size of the weld joining the outer clad to the retainer ring.

Examination of radiographs of the TEM-13A SN-1 revealed that the mica shift discernible after the completion of all outer-clad weldments did not increase after further processing. It was felt therefore that a reduction in the volume of weld could significantly reduce weld shrinkage and consequently decrease the observed mica shift.

Various approaches to the problem were considered. Examination of module weld sections indicated that the weld could be reduced on the modules and still maintain its structural integrity during processing and testing. An etched section of the weld indicated that the weld penetration resulted in a load-carrying

*These modules were prepared in support of the Compact Converter Program.

shear-area less than the full thickness of the outer clad. Consequently future modules will be fabricated with a reduced radial thickness of weld.

Since the clads of each of the modules yield at test conditions due to differential expansion of the circuit components and clad, each of the welds for the various size modules was scaled from the radial thickness of the clads.

The only modules to date which have been built with the new weld configuration include the TEM-13F SN-1 and TEM-13D SN-1 and -2 and other modules (TEM-9U SN-19, -21, and -22) built under a separately funded program.

Axial measurements were taken across the retainer ring and end outer-clad faces at various stages of assembly and processing. A summary of the measurements is given in Table V-2. Earlier modules are included to show the differences and effect of the decrease in weld size. Although there is a definite decrease in shrinkage a more positive indication was noted in a section of the TEM-9U SN-21 taken after processing. The mica shift was apparent only in the first couple and was obviously improved over the distortion condition previously observed.

TABLE V-2
MODULE AXIAL-MEASUREMENT SUMMARY

Module (9U)	SN-7	SN-8	SN-9	SN-19	SN-21	SN-22
Average distance across retainer ring (after assembly), in.	9.240	9.245	9.245	9.247	9.250	9.247
Average distance across retainer ring (after processing), in.	9.155	9.172	9.173	9.189	9.189	9.188
Average shrinkage, in.	0.085	0.072	0.072	0.058	0.061	0.059

Dimensional data for the TEM-13 modules did not indicate a significant decrease in weld shrinkage. However this has been attributed to the control of the weld size which effectively permits a greater percentage change in the weld for the TEM-13. Other methods of possibly reducing the effect of welding have included investigation of the EB method, and revising the welding sequence which could conceivably decrease the axial shrinkage by 50%. Preliminary examination of dimensional data taken of weld samples which were EB welded showed

virtually no radial or axial shrinkage after a weld penetration of approximately 0.10 in.

c. Status of Additional Modules

The TEM-13D-series modules are prototypes similar to the TEM-13C and the first of the reactor modules to utilize the stepped inner-conductor-ring concept. They incorporate two 0.0015-in. mica insulators within each couple assembly in lieu of a single-piece 0.003-in. barrier between the PbTe washers. The P-leg of these modules consists of TEGS-3P material. An additional modification of the outer-clad-to-retainer weldment was included. The radial thickness of the weld was reduced from 0.100 to 0.060 in. in an effort to minimize axial weld shrinkage. Two TEM-13D modules have been built and processed and are currently being instrumented for test.

The TEM-13F which has the same conductor-ring arrangement as the TEM-13D has been redesigned to incorporate the smaller weld as defined for the -13D. In addition a solid collector ring will be used instead of the standard "scalloped" ring. This design feature should tend to further reduce any mica movement as observed in the TEM-13C. The TEM-13F uses a W/Fe inner-conductor-ring and a Mo outer-conductor-ring in place of the Fe in the TEM-13D since TEGS-2P material is used instead of TEGS-3P. The Fe-stepped conductor ring in combination with a fitted W ring insures capture of the large-diameter mica between them, thus providing a more positive barrier between the P- and N-type PbTe.

The TEM-13E and -13G modules are now in the design stage. The TEM-13G is similar to the -13D except for the use of 2P rather than 3P washers and relatively minor dimensional changes in the circuit components designed to accomplish a better fit during assembly. The TEM-13E modules will be similar to the -13D except that the -13E module circuits will contain a total of 226 couples, bringing the power output of these converters up to the level required of prototype modules. The additional couples will result in a 2.0-in. increase in the active circuit length of these modules.

4. Destructive Examination of TEM-9U SN-7

A preprototype reactor power module containing W inner-conductor-rings, TEM-9U SN-7, was removed from test after 3500 hr at $\bar{T}_H = 607^\circ\text{C}$ (1125°F),

$\bar{T}_C = 302^\circ\text{C}$ (575°F). The module resistance increased during the test but the open-circuit voltage did not exhibit any increase. In contrast, modules containing Fe inner-conductor-rings have shown an increase in internal resistance and open-circuit voltage. Two sections were removed from the center of the module where the hot-side temperature was approximately 645°C (1193°F). Section 1 was polished to expose a P-type (TEGS-2P) washer. The washer and the W/TEGS-2P interface were examined metallographically with the Seebeck probe and the electron microprobe. Metallographic examination revealed no reaction either in the P-type PbTe or at the interface region. Seebeck probe traces were taken on a radial line from the inner-conductor-ring to the outer-conductor-ring. The data indicate that the Seebeck coefficient at the hot junction exhibited a very slight rise which may be seen in Figure V-6. The curve shown is an average of six probe traces taken every 45 degrees. The increase in the value of the Seebeck coefficient is much less than would be necessary to produce the measured increase in the internal resistance of the module.

The W/PbTe interface of the P-type washer was also examined using the electron microprobe. Figure V-7 is a step-scan taken over the interface measuring the W, Te, and Pb X-ray intensity. The steps were taken to 2μ intervals.

Figure V-6. Seebeck Coefficient vs Distance from Inner-Conductor-Ring for Section 1, TEM-9U SN-7 P-Type Washer: Average of Six Probe Traces $\bar{T}_H \approx 645^\circ\text{C}$

Figure V-7. Electron-Microprobe Trace for Tellurium, Tungsten, and Lead from P-Washer of TEM-9U SN-7, $T_H \approx 645^\circ\text{C}$. Trace Begun in Lead-Telluride Matrix and Moved to P-Type/Tungsten Interface

Figure V-8. Seebeck Coefficient vs Distance from Inner-Conductor-Ring for N-Type Washer of TEM-9U SN-7, $T_H \approx 645^\circ\text{C}$. Trace Taken in a Plane 79 mils in an Axial Direction from Large Mica Washer

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The traces revealed no apparent diffusion of W, Te, or Pb across the contact interface. Section 1 was then polished to expose the N-type washer immediately adjacent to the previously examined P-type washer. Metallographic examination of the W/N-type interface revealed the same result as found at the W/2P interface in the preceding washer, i. e. no interdiffusion of the major constituents.

Seebeck probe traces, however, revealed a considerable Seebeck coefficient change in the N-type PbTe near the hot inner-conductor-ring. Figure V-8 is a plot of the Seebeck coefficient vs distance from the inner-conductor-ring. Immediately following the Seebeck probe examination and without repolishing, the sample was examined in the electron microprobe. A step-scan was carried out across the W/N-type interface in 2μ steps. Figure V-9 is a plot of the step-scans for W, Te, and Pb. This figure shows the same results as those obtained from the examination of the W/2P interface, i. e. no apparent interdiffusion of materials across the interface. The sample was then ground away in 0.015-in. steps and radial Seebeck probe traces taken at 90° intervals around the washer. Figure V-10 is a schematic showing the axial position of the traces taken in the N-type washer. As the washer was ground away and the mica washer between the N- and P-type PbTe washers was approached, the Seebeck probe traces exhibited a marked change. The N-type PbTe began to show P-type electrical characteristics near the hot-side conductor ring. Traces were taken of intervals down to a plane whose axial distance was approximately 0.014 in. from the large mica washer. The large mica washer is defined as the one which, acting as a barrier between the N- and P-type PbTe, touches the OD of the inner-conductor-ring and separates the outer-conductor-rings. For simplicity, two of the four traces taken on this plane which was 0.014 in. from the large washer are shown in Figure V-11.

Section 1 was then polished to expose the next P-type washer immediately adjacent to the N-type washer discussed above. This washer was examined in a similar manner to that used in the previous N-type washer, i. e. Seebeck probe traces taken every 90 degrees and then ground away in 0.015-in. steps and the Seebeck examination repeated. The washer was examined at five axial positions ranging from 0.013 to 0.074 in. from the large mica washer. There was no apparent Seebeck rise near the hot inner-conductor-ring in any of the five axial positions.

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Figure V-9. Electron-Microprobe Trace for Tellurium, Tungsten, and Lead from N-Type Washer of TEM-9U SN-7, $T_H \approx 645^\circ\text{C}$.
Trace Begun in Lead-Telluride Matrix and Moved to N-Type/Tungsten Interface

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Figure V-10. Schematic of Axial Position of Seebeck Probe Traces
Taken on N-Type Washer TEM-9U SN-7, $T_H \approx 645^\circ\text{C}$

Figure V-11. Seebeck Coefficient vs Distance from Inner-Conductor-Ring for N-Type Washer of TEM-9U SN-7, $T_H \approx 645^\circ\text{C}$. Trace Taken in a Plane 14 mils in an Axial Direction from Large Mica Washer

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Section 1 was then polished to expose a second N-type washer which was exactly one-couple-length distance from the first N-type washer examined. The initial Seebeck probe traces taken on this washer were approximately 0.005 in. from the small mica washer. The probe traces revealed that some of the N-type PbTe near the hot inner-conductor-ring had been converted to P-type material. This conversion phenomenon was not present in the trace taken at the next axial position examined, 0.015 in. from the small mica washer. The washer was ground away in 0.015-in. steps and Seebeck probe traces taken at each interval. The washer exhibited a rise in the Seebeck coefficient at the hot junction at each step but no conversion to P-type was encountered until a plane 0.009 in. away from the large mica washer was reached. In this plane the N-type PbTe was converted to P-type in a manner similar to that observed in the first N-type washer examined.

From the data obtained by the examination of the washers it is evident that (1) only the N-type washers seem to be affected, and (2) the N-type washers are being contaminated at both extremes of their axial dimension. Further, it appears that the extent of the gross contamination (conversion to P-type) is greater at the axial end of the washer where the large mica washer separates the N- and P-type washers. In the first N-type washer examined, the gross contamination extends at least to an axial position 0.031 in. from the large mica washer. Conversion is not evident in traces taken 0.079 in. from the large mica washer (0.011 in. from the small mica washer). In the second N-type washer from Section 1 the gross contamination is not present at an axial distance of 0.070 in. from the large mica but is found again at a distance of 0.085 in. (0.005 in. from the small mica).

The present evidence indicates that the contamination of the N-type washers in TEM-9U SN-7 was due to transport of material around and/or through the mica washers which separate the P- and N-type PbTe. The extent of contamination is shown schematically in Figure V-12. Two N-type washers are shown with the converted contaminated areas as well as those areas which have a Seebeck coefficient greater than normal N-type material. It is evident from this figure that the contamination, as evidenced by P-type conversion, is greater near the mica which makes the butt joint at the hot-side conductor ring. It is possible that some contamination could occur in the region adjacent to the

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Figure V-12. Schematic Representation of Contamination of N-Type Washers of TEM-9U SN-7

small mica by means of P-type material penetrating the mica. However, surface diffusion along the ID and along the sides of the N-type washer could be a factor. Since the axial conversion-contamination-depth at the small mica is only about 15 to 20 mils, a comparable radial distance would be difficult to detect since the contact surfaces are cylindrical rather than plane. TEM-9U SN-22 has been designed to prevent interdiffusion via the butt-joint region by extending the large mica between split conductor rings. The module is on test and the data should give some indication as to the success of the proposed solution.

The diffusion of the P-type material which contaminates the N-type PbTe is temperature-dependent as evidenced by data obtained by examining N-type washers from sections of the module which experienced lower temperatures. Three successive N-type washers from the negative end of the module were examined and found to be free of any contamination, i. e. the Seebeck trace was flat from the inner- to the outer-conductor-ring. The hot-side temperature of these washers was about 400°C (752°F). Seebeck probe traces on N-type washers from the positive end showed that contamination had occurred in the first N-type washer and became greater in successive washers. The third N-type washer was the first to show that P-type conversion had occurred. The temperature at this end of the module ranged from approximately 500°C (932°F) at the first N-type washer to 550°C (1022°F) at the third N-type washer where conversion was observed.

Sections from TEM-9U SN-7 were examined in the electron microprobe to determine if any compositional differences could be detected between the contaminated and unaffected zones in the N-type washers. EB traces and counting techniques were carried out on axial sections. No differences were observed between contaminated and unaffected portions of the N-type washer nor between the N- and P-type washers.

The results of this examination allow a plausible explanation for why the internal resistance of the TEM-9U SN-7 increased during testing while the open-circuit voltage did not show any changes. The resistance increase was caused by high N- and P-type resistivity regions in the N-type washer. Since the resistance of either N- or P-type material is an additive phenomenon of absolute values, the total resistance increased. However, the Seebeck voltage is an algebraically additive function so that the converted P-type regions act as bucking voltages in the N-type washers, offsetting, in this case, the increased voltage regions which have not been converted. As more and more conversion occurs with time, decreases in the open-circuit voltage of the module can be expected. Further, the examination of the module did not produce any evidence that the W was not compatible with the P-type PbTe.

Figure V-13. Electrical Power Output Produced by the Pump Power Module TEM-14A SN-1 at Life-Test Conditions

Figure V-14. Load-Voltage Data Obtained from the Pump Power Module at Life-Test Conditions

B. REACTOR PUMP MODULE

1. TEM-14A SN-1 Static-Test Results

The first tubular module designed to power an EM pump, the TEM-14A SN-1, has accrued 1550 hr on static test at average temperature conditions simulating those of the ZrH two-loop reactor system ($\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$). Power output and load-voltage data recorded since the module was placed on test are shown in Figures V-13 and -14.

The pattern of stable operation which was noted during the first 500 hours of testing and which was reported in the previous progress report⁽²⁾ continues to be exhibited. The slight variations which exist from one data point to the next may be ascribed entirely to normal fluctuations in operating temperatures.

2. TEM-14A SN-2 Design Modifications

The initial success of the TEM-14A SN-1 module has essentially confirmed the basic design of the module and has enhanced confidence in the areas of design analysis and fabrication.

The first prototype module was fabricated using Ni power leads connecting the module circuit to the external collector ring. Ni was chosen because of its resistance to oxidation at high temperatures. However, the electrical resistance at the pins at operating conditions is $0.160\text{ m}\Omega$ (total for all pins). A study was made to find an alternate pin material which would be stable at up to 1200°F with improved electrical resistivity. To date no suitable material has been defined which will be compatible with the design components and test conditions.

The only major design improvement contemplated at this time for the second pump module (TEM-14A SN-2) is a reduction of the converter resistance by (a) substituting a Mo collector ring for the Fe ring and (b) increasing the number of Ni power pins from six to 12 per end.

C. CONVERTER MODULE

1. Mechanical Design of the Converter Assembly

The preliminary design of the converter assembly has been established to accommodate the functional characteristics of system operation. The assembly consists of six converter modules connected by tubular headers to form the

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module assembly, a structural frame which supports the assembly, and an outer casing which contains and houses the assembly.

The overall size of the assembly had previously been established at 12.5 by 19.5 by 24.0 in.; however, the preliminary design shows an increase in the 24-in. dimension to 25. This was necessary to accommodate the individual converter modules which were redesigned with a 90° bend connecting the tubular modules to the primary headers.

Preliminary analysis of the converter assembly piping and structure has been completed. The individual converter modules and assembly headers and piping have been designed to withstand 30 and 140 psi at 100°F internal pressure and 15-psi external pressure at 20 g's for both the HRL and primary loops respectively. The wall thickness required to meet these conditions as specified by the U.S.A. Standard for Pressure Piping (USAS 313.1.0) is shown in Figure V-15 as a function of pipe diameter. The figure shows that from strength considerations, with a pipe size of 1.5-in. diam (primary and HRL pipe size), a wall thickness of 0.007 would be adequate. From a fabrication standpoint, however, the practicality of such a thin wall is questionable, and to accommodate manufacturing and welding a minimum 0.035-in. wall has been selected.

The structural frame is comprised of four 1.25- by 1.25- by 0.125-in. angle columns. Between each side column is a 1.25- by 1.25- by 0.1875-in. angle which supports the individual converter modules. The support angle has been designed as a uniformly loaded beam and can adequately maintain the maximum loads experienced during launch and operation of the assembly.

The primary and secondary piping penetrations through the outer case have been positioned to correspond with a preliminary layout of gallery components received from AI. The inlet and outlet connections for the HRL will be anchored directly to the structural frame of the assembly. However, the primary connection must also be designed to minimize the thermal losses which would be present if direct attachment to the frame were considered. Evaluation of loads and stresses due to thermal expansion have not yet been completed.

The conceptual design originally proposed contained some 202 critical welds. This included some performed inside the hot manifold but which have now been eliminated, reducing significantly the number of welds and the amount of welding.

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Figure V-15. Converter Assembly Piping Thickness vs Pipe Diameter

The number of final assembly welds in the preliminary design has now been reduced to 24. These are the four inlet and outlet connections from each converter module. All of these welds are butt-welds which can be doubly contained if necessary. However, the high confidence with which these can be made make the necessity of double-containment questionable. An overall pressure test and helium-leak test of the assembly is also proposed to increase the reliability of the system.

Figure V-16. Conceptual Design of Thermoelectric Converter Module

2. Mechanical Design of the Converter Module

Preliminary studies of the conceptual design of the TE converter module have been completed, and Figure V-16 indicates the preliminary design proposed.

The TE converter module consists of two separate systems, the primary and HRL's. The latter consists of two tubular plenums 2.25 in. ID connected by four 1.9-in.-ID shrouds. Each plenum has a 1.1-in.-ID connection welded to the converter assembly header. Attached to the ID of each shroud is a helical spiral which is required to ensure good flow distribution and minimize the film-drop. The HRL subassembly is completely assembled and pressure-tested before assembly with the primary loop.

The primary loop consists basically of the tubular module inner clad, which forms the main flowpath. This is also orificed by a helical spiral on a hollow tube. A 90° elbow is welded at each end of the inner clad, which connects to the primary header. The bends are included to accommodate the differential

thermal expansion which will be experienced during operation and minimize the stresses at the connections.

The converter module is supported by a plate which attaches to each shroud and is bolted to the structural frame of the converter assembly.

3. Hydraulic and Thermal Analysis

The primary goal in this area at this time is to insure that the design of the converter assembly is consistent with the NaK pressure-drop limitation and a relatively high film coefficient at the primary and secondary heat-transfer surfaces. The design goal for the overall pressure-drop in both loops of the assembly will be 0.4 psi, with roughly half the pressure drop across the tubular modules. This sort of pressure-drop allocation will insure a uniform flow distribution between all tubular modules.

A conservative analysis of the pressure-drops associated with the design presented herein has been performed and the results indicate that the design goals can be met, particularly in the flowpaths external to the tubular modules. The helical flowpaths within the modules can be adjusted to vary the pressure-drop and film-temperature drop over a wide range. Predictions have been made for the pitches of the helical channels which will yield pressure drops of the order of 0.2 psia. Under these conditions the film-temperature difference between the bulk coolant and the clads was found to be $<20^{\circ}\text{F}$ on the hot side and $<7^{\circ}\text{F}$ on the cold side. A series of hydraulic tests to verify these predictions, discussed later, is planned.

As a prelude to comprehensive performance analyses of complete converter modules, work was started on a detailed hydraulic-TE performance model of a single tubular module. The object of this numerical model is to directly relate the important TE performance parameters to the inlet working-fluid flowrate and temperature in both loops. Two versions of a code called TEFLOW have been completed to perform these studies. The first version treats the module as a passive counterflow heat exchanger; it was primarily designed to check various numerical techniques and to serve as a test vehicle for evaluating various methods of computing heat-transfer coefficients and fluid-to-module film-temperature differences in the helical-flow channels.

A review of candidate heat-transfer correlations is underway. Previous analyses have used variations of the Lyon-Nusselt number correlation with no consideration of the fact that the heat transfer is not bilateral in either the inner or the outer annuli. Whether or not a more refined correlation is justified is still an open question.

The second version of the TEFLOW code augments the basic heat-transfer model with the inclusion of TE effects. With this tool, realistic axial and radial temperature distributions within a module can be obtained along with TE performance estimates. This code was completed during the last week of January and extensive parametric studies are now underway. The first application of the code will be a study of the effects of flow maldistribution within assemblies of tubular modules. This problem reduces to an examination of the effects of off-design flow conditions on the performance of individual modules, which can then be suitably combined to obtain overall assembly performance changes.

A thermal analysis to determine the optimum position of the shroud-weld on the tubular-module outer-clad has been completed. The study indicated that heat losses through the end-closure regions could be held to the design level of 150 watts per end if the shroud-weld is located on the axial planes of the first and last mica washers of the TE circuit.

4. Converter Assembly Stress Analysis

A preliminary stress analysis of the module assembly structural support and connecting headers was made to size the support angle and column and determine wall thickness of headers; however, further analysis is required to determine the thickness and rigidity of the outer casing through which the applied loads (100 lb in any direction, 70 ft-lb torque and 70 ft-lb bending) will be distributed.

a. Converter Module Stress Analysis

The Inconel-718 inner clad of the tubular module is utilized as the basic component in the primary loop of the assembly. The inner clad is connected to the primary plenum by thin-walled 316-stainless-steel tubing having a 90° bend. The stainless-steel tubing, besides functioning as the inner-clad/plenum connecting medium, must also serve as a compensator for the difference in differential-thermal-expansion between the primary and secondary system

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connections to the tubular module. This difference arises from the 600°F thermal gradient across the active portion of the tubular module.

The detail stress analysis of the various joints within the converter module is not yet complete. It is intended that in parallel with the analysis there will be structural mockups fabricated for low-cycle fatigue-testing to verify the results of the analyses.

5. Hydraulic Testing

The design of a hydraulic-test facility for conducting converter-module flow tests has been completed. Mockups of individual tubular modules and converter modules (4-packs) will be placed in the facility. Water-flow conditions will be used to simulate a variety of NaK-flow conditions within $\pm 50\%$ of those associated with the normal operating point of the reference converter system. A specific goal of these tests will be to determine the pressure-drops associated with various helical-channel-pitch values. The resulting pressure-drop-pitch relations will be used to set the pitches of the reference module flow channels and to verify and improve analytical predictions.

The hydraulic testing of the primary and secondary flow channels will begin with testing the helical-flow-channels associated with the individual tubular modules. These tests will permit the selection of a specific helix configuration for the primary and secondary channels. These configurations will then be incorporated into plastic mockups of the converter module so that overall pressure-drop, flow distribution, and header sizes can then be established with confidence.

Equipment required for the hydraulic-test facility has been ordered, and construction is expected to be completed by the end of January. The design of flowpath mockups has been completed and fabrication begun. Testing should begin by the third week in February.

Since a conceptual mechanical design for the assembly has been completed, work will be started during the next quarter to obtain estimates of the temperature distribution within the assembly and over the surface of the outer container. These results will be used to set up a heat inventory for the assembly.

The fluid-flow tests of the primary and secondary sides of the tubular module and converter module will be made with water at the same Reynolds number

TABLE V-3
LIQUID-METAL (NaK-78) FLOW CONDITIONS AND EQUIVALENT
WATER-FLOW CONDITIONS AT SINGLE-MODULE
DESIGN POINT

Condition	Flowpath			
	Primary		Secondary	
	NaK	Water	NaK	Water
Flow, lb/min	7.9	53.6	7.76	30.2
Velocity, ft/sec	1.7	8.5	0.541	1.69
Area (annulus), in. ²	0.244	0.244	0.692	0.692
Reynolds No.	16,800	16,800	3330	3330
Pressure-drop, psi	0.20	6.74	0.21	2.55
Temperature, °F	1125	70	570	70

TABLE V-4
WATER-FLOW REQUIREMENTS FOR SINGLE-MODULE
HYDRAULIC TESTS

Requirement	Flowpath	
	Primary (hot side)	Secondary (cold side)
Flowrate, lb/min	11.0 to 80.0	5.8 to 44.0
Flowrate, gal/min	1.32 to 9.6	0.7 to 5.3
Pressure-drop, psi	0.60 to 30.0	0.1 to 15.0

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and flow configurations as the operating NaK-filled units. The data generated with the water-flow test will be used to determine pressure-drop coefficients over a range above and below the design-point flow conditions. Table V-3 shows the approximate flow parameters of the NaK at the design points for the primary and secondary flow passages tabulated with the equivalent water-flow parameters. Table V-4 shows the required operating range for conducting the single-module flow tests for the primary and secondary flowpaths, and Table V-2 shows the flow requirements for the converter-module tests.

The single-module flow tests will be conducted with several different swirl generators in order to determine the particular spiral pitch that will meet the pressure-loss requirements.

When the data have been obtained for the single module, this configuration will be duplicated as a four-channel converter-module unit. The unit will be tested to determine if the header design is satisfactory with regard to pressure-drop and flow-distribution.

Further flow-testing will then be performed on the metal mockup later this year.

TABLE VI-1
STATUS OF S8DR ACTUATOR DESIGN-
VERIFICATION TESTING

Phase	Actuator No.		
	SN-011	SN-013	SN-010
Initial 55 thermal cycle	X	X	X
1000-hr endurance at 1125°F	X	X	X
1000-hr dwell at 1125°F	X	X	X
3000-hr endurance at 1125°F	X	X	X
Second 1000-hr dwell at 1125°F	X	X	X
Test hours of the 4000-hr endurance phase at 1125°F	2787	2737	2152

TABLE VI-2
ACTUATOR PERFORMANCE AT 1125°F

Phase	Actuator No.	Brake Current (amp)		Stepping Current* (amp)		Insulation Resistance (ohms)	
		Pull-in (minimum)	Release (maximum)	CW (minimum)	CCW (minimum)	Brake	Minimum Phase
Start of test	SN-011	1.55	0.35	2.9	2.9	7 M	300 k
After: first 1000-hr dwell		1.80					
second 1000-hr dwell		1.62					
Latest Data		1.72	0.34	2.7	3.0	1.2 M	1.2 M
Start of test	SN-013	1.44	0.68	3.9	2.8	7.5 M	5 M
After: first 1000-hr dwell		1.56					
second 1000-hr dwell		0.94†					
Latest Data		1.7	0.6	2.7	2.8	0.65 M	3 M
Start of test	SN-010	1.28	0.34	3.0	3.2	10 M	200 k
After: first 1000-hr dwell		1.6					
second 1000-hr dwell		1.6					
Latest Data		1.5	0.38	2.6	2.7	0.9 M	1.6 M

*To step 40 oz-in. load

†Brake not fully seated during dwell

VI. COMPONENT DEVELOPMENT

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 components include the bearings, actuator, 4π shieldable reflector, shielding, and NaK pump.

A. 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 schedule consists of an initial 55-thermal-cycle phase (400 to 1125°F) including 27 simulated startups, followed by periods of actuator operation and static dwells at 1125°F. During the actuator operational sequences the actuator is stepped at 1125 and 400°F, to which it is periodically cycled. The present status of testing is indicated in Table VI-1. The insulation resistances and current requirements listed in Table VI-2 are within design requirements. All units are operating satisfactorily. No unusual operational conditions were observed.

2. Actuator Technology Development

The effort in this program is directed towards material technology for the development of advanced actuators.

a. Coil Reliability

Tests are being made to evaluate potential high-temperature high-reliability conductor materials for actuator coils. These tests are being performed on coils wet-wound with AI-1 cement on rectangular-shaped alumina bobbins of the S8DR geometry. Three different conductor materials have been tested or are under test as candidate material for the coils; Ni-clad Ag, Cube alloy, and Oxalloy. The Ni-clad Ag conductor consists of a fine Ag core swaged in a Ni cladding. The composite wire is 80% Ag by cross-sectional area. The Cube-alloy conductor is a Cu base which has been dispersion-hardened by addition of fine particles of beryllia. The Oxalloy conductor consists of an OFHC Cu core swaged in a chrome-iron-alloy (430 stainless-steel) cladding. The composite wire is 72% Cu by cross-sectional area.

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 failure of one Ni-clad Ag conductor coil after 1084 hr of continuous testing and subsequent investigation was described previously.⁽¹⁾ The steady-state 1400°F temperature test has been resumed on the remaining five coils. These have shown no significant circuit-resistance variation after a total of 405 thermal cycles, to 1300°F, seven thermal cycles to 1400°F, 338 hr at 1300°F, and 5205 hr at 1400°F.

A second test, converted from the 1200°F thermal-cycle test after 407 cycles and 339 hr at temperature, is continuing at a constant temperature of 1200°F with 8545 hr and 411 thermal cycles accumulated. The four Cube-alloy coils and the five Ni-clad Ag coils have shown no significant change in resistance. The failure and investigation of one Cube-alloy coil was described previously.⁽¹⁾

A third test is being carried out with coils wound with Oxalloy conductors on rectangular-shaped alumina bobbins mounted in an S8DR actuator housing. The purpose of this test is to obtain long-term (12,000-hr) high-fatigue-loading (500 thermal cycles) testing of oxalloy wire at 1125°F. These coils were previously subjected to 200 thermal cycles and 350 hr at 1125°F. In the current test the coils have accumulated an additional 306 thermal cycles and 10,480 hr of testing. Operation to date has been satisfactory with no indication of conductor failure or insulation degradation.

b. Magnetic Materials

The previously described⁽¹⁾ testing of ring samples of Hipercob 27 (a 27%-cobalt material) and Armco iron (a pure iron) continues with 10,192 hr at 1200°F. Measurements of magnetic properties after 9000 hr indicated no significant changes from the earlier data.

B. 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 (1250°F) of 10^{-8} torr. A backup design bearing, TiC coated on Inconel-750 against P5N carbon-graphite, is also at 1150°F and 10^{-8} torr. The verification test sequence consists of: 30 initial thermal cycles from below 400 to the 1150°F (or 1250) test temperature; operation endurance at the constant test temperature (1150 or 1250°F) for 6000 hr with bearing operation approximately every 100; an additional 2000 hr of operational endurance with the bearing being operated before and after each thermal cycle conducted on approximately 100-hr intervals; 1000-hr static dwell at the test temperature; followed by a 3000-hr operational endurance with thermal cycles at 100-hr intervals to complete the 12,000-hr test. The bearing test status and performance are shown on Table VI-3.

TABLE VI-3
S8DR CONTROL-DRUM BEARING-TEST SUMMARY

	Test Assembly				
	1	2	3	4	5
Bearing material	Ref	Ref	Ref	Ref	Backup
(Ref = reference design)					
Nominal test temperature, °F	1150	1150	1250	1150	1150
Test pressure, torr	10^{-5}	10^{-5}	10^{-5}	10^{-8}	10^{-8}
Maximum friction, in.-lb:					
Initial at room temperature	1.8	1.6	1.1	2.7	3.8
Initial 30 thermal cycles					
450°F	3.2	3.7	2.8	3.1	4.0
Test temperature	1.3	1.7	2.2	2.8	2.4
Operational endurance at test temperature:	3.3	3.7	3.5	4.6	3.7
Thermal cycle during endurance					
Room	7.6	5.6	6.5		
450°F	5.3	4.2	4.6		
Test temperature	3.0	2.7	2.9		
Pre-1000-hr dwell	1.9	2.4	2.9		
Total thermal cycles	48	49	49	30	30
Endurance hours					
1150°F	7246	7238		6045	760
1250°F	623	623	7852	672	736
Dwell hours					
1150°F	624	624			
1250°F			624		
Equivalent drum cycles (270° bearing rotation)	160	152	160	86	33

Figure VI-1. Ambient Friction Torque as a Function of Load

Figure VI-2. 800°F Friction Torque as a Function of Load

[REDACTED]

The Al_2O_3 -against-P5 sliding-friction test was terminated after 4000 hr at 1250°F and 10^{-7} torr. The stabilized frictional characteristics previously established⁽¹⁾ remained unchanged. The friction coefficient was 0.29 during the last 550 hr of test. Post-test examination of the couple showed full-width contact between the Al_2O_3 and carbon-graphite. Carbon-graphite transfer was minimal but uniform. There were evidences of substantial sublimation of chrome from the Inconel sample holder to the carbon-graphite and Al_2O_3 surfaces. The eventual migration of chrome to the friction surfaces would have affected the test.

2. Reference ZrH Bearings

The reference ZrH bearing development is designed to advance the bearing technology from the SNAP 8 requirement to 20,000-hr life at temperatures to 1500°F . A prototype bearing of Al_2O_3 -plated Ta-10W socket and shaft, and solid carbon-graphite (Pure Carbon, Grade P5) is undergoing a series of tests. Other tests to evaluate the material performance of these and other combinations at temperatures to 1600°F are also being conducted.

a. UHV Bearing Test

The prototype bearing set with the 2° offset shaft previously tested through 27 thermal cycles to 1500°F ⁽¹⁾ is undergoing a life test at 1500°F and 10^{-8} torr. A parametric load test, duplicating that conducted on an identical bearing set using a straight shaft (see Reference 1) has been completed. Bearing-friction torque was measured at ambient, 800°F , and 1500°F using a matrix of nine load points: 5-, 25-, and 50-lb radial loads, and 25-, 75-, and 150-lb axial loads. Figures VI-1, -2, and -3 show the friction torque for the 2° offset shaft and comparison with the straight-shaft data. Data have been corrected for the zero-load system fixture friction and therefore do not agree with the data previously reported. The curves are anomalous because the friction torque with the offset shaft is less in some cases than the straight-shaft friction. These curves are based on starting torque which tends to be erratic and could be the cause of the anomaly. Similar curves are being prepared based on running friction torque for comparison with the curves in this report.

Following the parametric test, bearing temperature was stabilized at 1500°F and friction torque is being measured twice weekly. After 750 hr at

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Figure VI-3. 1500°F Friction Torque as a Function of Load

TABLE VI-4
LT-2-AGAINST-LT-2 WEAR TEST AT 1600°F

Event at 10^{-8} Torr	Friction Coefficient	
	Test A	Test B
Initial coefficient:		
at ambient	0.56	0.44
at 800°F	0.71	0.61
at 1600°F	0.55	0.72
Wear-test cycles:		
2 to 400	0.48 - 0.61	0.65 - 0.74
400 to 800	0.48 - 0.57	0.59 - 0.78
Final coefficient:		
at 1600°F	0.48	0.65
at 800°F	0.44	0.52
at ambient	0.65	0.69

temperature, friction torque is averaging 5.9 in.-lb after reaching a maximum of 7.1 in.-lb (after 50 hr). The bearing loads are 25-lb radial and 75-lb thrust.

b. Sliding-Friction Material Evaluation Tests

The short-term wear test of LT-2 (60%-W, 25%-Al₂O₃, 15%-Cr cermet) against LT-2 has been completed. Severe chattering occurred during testing and resulted in scoring of the specimens and transfer of material from the base to the riders. Table VI-4 shows a summary of the friction data. The scoring and material transfer between specimens can be seen in the post-test condition shown in Figure VI-4.

Based upon the results of the sliding-friction wear tests, TiC (Kenametal K-162B) vs carbon-graphite (Pure Carbon Co. Grade P5 or POCO AXF-5Q) was selected as the alternate bearing material combination. Long-term friction tests, duplicating the 2000-hr 1600°F Al₂O₃-against-carbon-graphite test previously reported, were started and have completed 1700 hr at vacuum. Figure VI-5 shows friction coefficient as a function of time and temperature for both the K-162B-vs-P5 and K-162B-vs-AXF-5Q couples.

c. Material-Compatibility Testing

The 5000-hr 1600°F material-compatibility test has accumulated approximately 1300 hr at 5×10^{-7} torr and 1600°F. Specimens are being tested for material transfer and compatibility of the various couples. The following couples are under test:

- 1) Al₂O₃ coating (Linde LA-2) on Ta-10W against P5 carbon-graphite
- 2) TiC (K-162B) against P5 carbon-graphite
- 3) TiC (K-162B) against AXF-5Q carbon-graphite
- 4) Solid Al₂O₃ against AXF-5Q carbon-graphite
- 5) Solid Al₂O₃ against LT-2 cermet
- 6) LT-2 against P5 carbon-graphite

Specimens are being removed from test and metallurgically examined after 500, 1000, 2000, and 5000 hr. The 500- and 1000-hr specimens have been removed and are currently being evaluated.

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Test Couple A

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Test Couple B

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Figure VI-4. Reference Reactor Materials Wear Test at 1600°F and 10⁻⁸ torr, LT-2 Against LT-2 (900 in. of travel at 1600°F)

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Figure VI-5. TiC (K-162B) Against Carbon-Graphite
Sliding-Friction Test

TABLE VI-5
LIMIT-SWITCH TEST DATA

SN	Description	Applied Voltage (volts dc)	Test (hr)		Resistance (Ω)	
			Previous	Total	Contact	Insulation*
001	Production unit	1	5,518	14,804	0.794	1.4 M
003	Alumina insulation	24	2,771	12,057	0.771	20 M
005	Alumina insulation	24	2,771	12,057	1.033	25 M
006	Development unit	24	15,518	24,804	1.077	182
032	Production unit	1	5,518	14,804	0.456	1.6 M
065	Production, with vent-holes	1	2,771	12,057	1.408	100 M
066	Production, with vent-holes	1	2,771	12,057	0.660	100 M
068	Production unit	1	2,771	12,057	0.351	113 K
069	Production unit, with vent-holes	1	2,771	12,057	0.474	12.5 K

*Lowest resistance measured between the two pins and case

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Figure VI-6. S8DR Limit-Switch Insulation Resistance

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d. Shock and Vibration Testing

Design of the test fixture to evaluate the effects of shock and vibration loads on the bearing has been completed. The basic fixture can also be used for the shock and vibration testing of the complete control-drum assembly. The fabrication of this fixture has been started.

C. INSTRUMENTATION

1. Position Sensors

The modified S8DR design position sensors completed the 12,000-hr testing as reported previously.⁽²⁾ The disassembly and detail post-test inspection are in progress.

2. Limit Switches

Endurance testing was completed on nine switches selected from previously run short-term static and dynamic tests. During the final phase of testing, 9286 hr, 58 temperature cycles, and 7592 switching cycles were accumulated. The final test data and a description of the test items are listed in Table VI-5. The insulation-resistance variation of each switch throughout its test history is shown on Figure VI-6. The initial beneficial effect of the alumina insulation, SN's 003 and 005, and of the added vent-hole, SN's 065 and 066, which had essentially disappeared, was re-established after the inadvertent air leakage. However, the data do not allow a conclusive judgment to be made concerning the merit of either of the modifications to the original switch design. All of the switches, including SN-006 which was tested to nearly 25,000 hr, completed the testing with insulation resistances above the minimum value ($>100\Omega$) required for S8DR operation. All contact resistances were likewise within S8DR requirements (15Ω maximum). Post-endurance operational sequence test will be conducted on all switches and selected units will be disassembled and examined in detail.

D. SHIELD MATERIALS DEVELOPMENT

1. Liquid-Lead Studies

The objective of the liquid-lead shielding program is to investigate the problems associated with the use of liquid lead as a high-temperature gamma

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shielding for the SNAP Reactor Program. In the initial phase of the program the corrosive effect of exposure to liquid Pb on three structural alloys was investigated; 1020 carbon steel, Croloy-2-1/4, and Type-446 stainless steel at 1000, 1300, and 1500°F for 625, 2500, and 10,000 hr. The current phase of the program is two-fold:

- 1) To determine the change in mechanical properties produced by the combined action of time, temperature, and molten metal on the three alloys.
- 2) To establish the level of carbon needed in Pb to prevent the decarburization of the structural alloys by the liquid metal and the resultant expected degradation of mechanical properties.

Two tensile specimens were machined from each of 27 test capsules representing the three above alloys. One specimen represented the capsule material exposed to liquid Pb, and the other, the material exposed to the Pb vapor. The mechanical testing at room temperature of these specimens for ultimate tensile strength, yield strength, and percent elongation is currently in progress. Preliminary data indicate no significant difference between the mechanical properties of the material exposed to the liquid and that exposed to the vapor.

The Pb-decarburization control effort is based on the premise that control can be established by adding carbon to the Pb. To obtain data on the extent and rate of the solubility of C in Pb, a sealed Croloy-2-1/4 capsule containing liquid Pb and powdered graphite was heated for about 300 hr at 1500°F. The Pb was solidified by quenching the capsule in water from the 1500°F test temperature. Chemical analysis indicated the Pb contained 88 of the 100 ppm of C originally placed in the capsule. The literature value for the solubility of C in Pb at 1500°F is 95 ppm.

The lack of decarburization of the capsule material exposed to the liquid Pb, contrasted to approximately 1 mil of decarburization on the material exposed to the Pb vapor is an indication that addition of C to Pb should control decarburization. Fifty-two capsules were prepared (21 1020-carbon-steel, 21 Croloy-2-1/4-steel, and 10 columbium-1-zirconium) and assigned into the test matrix shown in Table VI-6. Empty control capsules of all three alloys were also prepared to establish the effects of time and temperature on the alloys.

TABLE VI-6
MATRIX OF TEST CAPSULES

PA-32C	PB-32C	PD-32
PA+32C	PB+32C	PD-33
PA-32C1	PB-32C1	PD-34
PA-33C	PB-33C	PD-52
PA+33C	PB+33C	PD-53
PA-33C1	PB-33C1	PD-54
PA-34C	PB-34C	PD-5X
PA+34C	PB+34C	ED 52
PA-34C1	PB-34C1	ED 53
PA-52C	PB-52C	ED 54
PA+52C	PB+52C	
PA-52C1	PB-52C1	
PA-53C	PB-53C	
PA+53C	PB+53C	
PA-53C1	PB-53C1	
PA-54C	PB-54C	
PA+54C	PB+54C	
PA-54C1	PB-54C1	
EA 52	EB 52	
EA 53	EB 53	
EA 54	EB 54	

where

[REDACTED]

The first level of carbon addition was selected as 100 ppm for 1500°F test capsules and 70-ppm C for 1300°F capsules. The amounts correspond to the literature data on C saturation of Pb at the respective temperature. The second level was set at 50-ppm C for the 1500°F capsules and 35-ppm C for the 1300°F, equivalent to one-half saturation level. All carbon additions were made by the direct addition of weighed amounts of powdered reactor-grade graphite to the Pb. No corrosion inhibitors or C were added to the Cb-1-Zr capsules, since these additives are beneficial only to iron-base alloys, based on current knowledge.

After loading, the open end of each capsule was pinched shut and Heliarc-welded in argon leaving a small (~1/8-in.-long) gap in the weld. The final closure was made by electron-beam (EB) welding (at 10^{-5} -torr pressure). Each capsule was sealed in an individual evacuated quartz tube before being placed in the test furnace. The first group of capsules will be removed after 2500 hr of static heating.

2. Aggregate Neutron-Gamma-Shield Materials

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

The current program has three major activities:

- 1) The density-particle size study.
- 2) The H-relocation (in ZrH/LiH) test.
- 3) The physical and mechanical test program.

The results of the completed density-particle size study were detailed previously.⁽²⁾

The 10,000-hr portion of the H-relocation test is still in progress. The planned completion is mid-March 1969, based on 8500 hr completed to date.

The physical and mechanical properties test program was continued with the fabrication of 1-in.-diam rod specimens of LiH/ZrH and LiH/W. Four of each type were made. The compressive creep-test samples will be prepared from these rod specimens.

E. NaK-PUMP DEVELOPMENT

The NaK-pump development is focused on a multiloop dc conduction pump applicable to the reference zirconium hydride reactor - Compact Thermoelectric Power System.

The detail design of the two-pass pump, previously described⁽²⁾ was completed and released for fabrication. The throat section, with a 0.500- by 1.000-in. rectangular center section and a circular section at both ends, was successfully formed. The balance of the components are in various phases of fabrication.

Developmental efforts are in progress to define and qualify the various processes required for the manufacture and assembly of the pump. It has been verified that the high-emittance iron-titanite coating can be applied to the cast surface of the magnets. High surface emittance is desirable to lower the operational temperature of the magnets. The feasibility of EB-welding the throat-wall supporting splitter-plate has been demonstrated. Tests are currently in progress to confirm the selection of the brazing alloys for the sequential operation of brazing the flexible bus assembly followed by the brazing of the bus to the throat.

F. 4π SHIELD REFLECTOR

A thermal-stress analysis of the reference design control drum was completed. The maximum thermal stress in the BeO was shown to be 11,000 psi at the corners of the BeO block at the absorber-BeO interface. The minimum tensile strength of the BeO is expected to be 14,400 psi. The stress in the BeO at the mounting was calculated to be 10,800 psi, assuming a stress concentration factor of 4.

The detail design of the control-drive assembly, previously described,⁽²⁾ is in progress. The center of gravity and polar moment of inertia of the drum assembly were calculated. The dynamic analysis concluded that the BeO support bands and rivets have a positive margin of safety at the 50-g launch conditions.

The design of the test samples and test fixtures for the determination of the physical properties of BeO, evaluation of the Ta-10W coating performance,

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material-compatibility study of drum materials and coating, and the shock and vibration capability of the BeO and its mounting method were completed. The fabrication of these samples and fixtures is continuing, with many parts completed.

The basis of selecting a special grade of BeO (basic UOX powder plus 0.5% MgO) for the drum was discussed previously.⁽²⁾ The industry response to processing this BeO was poor with only one bidder. The procurement of this special grade BeO for the reflector blocks is being negotiated.

VII. FUEL DEVELOPMENT

One of two primary objectives of this program is to develop fuel elements to meet the requirements for the 600-kwt SNAP8 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.

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%

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.

Figure VII-1. Permeation History of the Eight 15-mil Hastelloy-N-Clad Developmental Elements

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

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A. FUEL-ELEMENT MATERIALS EVALUATION

1. Cladding and Barrier

a. SCB-Coating, Hastelloy-N Performance

The eight 15-mil clad developmental elements have completed the initial 5000-hr thermal endurance run, and are currently being subjected to five "A"-ramp thermal inputs. After these inputs the elements will be subjected to a second 5000 hr of endurance. This group of eight elements consists of three control, three with a complete inner cladding of 304 SS, one with an inner sleeve of 4-mil 304-SS, and the last with an inner sleeve of 2-mil Ni foil containing Gd_2O_3 .

As illustrated by Figures VII-1 and -2, the performance of these elements has been quite good. Figure VII-1 shows the average normalized leakrates of each of the three groups of elements. Figure VII-2 shows the cumulative hydrogen losses of each of the three groups as a function of time at temperature. The double-clad and sleeved elements show a definite improvement in performance over the three control elements. Extrapolating the current trend in hydrogen loss out to 12,000 hr, the control elements indicate a 6.9% loss vs a 3.6% and 1.9% loss for the double-clad and sleeved elements respectively at 1400°F.

Destructive examinations were performed on two Phase-III elements received from the S8DR performance tests after 4000 hr of endurance at 1400°F. Results of the permeation scans of the elements and coated-tubing flex tests are given in Table VII-2, along with the results obtained from Phase-I and Phase-II elements previously examined. (Phase I, Phase II, and Phase III denote the start, midcore, and end of core production, respectively.) The fuel-rod hydrogen-loss data are summarized in Table VII-3. The loss by isochore measurements vs that obtained by the computer-integrated permeation rates agrees fairly well for Element No. 170. More variance between the two methods is noted for Element No. 360, as was the case with Phase-II Element No. 100.

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TABLE VII-2

PERMEATION SCAN AND FLEX SUMMARY, S8DR
PERFORMANCE-TEST ELEMENTS

Phase No.	Element No.	Percent of Total Measured Leakrate*						Coating Adherence Test Number of Flexes*			
		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
I	32	-	-	-	-	-	-	9	5	-	23
	35	38	46	36	14	26	40	3	5	5	23
II	94	32	43	50	18	18	39	7	1	5	23
	100	-	-	-	-	-	-	19	7	-	13
III	170	-	47	-	14	-	39	17	3	-	23
	360	22	47	50	6	28	47	23	23	23	23

*Before and after 4000 hr of aging at 1400°F

TABLE VII-3

S8DR PERFORMANCE-TEST ELEMENT HYDROGEN LOSS
AFTER 4000 hr AT 1400°F

Phase	Element No.	Rod No.	Hydrogen Loss (%)	
			Isochore	Integrated Permeation Rate
I	32	392-D2	2.13	1.83
	35	389-B1	3.70	3.37
II	94	406-A2	6.01	5.42
	100	390-C4	3.73	2.56
III	170	405-C2	2.52	2.38
	360	402-A1	2.03	1.20

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Two biaxial-creep test specimens were fabricated from Element No. 170 and are being placed on test at 1400°F along with the Phase-I and -II specimens. The four specimens from the Phase-I and -II elements have currently exceeded the estimated 1000 hr of time-to-failure by a factor of two, but still fall within the scatterband of previous Hastelloy-N tests. The current biaxial tests are summarized in Table VII-4.

TABLE VII-4
1400°F BIAXIAL-CREEP TESTS S8DR PERFORMANCE-
TEST-ELEMENT CLADDING
(Environment: He + H₂ Internal, Vacuum External)

Phase	Element No.	Sample No.	Stress Level (psi)	Estimated Hours to Failure	Hours Under Test
I	32	32L	11,000	1,000	2,170
		32M	11,000	1,000	2,170
II	100	100L	11,000	1,000	2,170
		100M	11,000	1,000	2,170
III	170	170L	12,800	500	0
		170M	12,800	500	0

All of the numerous cleaning processes currently being used during the processing of cladding components are being reviewed and tabulated along with a list of possible contaminants which are believed to be associated with each cleaning step. Previous tests have shown that residual cleaning agents can produce defects in the glass hydrogen barrier. The various commercially accepted methods used in cleaning metallic surfaces before enameling were reviewed and brought up to date by means of a search of the Journal of Abstracts of the American Ceramic Society. This information is now being compiled and analyzed.

Efforts have also been started on the compilation of data in the classified literature concerning the physical properties of SCB enamel, and in the unclassified literature for data on enamels having compositions similar to that of SCB. Starting with literature issued in 1961, at which time a thorough review

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of enamel properties was made by Andrews, the American Ceramic Society Journal and Abstracts up to and including 1968 were searched. Pertinent abstracts were prepared and are being studied.

The chrome-redistribution study has essentially been completed. Six samples of S8DR Hastelloy-N-clad tubing in the as-coated condition were aged in hydrogen for 5000 hr. Three were aged at 1400°F and three others at 1600. These specimens, along with other samples taken before and after coating, were examined metallographically, by x-ray diffraction, x-ray fluorescence, and electron microprobe to establish what changes in the chromized substrate might occur as a function of aging.

The results of the analyses showed that the S8DR Hastelloy-N cladding and coating material, SCB, is quite stable at 1400°F. Chromium diffusion was found to be negligible, changes in the SCB were slight, and only minor changes in the amounts of $M_{23}C_6$ occurred in the surface of the chromized substrate after the 5000 hr at 1400°F.

At 1600°F significant changes occurred in both the coating and cladding materials. Appreciable bulk diffusion of chromium was observed, the crystallinity of the SCB increased markedly, and grain-boundary precipitation of carbides was observed throughout the entire cross-section of the tube wall. Additional compounds such as M_6C and M_3O_4 were shown to develop in the chrome-rich outer layer.

No significant difference between the rates of grain-boundary-vs-bulk-diffusion of the chromium was observed in either the 1400 or 1600°F specimens. In all cases the diffusion occurred without depleting the chrome in the chrome-rich layer, but rather from the original basic diffusion zone.

b. Shock and Vibration Suppression

A development program is being conducted to develop a damping or suppression device for inclusion within the fuel element to minimize or eliminate damage to the SCB hydrogen barrier and fuel rod resulting from dynamic-load conditions. As previously reported the major portion of damage to the glass coating was found to occur during sinusoidal vibration testing.⁽¹⁾ This mode of dynamic testing is being evaluated both experimentally and analytically with the aid of an analog computer.

The sinusoidal vibration tests performed on various mockup fuel-element configurations were described in Reference 1. Briefly, this test subjected 6 dummy fuel elements to vibration testing to determine and compare the following:

- 1) Attachment to the lower gridplate,
- 2) Absence of suppression device,
- 3) Presence of column-buckling device, and
- 4) 0.5 in. of felt-metal at both ends.

Two of each of the three basic types of elements were subjected to longitudinal sinusoidal vibrations. One of each type of element was attached to the lower gridplate while the remainder were unrestrained with an axial gridplate-element clearance of 41 mils. An accelerometer was attached to the top of each element (blind-end) to record the response.

Table VII-5 shows the sinusoidal vibration inputs used in these tests.

TABLE VII-5
LONGITUDINAL SINUSOIDAL VIBRATION INPUT

Frequency (cps)	Amplitude	Sweep rate
5 to 16	0.316-in. DA	One octave per minute
15 to 2000	5 g (peak)	
2000 to 16	5 g (peak)	
16 to 5	0.316-in. DA	

The up- and down-sweep output-response traces were essentially duplicates with the exception of Mockup No. 9, containing felt-metal. Representative curves are shown in Figure VII-3. These were obtained by averaging the actual response data.

The following conclusions can be made regarding these data:

- 1) Large-scale amplifications occurred with the felt-metal mockups No. 8 and 9. However, it should be pointed out that the response to the upsweep for No. 9 indicates a degree of attenuation before or during compaction of the felt-metal.

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a. Mockup No. 1

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b. Mockup No. 2

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7759-2563-3

c. Mockup No. 3

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7759-2563-4

d. Mockup No. 4

Figure VII-3. Representations of Output Response Traces Obtained from Mockup Elements During Vibration Testing

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e. Mockup No. 8

f. Mockup No. 9

Figure VII-3. Representations of Output Response Traces Obtained from Mockup Elements During Vibration Testing

- 2) The column-buckling device Mockup No. 3 (fixed to the lower grid-plate) shows the most promise. In this case the element follows the input over a fairly wide range of frequencies. The peak reached at about 800 cps is believed to represent a resonant condition in the cladding. At any rate, resonances occurring in this range result in very small displacements and should not be damaging.
- 3) The column-buckling device Mockup No. 4 (floating between the grid-plates) does not appear to be a desirable configuration.
- 4) Unattached elements or fixed elements with the normal 1/4-in. gap, have resonant conditions in the <120-cps range. Beyond this point the elements or rods appear to float between the gridplates of the test fixture.

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a. Mockup No. 1

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b. Mockup No. 3

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7759-2564-3

c. Mockup No. 2

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7759-2564-4

d. Mockup No. 4

Figure VII-4. Results of Study by Analog Computer

Efforts completed during this reporting period on the analytical study of SNAP 8 fuel-element vibration suppressors may be divided into three phases. The primary objective of Phase I was to determine the feasibility of using an analytical model for study on the analog computer. Positive results were obtained. Specifically the results suggested that the "collarbutton" vibration-suppression approach has the best chance for development into a system possessing stable operating characteristics. The felt-metal suppressor concept showed an increased tendency to resonance, which agrees with experimental mockup test data, Figures VII-3e and f. Control of this condition would require additional damping. In this connection a cantilever-beam test was conducted to determine the internal damping capability of two types of felt-metal. The experimental data indicate that its damping capability is not significantly better than that of a solid material. For these reasons felt-metal was eliminated as a primary suppression device. However, the use of a small amount of it remains a possibility as a means of distributing the dynamic-contact loads over a larger surface of the coating.

The primary objective of Phase II was to obtain good correlation between experimental data and output of the analog model. For this purpose two analytical models were prepared, one which simulated a fuel element tied to the lower gridplate with the cladding considered to be rigid, and the other a fuel element (cladding rigid) "floating" between the gridplates. The input levels correspond to the input level of the experimental test being simulated. For each model, damping and stiffness characteristics were varied over a wide range. Simulated on each model were a collarbutton and a "no-suppressor" fuel-element configuration at input frequencies of 24, 40, and 80 cps, respectively. Damping input was varied to determine the lowest magnitude required to achieve stable operating characteristics and to observe the effect it may have on the system's magnification factor. Similarly, spring characteristics were varied to determine the effect of the dynamic-force levels between fuel and cladding.

The results of the analog runs conducted under Phase II are shown in Figure VII-4, directly superimposed upon the representation of data shown in Figure VII-3. The symbols assigned to the parameters shown in Figure VII-3 represent various levels of system stiffness used as input to the analog model,

except parameter "EX" representing the mean value of the associated experimental results. Definitions of system stiffness and corresponding values for the spring constants in the analytical model are as follows:

S Δ Soft system at 1000 lb/in.

1/2H Δ half-hard system at 5000 lb/in.

H Δ hard system at 25,000 lb/in.

CB Δ estimated output representative of a collarbutton at $\sim 50,000$ lb/in.

The analog-output parameter plotted is given in units of acceleration defined as force through the upper fuel-to-cladding interface divided by the fuel mass. The definition of this parameter seems realistic since the experimental correlation response was obtained from an accelerometer mounted at the top of the cladding. The force through the lower cladding-to-fuel interface (force through the suppressor) is generally $\leq 10\%$ higher than at the upper interface, but is not shown in Figure VII-3 for reasons of clarity.

A review of Figure VII-3 and other analog-output data (not shown) supports the following observations:

- 1) The response characteristics indicated by the analog model(s) show better conformance with the corresponding experimental data for a collarbutton system than for a no-suppressor fuel-element configuration.
- 2) If the theoretical collarbutton stiffness ($\sim 50,000$ lb/in.) is used as input to the analog model, the estimated response is indicated by parameter "CB". This response level is approximately three and five times higher, respectively, than the maximum experimental response indicated for a floating and a tied-down cladding. Conversely, in the case of a floating cladding, input values for the effective collarbutton stiffness of about 3000 lb/in. will produce analog outputs closely matching experimental data.
- 3) The analog data clearly suggest that a softer collarbutton effectively reduces fuel-element response.

- 4) The analog results show that on gradual reduction of system damping coefficients the response output remains nearly constant until the damping value is sufficiently small for the system to rather suddenly go out of control. If damping is gradually increased, no significant reduction in response is apparent until unrealistically large damping values are reached. At that time, coupling of masses by damping forces becomes evident, and the response is reduced by 10 to 30% below the prevailing level.

The third phase of development involves the structural analysis of the collarbutton column. For this purpose a computer program was written containing all relevant parameters to evaluate a short column by the secant equation. Output from this program has not as yet been evaluated.

c. Alternate Cladding

Twelve of the 14 Incoloy-800-clad developmental elements continued the 5000-hr 1400°F thermal-endurance run. One of each of the cup-plug and double-blind-end type elements were removed from the test program due to a limitation of test furnace space. Six of each type of element remain on test and have now accrued 2,130 of the scheduled 5000 hr. Figure VII-5 illustrates the permeation history of the 12 elements and compares their performance with the "Special 7". The double-blind-end group appears to be decreasing in leakrate more than the cup-plug group, and after 2000 hr has a lower average rate. Both Incoloy groups now show a definite improvement in performance over the Special 7 group.

Four additional double-blind-end elements have been assembled using enriched (rather than normal) uranium fuel rods. This was done in an effort to establish differences, if any, in permeation performance and the formation of the fuel-barrier reaction product which might result from normal vs enriched fuel. The permeation history of these elements is given in Table VII-6. The average of the elements shows an upward trend in leakrate as a function of time/input. Normally after an endurance test the leakrates will decrease, as did the three groups of elements in Figure VII-1. One element (No. 9099) is the major contributor to the slope of the upward trend. These four elements have also received the post-720-hr "H"-inputs and are being permeation-tested before undergoing a 5000-hr thermal-endurance run.

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Figure VII-5. Incoloy-800-Clad Element Performance

TABLE VII-6
 INCOLOY-800-CLAD DEVELOPMENTAL ELEMENTS,
 DOUBLE-BLIND-END WITH ENRICHED FUEL

Element No.	Normalized Leakrate, cc(STP)/hr at 1400°F and 2 atm H ₂			
	Post-75-hr Thermal Soak	Post-"A" Input	Post-"H" Input	Post-720-hr Endurance
9097	0.45	0.40	0.53	0.51
9098	0.27	0.38	0.46	0.49
9099	0.54	0.64	0.74	1.1
9101	<u>0.24</u>	<u>0.31</u>	<u>0.30</u>	<u>0.51</u>
Average	0.38	0.43	0.51	0.65

The majority of the Incoloy-800-clad tubes have not met the TIR requirements (15 mils maximum over approximately 17 in.) after ceramic coating. Both the pre-oxidation and coating cycles contribute to this condition. This is probably due to a combination of uneven heating and/or cooling rates and the stresses introduced by the differences in coefficients of expansion/contraction between the SCB coating and Incoloy substrate. The stretch straightening over a mandrel method employed with Hastelloy-N tubes cannot be used with the Incoloy tubes due to the pre-oxidized (rather than chromized) ID surface which must be maintained undisturbed for proper coating results. A fixture was designed and fabricated which was intended to combine straightening (by leverage) with the coating firing cycle. Tests using the fixture have thus far been unsuccessful but have been somewhat handicapped by a lack of tubing. Additional tests are planned while other possible straightening methods are being considered.

An order of 750 ft of tubing is scheduled for delivery late in February. Three hundred feet of 5/8-in.-diam bar stock was procured and is now on hand. Both tubing and bar are from the same air-melted heat (HH7531A).

Tests are being conducted 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. This work complements the NAA-83 irradiation tests which are being made to establish irradiation effects. A measure of the ductility of fully processed Incoloy-800 during reactor operating conditions is needed to properly assess the design margin for future reactor fuel elements. The irradiation and NaK tests are being made separately. However, in each test duplicate control specimens are used in vacuum to facilitate cross-correlation of data.

NaK testing is being performed in two isothermal 1400°F loops (shown schematically in Figure VII-6); each loop contains eight specimens. The NaK flowrate is about 16 gal/min. and the temperature is maintained to about $\pm 3^\circ\text{F}$ across the length of the specimen. The configuration of the biaxial-creep specimens employed is shown in Figure VII-7. The cladding material is subjected to the specified processing, and fuel material is included to provide a source of hydrogen.

Figure VII-6. Schematic of NaK Loop

Figure VII-7. Biaxial-Creep Specimen Configuration

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The stress levels were selected for both the NaK and control specimens to produce failure in an estimated 1000 and 5000 hr. The stresses were obtained by extrapolating the log- σ -vs-log-rupture-time relation obtained from biaxial tests performed on as-received material (see Figure VII-8). The as-received material, reported by the vendor to be solution-annealed, has an ASTM grain size of about 8. A typical microstructure of this material is shown in Figure VII-9. The fully processed tubing has additional thermal cycles of 1 hr at 1900°F and about 10 min. at 2150°F, which causes an increase in grain size to ~4. A typical microstructure of this material is shown in Figure VII-10.

The test procedure being used requires that all specimens be removed from loops after initial failures to obtain a measure of diametral deformation and creeprate. These data will be used in a statistical evaluation which may allow early termination of the 10- and 20,000-hr tests. This procedure affords the additional benefit of allowing earlier metallographic examination of ruptured specimens and minimizes potential reactions between NaK-SCB coating-cladding which could mask the failure mode.

An intolerably high O₂ level (~125 ppm) was determined from the NaK sample in the initial 1000-hr loop test concurrent with two specimen failures. All eight samples were removed at this time, the loop was repaired, and eight new specimens were placed on test. The high-O₂ samples were found to have a fairly uniform green deposit of material tentatively identified from x-ray diffraction analysis as NaCrO₃. The failure times obtained from these specimens are shown in Figure VII-8. Analysis of NaK specimens taken during current operation of both the 1000- and 5000-hr loops indicates the oxygen level to be <10 ppm. The validity of the rupture time and ductility data from the initial 1000-hr specimens was questioned because of the possible O₂ effect. Even though the rupture-time data fall near the predicted curve based on as-received specimens, the parallel control specimens have run considerably longer than predicted. The initial failures of the low O₂ test are not significantly different from the initial tests. Therefore, until further resolution of the O₂ effect can be made, the data for both the low- and high-O₂ samples are being retained and are shown in Figure VII-8 and Table VII-7.

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Figure VII-8. NaK and Vacuum 1400°F Biaxial-Creep Tests on Incoloy-800 (fully processed tubing from Heat No. 810535)

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Figure VII-9. Typical Microstructure of As-Received Incoloy-800 Tubing

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Figure VII-10. Typical Microstructure of As-Processed Incoloy-800 Tubing

2. Fuel

a. Burnable-Poison Studies

Vapor deposition was used to coat fuel-specimens with Gd_2O_3 for thermal-endurance testing, compatibility testing, and evaluation of x-ray fluorescence as a nondestructive method for determining the Gd_2O_3 coating thickness. The vapor deposition was performed at Autonetics Division of North American Rockwell (NAR) using a fixture in which five rotating specimens, up to ~ 8.5 -in. long, could be coated simultaneously. Gd_2O_3 powder was placed in an electrically grounded tungsten crucible which was heated by neutron bombardment. The desired thickness of Gd_2O_3 was obtained by controlling (1) quantity of Gd_2O_3 in the crucible, (2) filament power, and (3) deposition time. The quantity of Gd_2O_3 deposited was indicated by a crystal oscillator located in the vacuum deposition chamber. The frequency of this crystal changes as the Gd_2O_3 film is deposited. Glass rods were coated with Gd_2O_3 and the thickness measured to within $\sim 1\mu$ in. to calibrate the crystal. Smooth steel or glass rods were coated at the same time as the fuel rods to provide a relative check on coating thickness. The

TABLE VII-7
DUCTILITY MEASUREMENTS OBTAINED FROM LOW-(<10 ppm)
AND HIGH-(125 ppm) OXYGEN NaK-LOOP SPECIMENS

Sample No.	Nominal Hoop Stress, σ (psi)	Time on Test (hr)	Diametral Deformation (%)	Average Creep rate (%/hr $\times 10^{-3}$)
<u>High-O₂ Loop</u>				
A13	6000	810	2.4	2.96
A66	6000	810	1.23	1.51
A16	6000	810	1.07	1.32
A31	6000	810	1.23	1.51
A42	6000	810	0.84	1.04
A40	6000	810	1.10	1.36
A65	6000	Failed 765	2.67	3.49
A1	6000	Failed 693	2.24	3.23
<u>Low-O₂ Loop</u>				
A109	6000	Failed 788	1.71	2.17
A110	6000	Failed 720	1.06	1.45
A1C2	6000	Failed 834	0.84	1.00
A41	6000	Failed 834	1.00	1.20
A112	6000	Failed 834	1.00	1.20
A111	6000	Failed 834	1.31	1.57
A21	6000	Failed 834	0.40	0.48
A43	6000	Failed 834	1.38	1.65

mechanical device used to measure these very thin films cannot be used for direct measure of thickness on fuel rods due to the roughness of the fuel surface. The thicknesses obtained agree well with the theoretical values based on Holland's equations for a small area source.

Conservation of material is an important consideration in the development of a coating process for deposition of the expensive (\$1400/gm) Gd-155-enriched oxide. The heating method used in this work is inherently more conservative of Gd₂O₃ and also involves less handling of this material than the use of either the Pierce or bent-beam types of electron beam guns. However, the

TABLE VII-8
THERMAL-ENDURANCE TEST MATRIX

Test Temperature (°F)	Test Duration (hr)	Specimen Type	Accumulated Time (hr as of 1/15/69)
1250	1000	A, B, C, F	190
	2000	A, B, C, D	550
	5000	A, B, C, F	190
1400	1000	A, B, C, F	190
	2000	A, B, C, D	550
	5000	A, B, C, F	190
1500	500	A, B, C, D	190
	1000	A, B, C, F	190
	2000	A, B, C, D	550
	5000	A, B, C, F	190

TABLE VII-9
THERMAL-ENDURANCE SPECIMEN DATA

Specimen Type	Surface Finish (rms)	Oxidation Level	Coating Thickness (mil)
A	32	Normal	0.05
B	32	Pre-oxidized	0.05
C	100	Normal	0.05
D	32	Normal	0.10
F	32	Normal	Foil

very low thermal and electrical conductivities in the powder form of Gd_2O_3 gave rise to experimental difficulties which were overcome by make-shift methods, not amenable to incorporation in production coating equipment. Independently of AEC-sponsored work, Autonetics personnel have designed and are currently fabricating a new gun which is expected to eliminate the observed difficulties. The design of production coating equipment will depend on the results obtained during tests of this gun.

Thirty-four specimens of fuel (0.529-in. diam x ~4-in. long) were coated for thermal-endurance testing. These specimens were assembled into ten full-length fuel elements together with similar specimens of noncoated fuel which were wrapped in Gd_2O_3 -bearing Ni foil. These fuel elements have been subjected to S8DR acceptance tests and are undergoing thermal-endurance testing based on the matrix given in Table VII-8. This matrix was designed to yield a comparison of the performance of coated vs foil-wrapped fuel, and to determine the effects of fuel oxidation, fuel-surface finish, and coating thickness on the adherence of the coating to fuel. Table VII-9 gives the surface finishes, average coating thickness, and oxidation treatment of the specimen types listed in Table VII-8. The normal oxidation level is that resulting from air storage of fuel at room temperature for 4 to 6 wk before coating; the pre-oxidized level is that resulting from exposure to air at 400°F for 1 hr followed by room-temperature exposure for 4 to 6 wk. The "F" specimens were undersize (0.524-in. diam) and were wrapped with 2-mil Ni foil containing Gd_2O_3 corresponding to 0.06 mil of coating.

Ten specimens of fuel (0.529-in. diam x ~8.4-in. long) were coated with Gd_2O_3 to thicknesses ranging from ~0.01 to ~0.09 mil for evaluation of x-ray fluorescence as a nondestructive method of determining the thickness of Gd_2O_3 on the fuel. Each specimen was coated to a different thickness and has been cut in half transversely. One half of each specimen was submitted for fluorescence measurements and the other for metallographic mounting and measurement of the thickness from photomicrographs. The results of the two types of measurements will be used to determine the correlation between coating thickness and the x-ray peak heights of Zr and/or Gd.

One side of each of 36 discs of fuel (0.500-in. diam x 0.125-in. thick) was vapor-coated with Gd_2O_3 for use in compatibility testing at 1250, 1400, and 1500°F. Fifteen of these were assembled into stacks of five pairs of diffusion couples as shown in Figure VII-11. The Hastelloy-N discs are 0.500-in. diam x 0.0625-in. thick, and the Ni-foil discs 0.500-in. diam x 0.0020-in. thick. The noncoated side of each Hastelloy disc is oxidized to reduce the adherence of two adjacent pairs of diffusion couples. Each stack of diffusion couples was placed in a compression fixture (see Figure VII-12), inserted into a furnace, and loaded to ~525 psi to assure contact of the several surfaces. The H_2

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Figure VII-11. Assembly of Diffusion Couples for
Compatibility Testing

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Figure VII-12. Compression Fixture
for Compatibility Testing

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overpressure in each furnace is monitored daily to maintain the fuel in the range 1.70 to 1.79 H/Zr. One pair of diffusion couples will be removed at intervals ranging from ~1500 to 15,000 hr and examined by metallographic and electron-microprobe methods to determine the compatibility of Gd_2O_3 and of Ni with the fuel and with SCB ceramic barrier. If possible, diffusion coefficients of the several species will be determined.

B. IRRADIATION TESTS

1. NAA-120-4

The NAA-120-4 irradiation experiment was statistically designed to evaluate SNAP fuel performance as a function of temperature and fuel composition at 0.9 met. at.% burnup. During this quarter the final summary graphs, histograms, and other data-display graphs were generated and the operations report was drafted.

Some difficulty was encountered in having capsules E and H shipped to the Atomics International Hot Laboratory (AIHL) due to new shipping cask regulations which became effective on September 30, 1968. To meet the new requirements a thick laminated plywood cover was made to house the shipping cask. Capsules E and H were finally shipped during the latter part of December. All capsules are now at the AIHL and the postirradiation examination has been started.

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 H_2 content will be used to evaluate the effect of δ -to- β -phase transformation after significant fission-product accumulation. An out-of-pile test is being made to differentiate between thermal and radiation effects on fuel-element behavior. This test consists of subjecting four fuel elements to thermal treatments which simulate as nearly as possible the thermal treatment of the in-pile elements.

During this quarter, irradiation and out-of-pile tests continued. The Hanford reactor power has risen about 10% during this quarter and the present experiment power level is back up to approximately 80% of design power.

The processing of data into condensed files is current. Review of these data indicates the experiment operation has been stable with no apparent anomalies. One additional thermocouple has failed during this quarter leaving 52 operating thermocouples out of the initial 58. Due to excessive problems in operating and interpreting magnetic data tapes, the magnetic recorder has been removed from the 121 system and returned to the manufacturer for repair. The out-of-pile fuel element tests continued; 5500 operating hours have been accumulated.

3. NAA-140-1

The primary objective of the NAA-140-1 irradiation experiment is to obtain irradiation-performance data on parameters representing future long-life fuel elements. This experiment will be designed to operate for 20,000 hr at design temperatures, and will contain fuel elements operating under three different conditions:

- 1) The full-length (ZrH-length) elements will have temperatures and power profiles representing the center element in a 600-kwt (295-element core) ZrH reactor with 1300°F NaK-outlet temperature.
- 2) The gradient elements will have a power representing the peak power of the above-referenced reactor and a peak cladding temperature of 1350°F and a ΔT of 100°F over 70% of the length of the cladding.
- 3) The isothermal elements will have a power that corresponds to ~2.25 times that of the peak in the reference reactor and a cladding temperature of 1300°F over 70% of the length of the cladding. This high power is required to obtain sufficient fuel-growth to strain these elements during the irradiation.

The detailed heat-transfer analysis that was begun last quarter continued through this report period. A thermal model was developed for each of three different fuel elements, using the techniques developed in the NAA-121-1 study. These analyses use the computer code TAP. An additional technique in these analyses that was not used in the 121 study is an addition to the code that sizes the control-gas annulus. This technique yields preliminary values in one computer run through an iteration process for the gas-annulus dimensions which

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are then refined to yield the design temperatures. The detailed heat-transfer calculations for the NAA-140-1 irradiation experiment are approximately 70% complete.

An analysis was also made to evaluate the temperature effect of thermocouples banded against the cladding. This was performed on the isothermal elements since they have the highest power and consequently the greatest effect. The results indicate that the temperature of the cladding directly under a thermocouple is spiked by approximately 5°F, and consequently should cause little concern.

Analyses were also carried out to evaluate the temperature effects of the fuel being off-center in the cladding which results from the elements being irradiated in a horizontal position. An R - θ analysis on an isothermal element which has an assembled diametrical fuel-cladding gap of 5 mils indicates a circumferential ΔT of $\sim 116^\circ\text{F}$ in the cladding. An R - θ analysis on a gradient element which has an assembled diametrical gap of 12 mils indicates a circumferential ΔT of $\sim 100^\circ\text{F}$ in the cladding. These analyses assumed an intimate contact between the fuel and H₂ barrier on the bottom side and no fuel-rod bowing. An analysis is in progress to determine the axial bowing which results from fuel-rod circumferential ΔT and the degree to which the fuel centers itself in the cladding

During this report period a study was made to determine the data-acquisition system requirements for this irradiation experiment and to evaluate the systems available to AI. Since this test will remain in-pile for approximately 3-1/2 yr, the system must be reliable and maintainable. In this study, existing data-acquisition systems were screened for accuracy, reliability, and maintainability. The resulting conclusions were:

- 1) The systems now at Hanford are not suitable for use because of limited capacity and accuracy, and potential maintenance problems associated with long-term use.
- 2) An alternate data-logging system located at AI Headquarters should be modified to provide the features required.

The detailed nuclear analysis for this experiment has continued during this report period and the calculations are ~90% complete.

The hydraulic analysis of this experiment has been completed.

Processing of cladding components for the NAA-140 fuel specimens was begun and, in general, has progressed to the point where design dimensions are required for final machining and assembly. The number of available components and the processing status are summarized in Table VII-10.

TABLE VII-10
NAA-140 FUEL-SPECIMEN COMPONENT STATUS

		Number of Components at Each Stage of Processing				
		Minimum Required	Chromized and Diffused	Pre-Oxidation Sandblasted	Oxidized	Coated
Hastelloy-N	Full-length tubes	2	19	-	-	8
	Sublength tubes	4	*	-	-	0
	Cup-plugs	5	15	-	-	0
	Drilled cup-plugs	1	**	-	-	0
Incoloy-800	Full-length tubes	1	-	15	0	0
	Sublength tubes	5	-	8	0	0
	Cup-plugs	2	-	10	0	0
	Closure end-caps	2	-	10	0	0
	Drilled closure end-caps	2	-	**	0	0
Fuel rods		12	20 hydrided, ready for final machining			

*To be cut from full-length tubes

**Standard plugs to be drilled

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. Even though SNAP fuel elements are designed for no fuel/cladding contact, the allowable cladding ductility must be known so that design margin can be assessed.

During this quarter, all ORR-209, ORR-211, and ORR-212 test specimens and the thermal controls were fabricated; the ORR-209 test assembly was completed and inserted in the P-5 position of the Oak Ridge Reactor (ORR).

The reactor was started up on January 18, 1969, a week later than scheduled. Because it will shut down as previously scheduled, the experiment will receive 150 to 200 hr less exposure than planned. Since the stress for the in-pile rupture specimens was specified to produce failure in 1000 hr with 97.7 probability (mean time-to-failure minus two standard deviations), the design pressures will still produce in-pile failures with better than 95.5% probability. Therefore, no change in the test pressures was made for the shorter cycle time.

The thermal and fast-flux profiles for the test position are shown in Figures VII-13 and -14. The absolute fluxes may vary from those shown, which are based on 1964 data, but the shapes should still be valid. The actual fluences received by the specimens will be determined with a series of Fe, Ni, Al-0.1% Co and Cd-covered Al-0.1% Co dosimeter wires, recessed in the face of the assembly enclosure box.

Fabrication of the equipment to test the control and post-irradiation creep-rupture specimens is in progress.

C. FUEL-ELEMENT TESTING

1. Preproduction-Element Performance Tests

The 12 S8DR preproduction fuel elements completed the scheduled 10,000-hr endurance test at 1400°F isothermal during this report period. The performance of these elements for the cumulative 11,216 hr of testing time (including pre-endurance tests) is shown graphically in Figure VII-15. The average hydrogen-permeation rate normalized to 2 atm of H₂ pressure and barrier-performance-parameter (K) plots show a small increase with time. However, the overall performance is considered to be excellent. The total H₂ loss was 4.97% at 1400°F or about 1.66% when adjusted to S8DR core-temperature profiles. This is well below the 5% H₂ loss in 10,000 hr allowable in the S8DR core.

Six of the 12 fuel elements were selected at random for continued endurance testing at 1400°F isothermal for an indefinite period, dependent upon availability of space in the endurance system. Because of the limited number of retorts now

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Figure VII-13. Thermal Flux, ORR Poolside Positions 5 and 6,
30-Mw Case

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Figure VII-14. Nickel Flux (>2.9 Mev)-ORR Poolside Positions 5 and 6, 30-Mw Case

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Figure VII-15. S8DR Preproduction Fuel Element Performance Data

TABLE VII-11
ZONE-SCAN PERMEATION-TEST DATA FROM
S8DR PREPRODUCTION FUEL ELEMENTS

Element No.	Percent of Total Permeation		
	Closure End (1-1/4 in.)	Center (14-7/8 in.)	Blind End (1-1/4 in.)
7121	42	9	49
7138	43	11	46
7159	39	9	52
7161	37	25	38
7162	43	13	44
7173	45	9	46
Average	41.5	12.7	45.8

available these six fuel elements are clustered in groups of three in two retorts. The fuel-element permeation rates will be computed as a ratio of the total H_2 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 1000 hr in this extended endurance cycle and now have accumulated more than 12,200 total hours at 1400°F.

The remaining six fuel elements are now undergoing postendurance evaluation. All have been zone-scan permeation-tested and are now being decanned for isochore testing of the fuel and evaluation of the cladding tubes. The results of the zone-scan tests, summarized in Table VII-11, show that most of the postendurance hydrogen permeation was from the ends of the fuel elements with only a low percentage from the center sector.

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 7800 of the scheduled 20,000 hr of endurance testing at 1400°F; Phase-II elements, midcore production, 7100 hr; and Phase-III, end-of-core production, 5500 hr. Phase-V elements, which are being endurance-tested at 1300°F, have completed 6300 hr.

Figures VII-16, -17, and -18 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-19. 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 Input-"H" cycles (Figure VII-20) at 2000-hr intervals in the endurance test program. In pre- and post-Input-"H" permeation tests for current endurance cycles, nine of the 30 S8DR fuel elements in test had a change in permeation rate of less than 0.02 cc/hr with the maximum change being only 0.11 cc/hr. A frequency-distribution analysis showed a nearly normal distribution indicating that there was no significant effect of thermal cycling on the hydrogen-permeation rates. These data all indicate that the ceramic H₂ barrier in S8DR fuel elements is not degraded by thermal cycles, and provide confirming evidence that the fuel elements can withstand all of the temperature cycles and transients probable during S8DR operation.

In the S8DR fuel-element performance-test program two elements are removed from test periodically for detailed destructive test and evaluation. To date, two elements each have been removed from Phase I, -II, and -III tests upon completion of the 4000-hr endurance and postendurance permeation tests. The destructive tests and evaluations are performed under the cladding barrier task and were reported in B-1a of this Section.

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Figure VII-16. S8DR Production Fuel-Element Hydrogen-Barrier Performance Compared with "Special Seven" and Preproduction Fuel Elements

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Figure VII-17. S8DR Production Fuel-Element Hydrogen Permeation Compared with "Special Seven" and Preproduction Fuel Elements

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Figure VII-18. S8DR Production Fuel Element Total Hydrogen Loss Compared with "Special Seven" and Preproduction Fuel Elements

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Figure VII-19. S8DR Production Fuel-Element Performance Parameters of Phase-V Endurance Tests at 1300°F Compared with 1400°F Tests

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● ACCEPTANCE TEST →
FIVE HIGH-RATE TRANSIENTS AFTER
75-hr AGING TO DEMONSTRATE EACH
ELEMENT'S THERMAL-SHOCK RE-
SISTANCE. TREATMENT GIVEN TO
REMOVE MARGINAL ELEMENTS.

● PRODUCTION TEST
TOTAL OF 10 CYCLES SIMULATES
OVER ONE-HALF THE MAXIMUM
EXPECTED CYCLES OF THE S8DR.

- RAMP 1 SIMULATES REACTOR
STARTUP, 10 TEST CYCLES OF
55 EXPECTED.
- RAMP 2 SIMULATES Hg INJEC-
TION, 20 TEST CYCLES OF 10
EXPECTED.
- RAMP 3 SIMULATES OSCILLA-
TIONS, 60 TEST CYCLES OF 50
EXPECTED.
- RAMP 4 SIMULATES A COLD
SLUG, 10 TEST CYCLES OF
ABOUT 25 EXPECTED.
- RAMP 5 SIMULATES LOSS OF
HEAT REJECTION, 10 TEST
CYCLES OF 10 EXPECTED.
- RAMP 6 SIMULATES LOW-FLOW
SCRAM, 10 TEST CYCLES OF 18
EXPECTED.
- RAMP 7 SIMULATES HIGH-
POWER SCRAM, 10 TEST CYCLES
OF 5 EXPECTED.

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Figure VII-20. S8DR Production Fuel Element Test Cycles

VIII. AEROSPACE SAFETY

A. INTRINSIC SUBCRITICALITY ANALYSIS

A computer program was developed and checked out for use in analyzing the transient experiments to be performed on the SNAPTRAN machine. The program, called STAC, reads and translates the symbolic data tapes produced by AI's Analog-to-Digital Converter (ADC), provides edits and plots, and stores the data to be used for analysis. It then correlates the various power and temperature measurements into a composite total-power and mean-temperature history. Reactivity and temperature coefficient of reactivity are also obtained. The code has been extensively checked using synthetic data corresponding to transients both below and above prompt critical, and also using data from an actual transient as recorded on strip charts.

The two synthetic tests produced STAC results which agreed very well with the original theoretical data on which they are based. The analysis of the actual transient gave results which likewise agreed well with those obtained in the SNAPTRAN-1 and -2 experiments, in spite of the rather crude method of obtaining the data.

Another computer code was developed to calculate reactivity on a real-time basis. This program can be translated to a symbolic language usable by the storage module of the ADC. Thus the ADC can be put to valuable use as an on-line reactivity monitor when it is not busy digitizing tapes.

An apparent disagreement between measured and calculated gadolinium worth in the SCA-4B environment ($\sim \$9.00$ vs ~ 10.50 measured) has not been completely reconciled. A mathematical model was developed to provide insight into the core reactivity held down by boron-containing splines. A symmetric 37-spline core was calculated in which the spline worth was determined to be $\sim \$10.40$. The experimental data indicate an excess reactivity of $\$11.00$ with 36 splines. As the boron content of the splines is not well defined at present, this effort has not been conclusively studied.

Recent Gd cross-section compilations are in the process of being adopted for use in the calculational predictions. These compilations along with ENDF (Evaluated Nuclear Data File) cross-sections have been incorporated into a new

28-group SNAP library. Initial checkout of the library against SNAP critical assemblies has demonstrated the superiority of this set over earlier compilations. Recalculations of the IWS (intrinsic water subcriticality) experimental checkpoints of the SCA-4B and SNAPTRAN machines will be made along with completion of the boron spline analysis to provide further insight into the use of Gd for intrinsic subcriticality applications.

B. INTRINSIC SUBCRITICALITY EXPERIMENTS

Approximately 150 ft of three types of Ni foil bearing various amounts of enriched Gd_2O_3 -155 were received for use in making sleeves for the intrinsic subcriticality experiments. All three foils met the specified Gd_2O_3 content, as shown in Table VIII-1.

TABLE VIII-1
 Gd_2O_3 -155 CONTENT OF FOILS PRODUCED

Type of Foil	Gd_2O_3 -155 Specified (wt %)		Gd_2O_3 -155 Produced (wt %)
	Average	Tolerance About Average	Average and Tolerance $\bar{X} \pm 2\sigma$
II	0.713 to 0.873	± 0.05	0.786 ± 0.02
III	1.495 to 1.655	± 0.05	1.587 ± 0.05
IV	2.292 to 2.452	± 0.05	2.399 ± 0.05

None of the foils exactly met the thickness specified but they were considered to be as close as practical to the rolling-process capability and were adequate for the planned physics experiments. All of the foils met the width requirement. Dimensional data on foils used to fabricate sleeves are shown in Table VIII-2.

The fabrication of sleeves for intrinsic subcriticality experiments was completed. The sleeves were fabricated to be 1.260 ± 0.002 -in. ID by 12.375 ± 0.025 -in. long. Initial use of the sleeves indicates satisfactory fit on the fuel elements.

The initial series of tests on the SNAPTRAN reactor was concluded with a set of five temperature-limited transients. Reactivity inputs up to \$1.30 were used to start the transients. The peak powers and energy releases fall within

TABLE VIII-2
DIMENSIONAL DATA ON FOILS PRODUCED

Type of Foil	Thickness Required (mils)		Thickness Produced (mils)	Widths Required (in.)	Widths Produced (in.)
	Average	Tolerance About Average	Average and Tolerance $\bar{X} \pm 2\sigma$	Average and Tolerance	Average and Tolerance $\bar{X} \pm 2\sigma$
II	1.75-2.25	± 0.15	2.19 ± 0.17	4.25 ± 0.1	4.253 ± 0.004
III	1.75-2.25	± 0.15	2.23 ± 0.16	4.25 ± 0.1	4.252 ± 0.002
IV	1.75-2.25	± 0.15	2.27 ± 0.20	4.25 ± 0.1	4.251 ± 0.002

the range of expected values from analytical studies. Some of the data were used to check out the code STAC. Final analysis of these data is awaiting the completion of the installation of a data digitizing and recording system. This system, the components of which have been received, will digitize the reactor-transient data directly from the FM analog tapes by frequency counting, a process which will preserve the accuracy of the original data. The data will be recorded on digital magnetic tape for processing by the STAC code.

The operation of the SNAPTRAN reactor has been resumed after a programming pause for the water-immersion tests. The reactor was checked out and the base core configuration was reloaded. The first concentration of Gd-Ni sleeves ($\Sigma_a t = 0.067$) was placed over the fuel elements and a change in reactivity of \$2.77 was measured. This value is in good agreement with the predicted value of \$3.00. The measurements will continue with reactivity coefficients and isothermal-temperature coefficients for each of the Gd concentrations.

Measurements using the SCA-4B critical assembly to study water-immersed, Gd-poisoned cores were completed. Gadolinium-155 in a Ni substrate was used to reduce the reactivity of a water-immersed SNAP reactor core. It was found that a $\Sigma_a t$ of 0.208 was sufficient to keep the reactor subcritical with water both reflecting and filling the interstitial spaces in the core. A graph of the excess reactivity of a SNAP 10A core as a function of Gd content is shown in Figure VIII-1.

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TABLE VIII-3
REACTIVITY OF SPECIAL CORE CONFIGURATIONS

Configuration	Excess Reactivity (\$)
Beryllium reflector + $\sum_a t = 0.15$ sleeves	9.1
Beryllium reflector + $\sum_a t = 0.23$ sleeves on central 19 elements only	11.9

The effect of Gd in beryllium-reflected cores and in nonuniform loadings was also studied. The results are summarized in Table VIII-3.

The water-immersion phase results established the amount of Gd required to achieve subcriticality in a water-immersed SNAP reactor. The measurements in the operational configuration using the SNAPTRAN reactor will determine the trade-off factors such as the loss in cold clean reactivity and the change in the prompt-temperature coefficient.

Figure VIII-1. Excess Reactivity of Water-Immersed SNAP 10A Core as a Function of Gd-155 Concentration

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