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SNAP REACTOR PROGRAMS
PROGRESS REPORT
AUGUST-OCTOBER 1971

AEC Research and Development Report

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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) SPF Zirconium Hydride Reactor Engineering
- 2) Reactor Development
- 3) Fuel Development
- 4) Thermoelectric Converter Development
- 5) Reactor-Thermoelectric System Engineering.

The SPF Zirconium Hydride Reactor Engineering effort provides for the engineering and design of a reactor to be tested in the NASA Space Power Facility at Plum Brook. The design power level will be 300 kw, with an outlet coolant temperature of 1200°F.

The Reactor Development tasks include heat transfer and hydraulic tests in support of the SPF reactor, as well as the development of the control drive components for the reactor.

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

The Compact Thermoelectric Converter Technology Program is being carried out 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 System Engineering tasks provide for reactor power system and component design and analysis. Major emphasis during this quarter was devoted to unmanned reactor thermoelectric systems in the 1- to 5-kw power range.

B. SUMMARY OF PROGRESS

SPF reactor design and performance analyses continued during the quarter on the 199-fuel-element core. Core thermal and hydraulic studies were conducted to further examine worst case core thermal conditions as the basis for fuel element and thermal stress analysis. Fin-on-tube structural tests were conducted to evaluate fin geometries and cladding contact stresses for both the Hastelloy X and Incoloy 800 fuel element configurations. The margins of safety for Hastelloy X are considerably greater in the as-fabricated conditions and the core assembly condition, as well as during normal operation. Trade-off studies were performed on burnable poisons with the results favoring a 50/50 mixture of natural Gd and Eu¹⁵¹ for the SPF reactor.

Additional NaK heat transfer testing was performed with smooth heaters simulating fuel elements spaced at a P/D of 1.05. The experimental results continue to be in good agreement with the analytical predictions.

An S8DR design actuator was subjected to vibration loads expected during launch of a Saturn V booster. No damage or effects on performance were detected at these loads or at loads several times higher. Design studies on the drum drive and launch lock mechanism, and on metallic sheathed electrical cables continue.

Effort continues on evaluation of hobbing techniques, weld parameters, and final closure parameters for both Hastelloy-X and Incoloy-800-clad fuel elements. A second group of 20-mil, smooth-wall Hastelloy-X-clad elements was assembled. Mechanical and thermal testing of these fuel elements continues.

Test results on the TEM-X series thermoelectric power modules continue to demonstrate the excellent performance stability of this module. Four modules have accumulated between 2500 and 4500 hours of operation at 1085° F, including thermal cycling, with essentially no power degradation. Testing of prototype pump modules is approaching 24,000 hours with total power degradation in the range of 3 to 4% per 10,000 hours.

II. SPF ZrH REACTOR ENGINEERING

A. INTRODUCTION AND SUMMARY

SPF reactor design and performance analyses continued during the quarter. The reactor design is based on operation at 300 kwt and 1200°F in the NASA Space Power Facility (SPF) at Plumbrook. The reactor core consists of 199 fuel elements arranged in a cylindrical geometry. Eight beryllium void-backed control drums are employed to control reactivity throughout the reactor life of 5 years.

Core thermal and hydraulic studies were conducted to further examine worst case core thermal conditions as the basis for fuel element thermal stress analysis. Random and preferred fuel element cluster patterns were studied including the effects of fuel segment asymmetry. Thermal and hydraulic studies were also conducted to explore the relative advantages and disadvantages of alternate internal reflector designs.

Fin-on-tube structural tests were conducted to determine optimum fin geometries and minimize cladding contact stresses. Cladding stresses were evaluated and design safety margins determined for both the Hastelloy X and Incoloy 800 fuel element configurations. The margins of safety for Hastelloy X are considerably greater in the as-fabricated conditions and the core assembly condition, as well as during normal operation.

Nuclear parametric studies were performed to further define the advantages and disadvantages of incorporating burnable poison combinations in the core design. Considering the trades between control margin requirements, burnout characteristics, temperature coefficient effects, burnable poison availability, material compatibility, and processing techniques, the initial burnable poison loading was selected as a 50-50 mixture of natural Gd and Eu¹⁵¹.

B. MECHANICAL DESIGN

The conceptual design parameters for the 300-kwt SPF reactor are summarized in Table 1. These parameters are independent of the final section of the cladding material. For equivalent nuclear performance, the cladding thicknesses are 0.028 and 0.032 in. for Hastelloy X and Incoloy 800, respectively.

TABLE 1
SPF REACTOR DESIGN SUMMARY

<u>Reactor Core</u>	
Number of Elements	199
Lattice Pitch	0.737 in.
Core Vessel ID	11.57 in.
Fuel Length	18 in.
Number of Elements (Inner/Outer Zones)	85/114
Hydrogen Barrier	2 mil SCB
Fuel-to-Barrier Gap (Diametral)	12 mils
Fin Type	Right-Left-Neutral
Number of Fins Per Element	3
Fin Pitch	5.5 in.
Fuel Element Spacing (Inner/Outer Zones)	30/15 mils
Internal Reflectors	316 SS + NaK
Grid Plate Material	316 SS
Core Vessel Material	316 SS
Coolant Inlet and Outlet Lines	4 each
Line Size	2 in. OD x 0.065 in.
<u>Reflector</u>	
Control Drums	8 Be Void-back
Drum Radius	3.52 in.
Drum Thickness	3.52 in. $\pm$ 1 in. shim
Drum Rotation Angle	135° Normal 145° to Lockout
Drum Envelope Diameter	26.1 in.
<u>Control Drives</u>	
Control Drum Actuator	Modified S8DR
Gear Ratio	12/1
Lockout	Actuator released
Drum Step size	0.6°

A sketch of the SPF reactor, arranged within a 4π shield, for test in the NASA Plumbrook Facility is presented in Figure 1. Further details of the SPF reactor design are presented in Figures 2 through 4. The reactor plan view, shown in Figure 2, illustrates the arrangement of the eight beryllium control drums and the two-zone core configuration. Figure 3 further illustrates the installation of the in-core thermocouples to monitor core performance during reactor operation. Figure 4 illustrates the control drum drive system arrangement.

The control drum drives are located outside the primary reactor shield and are connected to the drums by extension shafts. The drive for each drum consists of an actuator, a 12-to-1 spur gear reducer, and a lockout mechanism which is released by the initial rotation of the actuator. The actuators are identical to those used on the S8DR reactor except for details of mounting and electrical termination.

C. CORE THERMAL AND HYDRAULIC ANALYSIS

The possible combinations of fuel element bow and fuel segment asymmetry are limitless in a 199-element reactor core. In an attempt to bound the potential thermal gradients which may be imposed on individual fuel elements, eleven selected cluster and asymmetry configurations were analyzed using a computer model simulating a sector of the core. Two cases were run using cluster and asymmetry direction vectors generated by a random number generation routine. Two cases were run using cluster vectors indicated by the core cross element ΔT 's in a symmetrically configured core; one with random fuel segment asymmetry and the other with fuel segment asymmetry vectors in the same directions as the cluster. Seven other cases were run with coincident cluster and asymmetry vectors in various configurations designed to force thermal gradients. The following are the basic ground rules used in the cluster study:

- 1) Elements move in a specified direction until they contact a neighboring element. The element does not displace a neighboring element.
- 2) Fuel segment asymmetry is assumed to occur in the same direction over the full length of the fuel element. Fuel movement is 5 mils from concentricity.

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Figure 1. SPF Reactor Installation

Figure 2. SPF Reactor Plan View

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Figure 3. SPF Reactor Core and Vessel

Figure 4. Control Drum Drive

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Figure 5. Natural Cluster with Reinforced Fuel Segment Asymmetry
(Cluster No. 3)

- 3) Fuel element bow shape is assumed to be circular arc.
- 4) Fins are undercut 5 mils at the cold end and 8 mils at the hot end of each finned element.
- 5) The average fin height is 3.25 mils less than nominal fuel element spacing.
- 6) Mixing slippage factors are: 0.7 at the cold end, 0.9 at the center support plane, and 0.6 at the hot end.
- 7) Assumed bow direction does not necessarily coincide with the vector caused by thermal gradient.

The model also limited boundary element (elements on the outer ring of a 60° sector) to movement in two directions rather than the twelve allowed for all other elements. Further, the model assumed a mirror image occurs across the sector boundary. The model, however, is sufficiently extensive to accurately evaluate the thermal gradients developed in each fuel element under any conditions of clustering and fuel segment asymmetry with the possible exception of the center element in the core which must always be symmetric.

Figure 5 illustrates a natural cluster - asymmetry configuration in which the fuel elements were allowed to bow in the direction of the symmetrically configured thermal gradient vector and the fuel segments were asymmetrically located to reinforce this bow pattern. This figure shows the direction and amount of maximum bow by arrowheads and the direction of asymmetry by dashed circles offset from element centers. This configuration produced the maximum cross element ΔT of all the configurations studied.

The cross element ΔT distribution for this cluster configuration is shown in Figure 6. The element number is also presented for ready reference to the model sketch. The 60° segment is shown divided into two 30° segments with the left half of the model sketch shown as a negative normalized radius (r/R) and the right hand as positive r/R . Those elements on the dividing line are shown as both positive and negative. The neutral elements are shown as solid points and the maximum ΔT is flagged by a dashed circle. The cross element ΔT 's for a completely symmetric core and no fuel segment asymmetry is shown by a dashed line for reference.

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Figure 6. Cross Element ΔT 's for Natural Cluster with Reinforced Fuel Segment Asymmetry

In most instances, the peak cross element ΔT occurs at the coolant outlet end of the element. However, the location is dependent on cladding bow and mixing effects.

Neutral elements, in general, exhibit larger cross element ΔT 's than finned elements. The finned elements have a complete circumferential sweep of coolant around their periphery enhancing a good average of coolant temperatures. The neutral elements are exposed to a mixing effect consisting essentially of three sweeps of 1/3 periphery each. There is a lower degree of interchange between the 6 channels surrounding a neutral element than there is between the 6 channels adjacent to a finned element and, therefore, a larger thermal gradient on the unfinned elements.

Table 2 summarizes the results of the cluster study. Of the 11 cases run, only one (Case 10) had a peak ΔT on a finned element and only three (Cases 4, 8, and 9) had peak ΔT 's in the inner zone of the core.

The eleven cluster and asymmetry patterns studied are considered to encompass the probable thermal environment conditions in the SPF reactor. Fuel

TABLE 2
CLUSTER STUDY SUMMARY

Case Number	Element Cluster Description	Fuel Segment Asymmetry	Element Number with Peak ΔT	Peak ΔT ($^{\circ}$ F)
1	Random	Random	29	25.0
2	Random	Random	31	21.8
3	Natural	Reinforcing	27	32.0
4	Natural	Random	9	22.9
5	Neutral Element	Reinforcing	30	31.3
6	Four Element	Reinforcing	27	28.3
7	Total Inward	Reinforcing	28	18.0
8	Row	Reinforcing	17	20.9
9	Diagonal	Reinforcing	4	20.7
10	7 and 3 Element	Reinforcing	21	23.1
11	Ring	Reinforcing	29	30.6

Figure 7. Fin Profiles Tested

rod asymmetry has been shown to have a much greater influence than clustering on equivalent cross element ΔT . It has also been shown that asymmetry does not appreciably affect peak fuel temperature. In most instances, peak temperature is decreased slightly.

In the majority of the configurations examined, maximum cross element ΔT was associated with a neutral element. However, finned elements also exhibited appreciable ΔT 's. In addition, most of the high ΔT 's occurred in the outer zone.

D. STRESS ANALYSIS

1. Local Contact Strain Test

A test was conducted to determine the strain on the inside surface of the fuel element cladding due to the fin from an adjacent element contacting the element in question and applying a force to it. The strain on the inside of the cladding tube is considered to be of primary importance because of hydrogen barrier degradation which may be caused by excessive strain in the cladding. Ten strain rosettes were installed in the instrumented tube to map the strain pattern. The instrumented tube was 0.032-in. thick with an outside diameter of 0.748 in.

The test objective was to determine the maximum strains (or stresses) on the inner surface of a cladding tube for a variety of fin geometries and to compare the results of the tests to the theory of stresses induced by a pad load on a cylinder which was developed by Bijlaard.⁽¹⁾

The fin geometries tested were on a 5-1/2-in. pitch helixed around a 2-in. segment of a tube. This tube segment was then inserted on a steel mandrel and pushed against the strain-gaged tube. The fin profiles tested are shown in Figure 7. The first test was with a 0.072-in.-wide fin. This fin was machined flat for the second test. The same procedure was followed with a 0.125-in.-wide fin. The 0.125-in.-wide fin was also machined with a reversed curvature in the fin. With the axis of the radius of curvature parallel to the tube axis, radii of 0.4 and 0.5 in. were machined into the fin. A 0.160-in. fin was also tested.

The tests were conducted by applying different loads to the mandrel causing the fin to contact the instrumented tube. A data reduction computer program

Figure 8. Test Results — Maximum Stresses, 0.072-in. Fin

was written to reduce the strain data from each rosette and compute the stresses. Typical results for the initial 0.072-in. fin test are shown in Figures 8, 9, and 10.

A summary of test results for the various fin configurations is given in Table 3.

2. Fuel Element Stress Analysis

Fuel element stresses were computed for the reference fuel element and core configurations based on using either Hastelloy X or Incoloy 800 as the cladding material. A summary of the fuel element design parameters and the stresses which are developed in each design as a result of fabrication are summarized in Table 4. The proportional limit is assumed the same as the elastic limit for determining the margin of safety against thermal ratcheting.

When the fuel elements are assembled in the core, it is assumed that a 0.012-in. TIR exists on each element and that two adjacent elements are arranged in their most adverse position. The summary of fuel element stresses in the core assembly configuration is shown in Table 5.

Fuel element operational stresses were computed for the element with the worst thermal environment. From the results of the core thermal hydraulic studies, stresses were evaluated for fuel elements with a temperature gradient of 32°F. The summary of the short-term operational stresses, based on 300 kw and 1200°F outlet temperature, is presented in Table 6.

3. Fin Undercut Requirements

The fins on each fuel element are undercut in the regions away from the chosen contact point (11.7 in. from the cold end) to prevent secondary contact points and the development of excessive stress levels in the fuel elements. Figure 11 illustrates a potential fuel element bow pattern (fin undercut requirements) to prevent secondary contact with an element temperature gradient of 32°F.

If a secondary contact does occur, through a tolerance condition or change in cross element temperature gradients, the effect is shown in Figure 12.

Figure 9. Circumferential Stress with Angular Position, 0.072-in. Fin, 30-lb Load

Figure 10. Circumferential Stress with Axial Position, 0.072-in. Fin, 30-lb Load

TABLE 3
FIN-ON-TUBE STRESS SUMMARY

Fin Configuration	Circumferential Stress Per Pound Load (psi/lb)	Axial Stress Per Pound Load (psi/lb)
0.072 in.	1030	812
0.072 in. flat surface	855	710
0.125 in.	970	750
0.125 in. flat surface	855	600
0.125 in., 0.4 in. R reverse curvature	720	580
0.125 in., 0.5 in. R reverse curvature	700	650
0.160 in.	950	875

TABLE 4
AS-FABRICATED STRESSES

	Hastelloy X	Incoloy 800
Geometry (in.)		
Pitch	0.75	0.75
Mean Radius (Outer zone)	0.354	0.3515
Tube Thickness	0.027	0.032
SCB Thickness	0.0025	0.0025
Processing Stress (psi)		
Thermal Cooldown	6625	6960
Eccentricity (SCB)	860	900
Eccentricity (Tube)	230	245
Total Stress in Metal (psi)		
Hoop	7715	8105
Axial	6990	7340
Equivalent	7380	7750
Margin of Safety		
Proportional Limit (psi)	33,900	16,900
Margin of Safety	3.59	1.24

TABLE 5
CORE ASSEMBLY STRESSES

		Hastelloy X	Incoloy 800
Contact Side of Element, Outside Surface (psi)			
Bending Compression		-2225	-1990
Contact Stress	Load	5#	5.25#
	Hoop	-7940	-6030
	Axial	-5100	-3960
As Fabricated	Hoop	7715	8105
	Axial	6990	7340
Equivalent Stress		285	1830
Contact Side of Element, Inside Surface (psi)			
Bending Compression		-2225	-1990
Contact Stress	Hoop	6110	4430
	Axial	2880	1970
As Fabricated	Hoop	7715	8105
	Axial	6990	7340
Equivalent Stress		12,000	10,900
Side Opposite Contact (psi)			
Bending Tension		2225	1990
As Fabricated	Hoop	7715	8105
	Axial	6990	7340
Equivalent Stress		8560	8780
Margin of Safety			
Proportional Limit (psi)		33,900	16,900
Margin of Safety		1.82	0.55

TABLE 6
SHORT-TERM OPERATIONAL STRESSES

		Hastelloy X	Incoloy 800
Contact Side of Element, Outer Surface (psi)			
Bending Stress		-4614	-5176
Six Theta Stress		-1720	-2032
Contact Stress	Hoop	-19,050	-18,100
	Axial	-12,200	-11,900
Equivalent Stress		18,790	18,621
Contact Side of Element, Inner Surface (psi)			
Bending Stress		-4614	-5176
Six Theta Stress		-1720	-2032
Contact Stress	Hoop	14,680	13,250
	Axial	6900	5910
Equivalent Stress		14,500	13,680
Side Opposite Contact (psi)			
Bending Stress		5345	5996
Six Theta Stress		23	27
Vibration Suppression Load		710	835
Equivalent Stress		6078	6858
Margin of Safety			
Onset of Gross H ₂ Leakage (psi)		32,500	16,500
Margin of Safety		1.24	0.21

Figure 11. Fin Undercut Required to Prevent Secondary Interclad Loading

Figure 12. Load Undercut Relationship

The condition shown is for the 0.032 Incoloy 800 design which requires a 7.6-mil undercut to prevent a second contact. Assuming contact at any distance from the cold end and fins with from 3 to 6 mils undercut, the primary and secondary forces can be determined. For instance, for a secondary contact 16 in. from the cold end and a 4-mil fin undercut, the primary contact load is reduced from 15.7 lb to 6.9 lb while the second contact load is 18.2 lb.

E. NUCLEAR ANALYSIS

1. Burnable Poison Evaluation

As a result of a series of analytic investigations, the prepoison recommended for use in the SPF reactor is a 50-50 mixture (equal worth loadings) of europium enriched in Eu^{151} and natural gadolinium.

A design objective has been that the selected prepoison mixture for the SPF reactor be suitable for use in a higher-powered reactor. Since higher power reactors tend to be larger and have less control capability in the reflector drums, they require larger poison loadings to offset the reduced control capability.

Natural Sm, natural Gd, Gd enriched in Gd^{155} , and Eu enriched in Eu^{151} were considered as poisons because they are the only chemically stable isotopes having burnout characteristics suitable for use in ZrH reactors. When used singly, the burnout characteristics can be summarized as follows:

a. Natural Sm or Natural Gd

These have similar burnout rates. The burnout rate is compatible with reactor operation at 300 kwt due to the small required loading. A very small amount of residual poison would remain at end-of-life (EOL) for these poisons. However, the burnout rates are too high for a high powered reactor.

b. Gadolinium Enriched in Gd^{155}

The Gd^{155} burnout rate closely approximates the reactor reactivity loss rate, resulting in a minimum reactivity mismatch. However, this poison has a relatively high EOL residual reactivity penalty due to unburned poison.

Figure 13. 300-kwt SPF Reactor, Percent Prepoison Remaining vs Time

c. Europium Enriched in Eu^{151}

The Eu^{151} burnout rate is too slow for an optimum reactivity loss rate match. Its burnout rate is made effectively slower by absorption of neutrons by the daughter nuclide, Eu^{152} . This poison has a large EOL reactivity penalty due to unburned Eu^{151} and Eu^{152} .

The burnout characteristics of the important isotopes are shown in Figure 13. Sm^{149} is the only absorbing isotope in natural Sm, while absorption in natural Gd is about 80% due to Gd^{157} and 20% to Gd^{155} .

Of the four poisons, all but Eu^{151} have a pronounced positive effect on the prompt fuel temperature coefficient. Eu^{151} has a negative effect. Since the higher powered ZrH reactors require a heavy poison loading, any one of the four poisons, considered alone, will result in an unacceptable perturbation of the coefficient. The solution to this is the use of Eu^{151} in combination with one of the other three poisons in order to achieve a relatively unperturbed coefficient. It should be noted, however, that this requirement does not strictly apply to the 300-kwt SPF reactor, where a smaller poison loading is required, and a single poison could be used without unacceptable changes in the coefficient.

2. Burnable Poison Mixture Selection

A series of analytical studies were undertaken to determine which of the other three poisons could be used with Eu^{151} and in what ratio. The selection of the poison material to be used in combination with Eu^{151} was based primarily on considerations of mismatch and EOL residual poison.

Due to the exponential burnout of the poison materials and the more nearly linear shape of the reactivity losses that occur over life, a perfect match of reactivity losses and gains is not possible. The reactivity mismatch results in a change in net reactivity over life for which the reactor control system must be capable of compensating. The mismatch in reactivity for various combinations of $\text{Gd}^{155}/\text{Eu}^{151}$ and $\text{Sm}^{149}/\text{Eu}^{151}$ are shown in Figures 14 and 15, respectively. Also shown in these figures are typical desired ranges of mismatch and the resulting initial loading for the 300-kwt system.

For any mixture of prepoisons, there is a significant penalty in EOL reactivity due to unburned prepoison. The residual (EOL) prepoison works as a

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Figure 14. 300-kwt Reactor Mismatch with Gd^{155}/Eu^{151} Prepoison

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Figure 15. 300-kwt Reactor Mismatch with Sm/Eu^{151} Prepoison

function of initial loading are shown in Figures 16 and 17 for the same poison mixtures given in Figures 14 and 15.

For poison loadings below \$8, the combination of Sm^{149} and Eu^{151} results in a lower residual than Gd^{155} and Eu^{151} . This can be seen more clearly in Figure 18 in which 50-50 mixtures of poisons are plotted together.

The substitution of natural Gd for Gd^{155} in the SPF reactor offers significant advantages. When compared on a nuclear basis, Sm^{149} and natural Gd behave similarly. This is shown in Figures 19 and 20. Since Gd appears to be more stable from a material compatibility standpoint, it is currently preferred over Sm.

The preferred ratio of natural Gd to Eu^{151} is that mixture which gives a prompt temperature coefficient closest to the unperturbed value. An initial loading of 50% natural Gd/50% Eu^{151} satisfies this objective.

Figure 16. 300-kwt Reactor Residual Prepoison with Gd¹⁵⁵/Eu¹⁵¹ Prepoison

Figure 17. 300-kwt Reactor Residual Prepoison with Sm/Eu¹⁵¹ Prepoison

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Figure 18. 300-kwt Reactor Mismatch and Residual Prepoison,
50-50 Mixtures of Sm/Eu¹⁵¹ and Gd¹⁵⁵/Eu¹⁵¹

Figure 19. 300-kwt Reactor Mismatch and Residual Prepoison, Natural Sm, Natural Gd, and Gd^{155}

Figure 20. 300-kwt Reactor Mismatch and Residual Prepoison, 50% Mixtures of Eu^{151} with Sm, Gd, or Gd^{155}

III. REACTOR DEVELOPMENT

A. INTRODUCTION AND SUMMARY

The current SPF reactor development efforts comprise the fuel element and reactor core heat transfer and hydraulic studies, and the design and testing of components for the control drum drive system. The components include the modified S8DR actuator, bearings, and limit switch and the improved design of the drum drive and launch lock mechanism. A metallic sheathed "hard wire" electrical cable and terminations are being developed to improve the operational reliability of the electrical components. The previously used process to anodize the beryllium reflector surfaces to enhance thermal emittance and inhibit oxidation will be used. The complete drum drive system will be tested to verify the design under both simulated space and SPF environments.

The second series of the fuel element heat transfer tests with the smooth wall heater assemblies spaced at a pitch-to-diameter ratio (P/D) of 1.05 were completed. While good agreement with the expected values of film coefficients was achieved, essentially no circumferential temperature gradient of the element cladding could be noted. The latter is attributed to increased fluid turbulence created by the ring spacers used to obtain the P/D of 1.05. Some inter-channel fluid mixing was indicated by the test data.

These smooth-walled heater assemblies were converted to finned elements by brazing ribbon fins. The test assembly with R-L-N finned elements ($P/D = 1.05$) is being assembled in preparation for the next series of heat transfer tests. The effect of fins in promoting interchannel coolant mixing should be observable from these tests.

Further hydraulic studies were conducted to improve the methods of determining the change of the fluid conductivity along the path of fluid flow, which is the basis of calculating the interchannel mixing. The results of these studies are being incorporated into the next mixing test assembly now being completed. The simulated fuel elements of this test assembly are more representative of the SPF design both in diameter and configuration of the R-L-N fins.

An S8DR design actuator was hard mounted on the vibration shaker and subjected to loads expected during a launch on a Saturn V booster. No damage or

effects on performance were detected at these loads and loads several times higher. The specification and design modifications of the S8DR actuator for the SPF application were initiated.

The review of the ZrH drum drive and launch lock mechanism, previously tested, showed that the friction between the driving cam and driver pin is very critical. Design refinements to decrease the dependence on friction and to improve the lubrication provided by the dry-film lubricant are in progress. Tests to evaluate the friction and durability of the dry-film lubricant under simulated operational reactor startup condition have been designed and test samples are being fabricated.

A potential source for duplicating the anodizing process for the beryllium reflector, successfully demonstrated on other SNAP 8 reactors, has been located.

Studies to optimize the design of the metallic sheathed electrical cables, their terminations, and the routing on the reactor system is in progress.

B. FUEL ELEMENT HEAT TRANSFER STUDIES

Studies are in progress to generate experimental data pertaining to the heat transfer characteristics of NaK flowing through channels formed by closely spaced fuel elements, a configuration representative of the ZrH reactor core. These data will provide the basis for more reliable prediction of the fuel and cladding temperatures. Initial tests with closely packed arrays and with elements spaced by rings to provide a $P/D = 1.05$ have been completed and a Test Report has been prepared and released.

For the $P/D = 1.00$ test configuration, the experimental results were found to be in excellent agreement with theoretical predictions. In the flow range investigated, the rod azimuthal temperature gradients and Nusselt numbers were shown to be independent of the Peclet number, in contrast to the test results of other investigators. In comparing theoretical and experimental results for the $P/D = 1.05$ rod arrangement, good agreement was obtained for the rod average film coefficients; however, poor agreement was found for the cladding circumferential temperature gradients. The test results for $P/D = 1.05$ showed essentially no azimuthal temperature gradients. This was attributed to a

significant increase in the fluid turbulence intensity caused by the presence of spacer rings, which were used to maintain a symmetrically spaced rod bundle.

The next series of tests will have a simulated fuel bundle made up of 12 finned elements and seven smooth elements. Six of the 12 finned elements have right-hand-wound fins and six are wound left hand. This arrangement provides for fin-on-tube spacing and, with a fin width of 0.030 in. and an element base diameter of 0.625 in., provides for a $P/D = 1.05$.

Modification of the elements and the installation of the test section into the test loop were completed. The connection of the electrical and instrumentation leads is in progress. A view of the test section with the R-L-N heater elements is shown in Figure 21.

A development effort on the fabrication methods of future heater assemblies for heat transfer tests is being undertaken. Prototype heaters of the size representative of the SPF fuel element are being built with both thick and thin sheath wall. The thick-wall heater will subsequently be hobbled to form the fins. A ribbon of stainless steel will be wrapped and brazed to the thin-wall heater to form the fins. The optimum fabrication method evolving from this study will be utilized to fabricate the heater for future SPF fuel element and core heat transfer studies.

C. FUEL ELEMENT AND CORE HYDRAULICS

1. Fuel Element Hydraulic Tests

The fuel elements of the SPF reactor core are designed with R-L-N fins to provide positive placement of the element and promote the interchannel coolant mixing. The amount of interchannel mixing influences the thermal gradient of the fuel elements. Hydraulic studies are being conducted to evaluate the effects of the various fin configurations on the interchannel mixing and the pressure drop.

Test data on the earlier mixing tests were partially compromised by the capacitive interaction between the electrical cables which run between the test section and the instrumentation console. A series of mixing tests were conducted on the previously described test assembly to determine the magnitude of these effects. The leads to the modulation detector circuit used to excite and

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Figure 21. Assembly of Finned Element Heat Transfer Test

extract the signal from the conductivity probes were separated. In addition the probe excitation frequency was lowered from 200 to 50 kHz. Both of these modifications were made to minimize any reactive coupling between probe leads.

Salt was injected into Channels 10 and 4 of Figure 22. Salt flow weight percent measurements were then made along the axial length of the selected channel gaps by the conductivity probes. The variations of the salt concentration at the various probe locations on an instrumented channel group were calculated from the test data. Results of these tests indicated that the isolation of leads and lowering of the excitation frequency virtually eliminated any interaction of signals in the probe leads. The improved measurements indicated the same general results as previous measurements insofar as axial locations of the occurrence of maximum salt concentration in a channel are concerned.

From the results of these and prior tests it is concluded that the flow in a bundle tri-cusp channel has an axial component along the surface of an unfinned rod and a component moving in a spiral upward along the finned rods. Thus the data measured depend on whether the probes are located on finned or unfinned elements. Further, if the probes are located on a finned element, results depend on whether the fins are sweeping salty or fresh water from an adjacent channel. This is apparently due to the lack of complete mixing of the "fresh" and "salt" water within the flow channel. Better mixing is achieved within a channel with a single spiral inlet compared to a channel with a double inlet.

Preparations are continuing for the assembly and testing of a 19-element bundle with elements closely representing the SPF design.

2. Inlet Plenum Evaluation

To provide some experimental basis for the conceptual design of the core inlet plenum, two-dimensional hydraulic tests are being conducted. The effects of the inlet nozzles, flow baffles, and dividers on the hydraulic characteristics within the inlet plenum are being evaluated.

The open channel model consists of a full-scale simulation of the inlet plenum in an inverted attitude. The four inlet lines are of a half-round cross section with extended walls. The grid plate, which is adjustable in the vertical

position relative to the inlet lines, is orificed to the anticipated flow requirements of the three-zone core. In addition, the grid plate is instrumented with 199 pressure taps located at lattice positions coincident with the virtual axes of the fuel rods. The pressure measurements at the surface of the grid plate provide the principal basis for the evaluation of inlet plenum flow structure. The model is installed in a distilled water loop which provides individually variable constant head inlet flow rates up to 30 gpm per inlet. The inlet flow rates are monitored by calibrated total head impact tubes. In addition to the grid plate pressure measurements obtained from individual water manometers, flow visualization techniques employing flow streamers are used to visually and photographically record the plenum flow patterns.

The initial series of tests was conducted with the model simulating the case with four constant area inlets, resulting in abrupt velocity changes. In order to achieve the desired uniform pressure at the grid plate and uniform flow data in the core, several diffuser and anti-swirl baffle designs have been evaluated. An inlet geometry consisting of 4, four-vane, equal-area diffusers without the anti-swirl baffle provided the closest approach to the ideal flat grid plate pressure profile. When the grid plate was positioned below the inlet lines, the same inlet geometry coupled with an additional perforated plate, to support the diffusers and provide additional flow straightening, gave the best results.

The model is being modified to evaluate the benefit of several flared inlet nozzle configurations.

D. ACTUATOR AND BEARING ENGINEERING

To minimize the development efforts for the SPF control drum drive system, the actuator and the control drum bearings are to be of the S8DR design with only minimal modification required for integration. The extensive development and design verification test history of the S8DR actuator and bearing will be applicable to the SPF system.

1. Actuator

As the S8DR actuator had not previously been subjected to vibration loads expected during reactor launch on Saturn V,⁽²⁾ one phase of the SPF reactor actuator review includes performing the vibration evaluation. An actuator

Figure 23. S8DS Actuator - Random Vibration Test, Qualification Level (2.4 g rms), X Axis

(S8DS Dev. S/N 010) has successfully completed random vibration, resonance frequency searches, and a complete post-test mechanical and electrical inspection.

Random vibration was conducted in the axial (X) and transverse (Y and Z) axes for four energy levels. Tests were run at system acceptance levels (1.6 g rms total input energy), at qualification levels (2.4 g rms), at twice acceptance levels (3.2 g rms), and at three times acceptance levels (4.8 g rms). The later two tests were conducted to study design margin of safety. Figure 23 shows the axial qualification test data with the vibration input and the output at the actuator shaft and brake housing. These data are typical of other runs. Output data showed only moderate resonance and the actuator proved capable of meeting launch loads with a large margin of safety.

During vibration in the Y axis at three times acceptance level, small amounts of black powder were observed coming from a vent hole in the actuator body. The testing was suspended and the actuator was disassembled for inspection of the MoS₂ dry-film-lubricant-coated brake disks and the carbon-graphite shaft bushings. Small additional amounts of powder were found inside the actuator. They were analyzed by x-ray diffraction techniques and determined to be carbon-graphite with small amounts of iron. Examination of the carbon-graphite bushing showed some wear to the journal surface (Figure 24). The bushing was in generally good condition and the wear present did not hinder nominal operation of the actuator. Random vibrations prior to inspection had accumulated a total of 45 min, twelve times actual launch conditions, which would be significant when considering fretting of the carbon-graphite bushing. Some oxidation of this bushing had probably occurred during the previous 12,000-hr design verification test of this actuator in a 10^{-5} torr and 1125°F environment.

The actuator was reassembled and a series of resonance frequency searches were conducted to determine the frequencies and their amplification factors. Sweeps were made between 8 to 2000 Hz at 1/4 g (at a rate of 1 oct/min) in both the axial (X) and transverse (Z) axes. Sweeps in the X axis were also conducted at 1/2- and 1-g levels. Table 7 shows the resonance frequencies and their amplification factors measured at 1/4 g in the X axis. These are typical

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Figure 24. Actuator Thrust Bushing After Vibration Testing

TABLE 7
RESONANCE FREQUENCY SEARCH, 1/4 g IN X AXIS

Input Frequency (Hz)	Input (g)	Response (g)			
		Shaft	Endbell		
		X Axis	X Axis	Y Axis	Z Axis
15	0.25 ↓	-	-	0.75	-
58		0.44	0.38	-	0.30
85		-	0.86	-	-
120		-	0.43	-	-
150		0.62	0.36	-	-
170		0.48	-	-	-
250		1.40	0.36	-	-
320		1.2	0.72	-	1.75
400		-	-	0.60	0.50
580		1.0	-	-	0.60
700		1.9	2.0	2.6	1.7
760		1.5	-	1.9	2.0
800		-	2.3	0.9	4.0
1680		-	1.2	0.9	0.4
1900		-	2.0	3.2	2.5

of other runs and like the random vibration tests show no large amplification or areas of concern.

The actuator was reinspected following the resonant frequency searches and no additional wear was evident to the carbon-graphite bushing. No other damage was present and the post-test operation of the actuator was completely within design requirements.

The S8DS actuator specification and drawings are currently being reviewed to incorporate minor changes for the SPF reactor application.

2. Bearings

S8DS bearing drawings and the bearing specification are being reviewed to include minor changes for the SPF reactor system. In the conceptual design of

the SPF Control Drum Drive System, the S8DR control drum bearing is slightly modified to simplify fabrication and installation. The changes are limited to nonbearing surfaces. Unlike the S8DR design both bearing shafts on the SPF will have a common diameter, a further simplification.

E. HARD WIRING

Because of the importance of the control drum actuator and, to a lesser extent, the drum limit switches to the reactor control and scram functions, studies are being conducted on methods of hardening the cables feeding the actuator and limit switches. In the past, cables have been made up of glass-fiber-insulated conductors covered with alternate layers of glass and stainless-steel braid. This type of cable is susceptible to damage during assembly of the cable to the components and during subsequent system assembly and checkout operations.

The cable hardening techniques being investigated are those related to the use of metal sheathed cables swaged from assemblies of metal tubing ceramic insulation and conductors. Aspects of the application of swaged cable which are being investigated include: (1) optimum conductors per cable, (2) optimum conductor and sheath materials, (3) methods of sealing high and low temperature cable ends, and (4) methods of attaching cable to components and of terminating the conductors to the component leads. Preliminary designs of termination methods have been made and cable samples are being procured.

IV. FUEL DEVELOPMENT

A. INTRODUCTION AND SUMMARY

The major goals of the fuel development effort are to characterize the candidate cladding materials, establish the detail process parameters for fuel element fabrication, and define the influence of the overall reactor operating environment on ZrH reactor fuel element performance.

The primary effort in fuel element processing during the quarter was on evaluation of hobbing techniques and on establishing weld and final closure parameters for both Hastelloy X and Incoloy 800. A second group of 20-mil, smooth-wall Hastelloy X elements was assembled. Effort on cladding property evaluations was concentrated on the Hastelloy X test program. Mechanical and thermal testing of fuel elements continued. The barrier creep tests on Incoloy 800 fuel elements were completed and tests on Hastelloy X elements were begun.

B. FUEL ELEMENT MATERIALS EVALUATION

1. Fuel Element Assembly and Test

Four 25-mil-wall, finned Incoloy 800 fuel elements have accrued over 6000 hours of endurance testing at 1400°F. Each element has received ten ramp thermal cycles at intervals during the course of isothermal endurance testing. The current average leak rate and hydrogen loss are 0.25 cc(STP)/hr and 2.3%, respectively. The average normalized leak rate and cumulative hydrogen loss are shown as a function of endurance time in Figure 25.

The eleven 15-mil-wall, Incoloy 800 fuel elements were removed from the test program for post-mortem analysis after accumulating in excess of 20,000 hours at 1400°F with periodic interrupts for ramp thermal cycles. Final leak rate and hydrogen loss data for these elements were reported previously.⁽³⁾

Processing of Incoloy 800 cladding components for fabrication of additional fuel elements has been delayed due to difficulties in producing acceptable coated end cap-to-coated tube welds in the 32-mil-wall configuration. Acceptable metal weld quality has been demonstrated. However, when full weld penetration of the cladding is achieved in welding coated components, spalling of the glass barrier in the weld area occurs. A series of parametric studies are in progress.

a. Average Normalized Leak Rate

b. Cumulative Average Hydrogen Loss

Figure 25. Permeation History of Four 25-Mil Incoloy-Clad Elements

Twelve additional Hastelloy X fuel elements were assembled and permeation tested during the reporting period. These elements contained segmented fuel, a nominal H/Zr of 1.81 and were clad in 20-mil-wall, unfinned tubes. The status and planned test program for the fuel elements are summarized in Table 8, which includes two elements reported previously. Of the fourteen Hastelloy X elements fabricated to date, thirteen are considered acceptable after initial permeation testing.

TABLE 8
HASTELLOY X ELEMENTS - PERMEATION RATES

Element Number	Permeation Rates, As-Measured [cc(STP)/hr]					End Use
	Initial Test		1300° F Tests			
	1300° F	1400° F	Post Ramp	Post Scan	Endurance *	
X-3	0.27	0.96	0.23			Uniaxial
X-6	0.20	0.63	0.22			Uniaxial
X-9	0.11	0.48				Load Test
X-11	0.13	0.62	0.18		0.24	1300° F End
X-12	0.24	0.63				Load Test
X-15	0.14	0.46		0.18		Scan, Ramp, End
X-16	0.22	0.95				Load Test
X-14	1.55	4.19				Reject
X-7	0.13	0.36	0.16		0.25	1300° F End
X-8	0.17	0.84	0.22		0.23	1300° F End
X-10	0.18	0.66	0.18		0.08	1050° F End
X-13	0.17	0.60	0.20			1050° F End
X-17	0.20	0.61				Scan, Ramp, End
X-19	0.25	0.72				Scan, Ramp, End

*1000-hr Data

It had been previously reported that coated butt weld parameters for Hastelloy X were tentatively established based on a yield of >90% on the initial group of tubes processed after completion of parametric studies. A second

group of tubes was processed with a 90% yield, tending to confirm the validity of the developed parameters. However, additional work will be required to establish parameters for the reference Hastelloy X cladding thickness of 27 mils.

Some degradation of mechanical properties (primarily ductility) has been observed in SP (simulated process) cladding specimens as opposed to STT (simulated thermal treatment) specimens. The SP specimens are exposed to the standard processing environment, i. e., static air during the coating firing operation. The STT specimens are subjected to standard processing thermal treatments, but are encapsulated in a vacuum or inert atmosphere. Metallographic examination of Incoloy 800 samples indicated evidence of nitride formation on the outside diameter after the air coating firing cycle. The same phenomena, although not observed metallographically, is postulated for Hastelloy X. As a result, the coating firing process is being modified to provide an inert gas flow over the cladding OD during the firing cycle, similar to that already employed during the oxidation process. Preliminary results are encouraging although further process refinements are required.

An experiment was performed to determine whether the fuel element hydrogen permeation rate would increase if subjected to lateral loading under a reactor temperature gradient. Both Incoloy 800 and Hastelloy X fuel elements were deflected 0.050 in. under a single point load at about $Z/L = 0.6$ in a simulated reactor profile of 1125°F inlet and 1200°F outlet temperature. There was no increase in the hydrogen permeation rate after 250 hours at temperature. This indicates that the fuel elements should be able to withstand the maximum anticipated side loading during reactor operations without any adverse effects.

2. Burnable Poison

Evaluation of the compatibility test specimens removed from test after 3000 hours at 1400 or 1500°F is nearly complete. These specimens and specimens from earlier 1500-hour tests were processed as a group through x-ray diffraction, x-ray fluorescence, and electron microscopy. All had similar visual appearance upon decanning. The results of the analyses and evaluations to date are summarized below.

- 1) Gd_2O_3 , Y_2O_3 , and Gd_2O_3/Y_2O_3 vapor-deposited coatings are highly adherent and stable on the fuel rod surface. Only trace quantities

were found on the adjacent barrier with no differentiation between times or temperature. The presence of these vapor-deposited films inhibits the fuel barrier reaction normally observed in U-ZrH fuel elements. No fuel-barrier reaction products, ZrO_2 and UO_2 , were detected on the barrier although some transfer of silicon from the barrier to the fuel was found.

- 2) Gd_2O_3 and/or Y_2O_3 , whether applied as solid solution or as a sandwich layering, did not stabilize Eu_2O_3 on the fuel rod surface. A large fraction of the Eu transferred to the SCB barrier. Correlation of all data indicates that the majority of this transfer occurred during the period of the fuel element permeation tests conducted at 1200 or 1300° F. This is the reason little if any change in visual appearance was observed between the 1500-hour and 3000-hour tests. Once transfer is effected, the Eu appears to reside stably on the barrier and the Gd_2O_3 and/or Y_2O_3 remain stable on the fuel rod. The Gd_2O_3 and/or Y_2O_3 on the fuel still appear to inhibit the fuel-barrier reaction.
- 3) Transfer of Sm from the fuel rod to the SCB barrier was also observed. It was analogous to the Eu_2O_3 transfer although apparently at a slower rate.

Because of these results, additional emphasis is being placed on the alternate methods for incorporating Eu_2O_3 in the fuel element, e. g., encapsulation (in coated form) in thin metal foils.

Compatibility testing of fuel elements is being continued to obtain long-term data on the stability of Gd_2O_3 and Y_2O_3 vapor coatings on fuel. Status of these tests is given in Table 9.

C. CLADDING PROPERTIES

Evaluation of Hastelloy X Heat E9504 (0.29% Co) continued with tensile tests in both the simulated thermal treatment (STT) and simulated process (SP) conditions. Results are listed in Table 10.

TABLE 9
COMPATIBILITY TEST STATUS*

Test Number	Fuel Specimen Length (in.)	Number of Interfaces	Accumulated Time 10/20/71 (hr)	Target Time (hr)	Test Temperature (° F)
D	4	2	9730	-	1400
F	1	17	6911	10,000	1300
G	1	17	6911	10,000 ⁺	1300
J	1	17	6909	10,000	1400
L	1	17	5761	10,000	1300
O	2	3	6909	10,000	1400

*Specimens are coated with Gd₂O₃, Y₂O₃, and Gd/Eu in ratios of 6/3, 4/6, and 2/3 with and without Y₂O₃ in solution with the Eu₂O₃. Uncoated fuel rod segments are contained in each test element as controls.

TABLE 10
HASTELLOY X TENSILE DATA, HEAT E9504

	Test Temperature (° F)	Aging at 1200° F (hr)	Yield Strength (0.2% Offset) (ksi)	Ultimate Strength (ksi)	Elongation (% in 2-in. gage)
Simulated Thermal Treatment (STT) Condition	RT	-	53	114	36
	RT	300	57	119	29
	RT	1000	64	125	24
	1100	--	37	90	39
	1200	--	36	84	46
	1200	300	37	84	53
	1200	1000	44	87	48
Simulated Process (SP) Condition	RT	--	54	110	19
	1100	--	40	91	28
	1200	--	37	84	29
	1200	300	38	84	35
	1200	1000	41	86	32
Union Carbide (0.109-in. Sheet)	RT	--	52	114	43
	1200	--	40	83	37

NOTE: Each STT and SP data point represents the average of three tests.

The yield and ultimate strength values obtained at room temperature and 1200°F from unaged specimens, STT or SP condition, are very close to the vendor data for sheet specimens. Aging tends to increase the yield and ultimate strength slightly, with no significant difference observed between STT and SP condition. The STT specimens did show significantly higher ductility than SP specimens in both unaged and aged tests. The effect of aging at 1200°F cannot be determined, for either STT or SP specimens, because the data appear random. Additional specimens are being aged at 1200°F for 5000 and 10,000 hours which should clarify any long-term aging effects.

Six uniaxial creep specimens were tested at 1100, 1200, and 1300°F. The specimens all had high temperature extensometers attached directly to the gage length to monitor strain. The results are given in Figure 26. The creep rates were obtained from creep curves after ~1000 to 2000 hours when the rates were linear. The curves parallel literature data but are offset at about 2000-psi lower stress.

The mechanical properties of five heats of Incoloy 800 have been studied to determine the effects of alloy chemistry and cladding processing. As a result of this study, a recommended heat chemistry and melting procedure to achieve the optimum combination of strength and ductility for processed Incoloy 800 fuel element cladding have been generated and are shown in Table 11.

Figure 26. Hastelloy Uniaxial Creep, Heat E9504 - STT Condition

TABLE 11
RECOMMENDED INCOLOY 800 ALLOY CHEMISTRY

Element	Desired Heat Chemistry (wt %)
Ni	30.0 to 35.5
Cr	19.0 to 23.0
Fe	Balance
C	0.025 to 0.005
Ti	0.40 to 0.05
Al	0.20 to 0.05
Si	1.00 maximum
Cu	0.75 maximum
Mn	1.50 maximum
Mo	0.15 maximum
Co	0.15 maximum
B	0.0005 maximum
Melting Procedure	Air melting followed by vacuum remelting

Figure 27. Electrical Power Output vs Elapsed Time for Refractory-Clad Reactor Power Modules

V. THERMOELECTRIC CONVERTER DEVELOPMENT

A. REACTOR POWER MODULES

1. Refractory-Clad Modules

During the past quarter, primary emphasis was placed on the evaluation of performance data from TEM-X series refractory-clad modules, and on the identification of end closure design modifications required to incorporate glass-to-metal seals at the exposed end closure pin insulators.

Four TEM-X series modules were tested during this reporting period; two in the Atomics International liquid metal loop system, and two in electrically heated static test stands at Westinghouse Astronuclear Laboratories. Electrical power output data from these identical TEM-X S/N-3 series modules tested are presented as functions of testing time in Figure 27. The data presented in this figure exhibits the high degree of reproducibility of performance, the degree of insensitivity to cyclic operation, and the high degree of performance stability achieved in these modules. A comparison of pre- and post-cyclic test data from TEM-X S/N-3C and -3D in Figure 27 indicates the lack of sensitivity of all performance parameters to cyclic operation.

A statistical analysis (linear curve fit) of all life test data from TEM-X S/N-3C is presented in Table 12 along with data from the other TEM-X modules which have been tested to date. The latter four modules in the list differ from the first two in that they incorporate hermetically sealed end caps and the conductor ring dimensions were modified to tighten the inner conductor ring components, reducing tellurium transport during processing. Power output degradation rates in the latter four modules range from +2.4 (increase) to -1.6 (decrease) % per 10,000 hours, attesting to the extremely stable operational characteristics of the TEM-X series modules operated at peak temperatures of 1185° F.

During this reporting period the liquid metal loop system in which TEM-X S/N-3D was being tested, was shut down to allow installation of an additional module, TEM-X S/N-3F. After completion of the installation work, both modules were heated to $\bar{T}_H = 1085^\circ\text{F}$ and $\bar{T}_C = 507^\circ\text{F}$ operating conditions. Beginning-of-life test data from TEM-X S/N-3F are compared to similar data from all

TABLE 12
REFRACTORY-CLAD REACTOR MODULE PERFORMANCE STABILITY SUMMARY

Module	Testing Time (Hours)	Rate of Change: % Per 10,000 Hours				Thermal Cycles		
		R _g	V _{oc}	P _e	η	Uncontrolled Number	Controlled	
							No.	Rate (°F/Min)
TEM-X S/N-3A*	1166S	4.0	1.1	-1.9	-2.0	1	49	30
TEM-X S/N-3B*	2584S	2.8	0.1	-2.5	3.0	2	0	--
TEM-X S/N-3C	4310S	1.5	1.2	+0.9	2.1	1	53	30
TEM-X S/N-3D	1831L	4.9	1.7	-1.4	9.6	1	53	30
TEM-X S/N-3E	2512S	2.0	1.9	+1.8	0.8	0	0	--
TEM-X S/N-3F	623L	2.0	0.3	-1.5	5.2	0	1	--

*Testing Terminated
L Indicates Loop Test
S Indicates Static Test

All modules tested with $\bar{T}_H = 1085^\circ\text{F}$, $\bar{T}_C = 507^\circ\text{F}$
All data included through October 22, 1971.

TEM-X S/N-3 series modules as well as calculated values in Table 13. All electrical parameters agree very well with calculated values as well as data from other modules. In particular, data from the last four modules fall in an extremely tight band.

TEM-X S/N-3D and -3F are being tested for 1000 hours at the initial S/N-3D operating conditions ($\bar{T}_H = 1085^\circ\text{F}$, $\bar{T}_C = 507^\circ\text{F}$ with 200°F axial NaK temperature drops) to allow a comparison of performance data from the two loop-tested modules. After completion of the first 1000 hours, the NaK axial temperature drops will be reduced to 100°F and TEM-X S/N-3F operating conditions will be adjusted to $\bar{T}_H = 1140^\circ\text{F}$ and $\bar{T}_C = 523^\circ\text{F}$. The high temperature operation of TEM-X S/N-3F will allow a determination of the sensitivity of performance stability to module operating temperatures. The 100°F loop axial temperature drops are more consistent with projected operating conditions for reactor-thermoelectric systems.

Three additional TEM-X series refractory-clad modules were identified during the past quarter. These modules, designated TEM-X S/N-3G, 3H, and 3J, will include end closure modifications to permit the application of glass-to-metal seals at each power pin base. The modifications consist of moving the power pin bolt circle outward radially 0.042 in. to provide additional clearance to apply the glass-to-metal seal. In addition, the boron nitride will be recessed 0.050 in. (axially) to capture the glass during application.

2. Prototype 5.25-Volt Reactor Power Modules

TEM-13G S/N-2, a first generation 5.25-volt reactor power module fabricated with an Inconel inner cladding, remained on test in a two-loop liquid metal system at Atomics International during the past quarter. This module has accumulated over 17,400 hours of operation at $\bar{T}_H = 1025^\circ\text{F}$ and $\bar{T}_C = 538^\circ\text{F}$ operation conditions. Previously reported performance trends continued for this module during the past quarter. Power output and overall efficiency degradation rates of 18 and 16%, respectively, have been determined for this module which has been thermally cycled six times. These data are in close agreement with similar data from TEM-13G S/N-4 which was previously tested at the same temperatures in a static test stand at WANL.

TABLE 13

A COMPARISON OF CALCULATED AND EXPERIMENTAL BOL PERFORMANCE
PARAMETERS FROM TEM-X S/N-3 SERIES MODULES

Module	Room Temperature		B.O.L. $\bar{T}_H = 1085^\circ\text{F}$, $\bar{T}_C = 507^\circ\text{F}$					
	R_g (Milliohms)		R_g Circuit (Milliohms)	R_g Module (Milliohms)	V_{oc} (Volts)	P_e (Watts)	Q_{in} (Watts)	η_o (Pct)
	Circuit	Module						
TEM-X S/N-3	2.7	*	**	**	**	**	**	**
TEM-X S/N-3A	2.7	3.0	23.7	24.9	5.24	276	5847	4.73
TEM-X S/N-3B	2.7	*	*	25.0	5.29	280	5933	4.72
TEM-X S/N-3C	2.7	*	*	25.1	5.44	294	5995	4.90
TEM-X S/N-3D	2.7	3.2	*	25.0	5.52	304	5974	5.10
TEM-X S/N-3E	2.6	2.9	*	25.0	5.46	297	5978	4.98
TEM-X S/N-3F	2.6	2.9	*	25.0	5.54	306	5861	5.23
CALCULATED	2.7	3.0	23.8	24.9	5.41	291	6272	4.64

* Pin Resistance Not Measured

** Not Tested

It is significant to note that the rate of degradation increased after each thermal cycle. In the first 3255 hours of operation in the liquid metal loop system, during which no thermal cycles were experienced, power output and overall efficiency degradation rates of 10 and 8%, respectively, were observed. The increased rate of degradation is caused by accelerated tellurium flow through mica cracks and tears produced by component distortions during transient operation. More recent reactor power modules, viz. the TEM-X series incorporating refractory inner claddings, have exhibited greatly improved performance stability in comparison to these TEM-13G data by virtue of the reduced mica deformation. In addition, test data from TEM-X S/N-3A and S/N-3C, which were thermally cycled 50 and 52 times, respectively, showed that the improved performance stability is not altered by cyclic operation.

B. PUMP TUBULAR POWER MODULES

Performance Summary of Prototype Pump Modules

One tubular module designed to deliver power to an electromagnetic pump continued on test at WANL throughout the past quarter. This module, designated as TEM-14A S/N-1, is being endurance tested at nominal average cladding temperatures of $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$. A second module, TEM-14A S/N-2, was shut down as a result of an electrical heater failure. A new heater is currently on order and will be installed to permit continued testing of the module.

Electrical current and electrical power output data from all TEM-14A series modules are presented in Figures 28 and 29, respectively. Data from TEM-14A S/N-3, which was removed from test for destructive examination after completing 50 thermal cycles with no measurable degradation, are included for comparison. Performance levels achieved in S/N-2 and S/N-3, designed with the number of nickel electrical pins increased to 12 from 6 to minimize electrical resistance, are in excellent agreement with the design levels (110 watts at 500 amp).

Table 14 presents a summary of performance data from the electromagnetic pump modules. Degradation rates for four parameters were determined from a statistical analysis of experimental data from these modules. An electrical heater failure in TEM-14A S/N-1 after 17,847 hours resulted in a shutdown

Figure 28. Electrical Current Data from Three Prototype Pump Power Modules

Figure 29. Power Output Data from Three Prototype Pump Power Modules

TABLE 14
PUMP MODULE PERFORMANCE STABILITY SUMMARY

Module	Testing Time (Hours)	Rate of Change: % Per 10,000 Hours				Thermal Cycles	
		R_g	V_{oc}	P_e	η_o	Uncontrolled Number	Controlled No. Rate (°F/Min)
TEM-14A S/N-1	17847	*	*	-4.8	-6.2	9	0 -
	5926***	1.7	0.3	-1.2	-1.2	1	2 30
TEM-14A S/N-2	18784	6.5	2.0	-2.7	-2.7	8	1 30
TEM-14A S/N-3**	7727	4.5	3.0	1.3	-0.4	2	50 30

* No open circuit data taken for S/N-1 during first 17847 hrs of testing. All modules tested at $\bar{T}_H = 1125^\circ\text{F}$, $\bar{T}_C = 570^\circ\text{F}$

** Testing terminated.

All data included through October 15, 1971

*** S/N -1 data taken after reinstrumentation.

TABLE 15
PROCESS AND DESIGN FEATURES OF TEM-15 SERIES MODULES
(Sheet 1 of 2)

Module	Process and Design Innovation
15A S/N-1	Same as 15A-4 except 300 lb axial preload and EB welded.
15A S/N-4	Baseline design: 36 couples, .050 inch T/E washers, .005 inch mica, .001 inch mica radial clearance, 1100 lb axial preload, TIG welded, hydrogen purged, tungsten insert under p-type washer with axial clearance at assembly.
15A S/N-4A	Same as 15A-4. (Destructively examined after processing.)
15A S/N-4B	Same as 15A-4. (Thermal cycled at 2.5°F/min ramp rate.)
15A S/N-4C	Same as 15A-4. (Thermal cycled at 30°F/min ramp rate.)
15A S/N-4D	Same as 15A-4. (Steady state endurance tested.)
15A S/N-4E	Same as 15A-4 except not hydrogen purged.
15A S/N-4F	Same as 15A-4 except not hydrogen purged and tungsten insert clearance eliminated.
15A S/N-4G	Same as 15A-4F.
15A S/N-5	Same as 15A-4 except 300 lb axial preload, EB welded, and .010 inch mica radial clearance incorporated.
15A S/N-8	Same as 15A-4 except .010 inch mica radial clearance incorporated.
15B S/N-4	Same as 15A-4 except each intercouple insulator consists of .0004 inch W foil (full radius) sandwiched between two .0025 inch mica washers.
15B S/N-4A	Same as 15A-4 except each intercouple insulator consists of .0004 inch W foil (1/2 radius) sandwiched between two .005 inch mica washers (33 couples).
15B S/N-4B	Same as 15B-4 except W foil insulators prevented from extending beyond mica at O. D. and tungsten insert clearance eliminated.
15C S/N-4	Same as 15A-4 except each intercouple insulator consists of 2000 - 4000Å Mo barriers deposited between two .0025 inch mica washers.
15C S/N-4A	Same as 15A-4F except each intercouple insulator consists of 500 - 1500Å tungsten barriers EB deposited between two .0025 inch mica washers.
15C S/N-8	Same as 15C-4 except .010 inch mica radial clearance incorporated.

TABLE 15
PROCESS AND DESIGN FEATURES OF TEM-15 SERIES MODULES
(Sheet 2 of 2)

Module	Process and Design Innovation
15D S/N-4	Same as 15A-4 except .010 inch mica insulators incorporated. (33 couples)
15F S/N-4	Same as 15A-4F except ANX-6 type PbTe-GeTe substituted for Ge n-type material.
15H S/N-4	Same as 15A-4F except W inserts used under n-type washers at the inner conductor rings.
15N S/N-4	Same as 15A-4F except Ge-nl material used in both T/E legs. Mo inserts and glass-to-metal access pin seals used.
15P S/N-4	Same as 15H-4 except TEGS-2P material used in both T/E legs and glass-to-metal access pin seals used.

and subsequent re-instrumentation of the module. Prior to the shutdown, the module was being tested with a permanently fixed load (0.46 milliohms). Because the load was bolted to the module leads, it was impossible to make meaningful instantaneous open circuit measurements, hence there was no open circuit voltage or internal resistance data recorded.

After the re-instrumentation, however, the TEM-14A S/N-1 test assembly was equipped with the load resistance network previously used in the TEM-14A S/N-3 test, thus allowing open circuit measurements. This modification makes it possible to perform temperature adjustment calculations. Data from TEM-14A S/N-1, then are handled as two separate line segments; the first covering the initial 17,847 hours and the second covering all subsequent data. As can be seen in Table 14, the degradation rate data measured after the re-instrumentation and load modification are somewhat lower than the initial data.

The long-term stable performance capabilities of the tubular module incorporating lead telluride thermoelectric materials are clearly demonstrated by the performance data from the TEM-14 series pump modules. Successful performance following the fifty controlled thermal cycles on S/N-3 (see Figures 28 and 29) and the various uncontrolled shutdowns of all three modules indicate that the TEM-14 design is insensitive to thermal cycling.

C. SUPPORTING TECHNOLOGY DEVELOPMENT

1. Sublength Module

The TEM-15 series modules were designed as a part of the support technology development phase of the Compact Thermoelectric Converter System Technology Program. These modules were designed with radial dimensions identical to those of reactor power modules, but with circuit lengths reduced to 4.0 in. from 15.0 in. The purpose for fabricating and testing these modules is to evaluate the effects of design, processing, and fabrication modifications on module performance. All TEM-15 series modules tested to date have been operated at nominal $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$ cladding temperatures in static test facilities at WANL.

A list of all the process and design features of each TEM-15 series module fabricated to date is presented in Table 15. Eleven TEM-15 series modules were tested continuously and two new modules were placed on test during the past quarter.

TABLE 16
TEM-15 PERFORMANCE ANALYSIS SUMMARY

Module	Testing Time (Hours)	Rate of Change: % Per 10,000 hrs				Thermal Cycles		
		R _g	V _{oc}	P _e	η_o	Uncontrolled No.	Controlled	
							No.	Rate (°F/Min)
TEM-15A S/N-1	15,235	32	2.1	-19	-16	6	1	30
TEM-15A S/N-4D	15,510	31	2.5	-17	-15	5	1	30
TEM-15A S/N-4E	3,090	29	7.1	-14	-10	1	0	--
	7,315**	50	2.4	-27	-24	2	1	30
TEM-15A S/N-4F	4,647	28	3.4	-18	-13	0	1	30
TEM-15A S/N-5	14,764	29	2.6	-17	-13	8	51	30
TEM-15B S/N-4A	9,740	14	3.3	-6	-5	2	1	30
TEM-15B S/N-4B	4,648	18	2.7	-12	-10	0	1	30
TEM-15B S/N-4C	1,074	21	2.4	-16	-6	0	0	--
TEM-15C S/N-4A	1,405	31	9.5	-12	-3	0	0	--
TEM-15C S/N-8	12,928	25	2.4	-15	-12	5	1	30
TEM-15D S/N-4	10,976	18	2.4	-11	-8	4	1	30
TEM-15F S/N-4	2,777	48	1.4	-40	-38	0	0	--
TEM-15H S/N-4	5,056	23	3.0	-16	-15	0	1	30

*Insufficient testing time to allow meaningful performance analysis.

**After 190 hours of $\bar{T}_H = 1200^\circ\text{F}$ operation.

All data included through October 15, 1971
All modules tested with $\bar{T}_H = 1125^\circ\text{F}$, $\bar{T}_C = 570^\circ\text{F}$

Table 16 lists all TEM-15 series modules tested during this reporting period and summarizes the results of a degradation analysis of all modules having in excess of 1000 hours testing time.

a. TEM-15A Series Modules

TEM-15A series modules were designed to investigate the influence of minor component dimensional variations and the effects of the hydrogen purge operation on module performance stability, the influence of weld shrinkage on circuit alignment and mica deformation, and the influence of axial pre-load on mica deformation and circuit alignment in a module incorporating standard 0.005-in.-thick mica insulators. No diffusion barriers have been incorporated in any of these modules.

Endurance testing of five TEM-15A series modules continued throughout the past quarter. Electrical power output data from these modules are presented in Figure 30, as functions of elapsed time. As can be seen in the figure, excellent long-term performance reproducibility has been achieved in all TEM-15A series modules, indicating that none of the design, processing, or fabrication modifications has had a gross effect on module performance.

TEM-15A S/N-5 completed a series of fifty 30°F/min thermal cycles during the past reporting period. The purpose of these cycles was to determine the long-term effects of thermal cycling, if any, on a 4-in.-long module incorporating an Inconel inner cladding. Although insufficient test data have been generated to date to allow a meaningful post-cyclic performance stability (degradation) analysis, preliminary analysis indicates that the cycles have had no measurable effect on performance. Steady-state testing of TEM-15A S/N-5 is being continued at $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$ operating conditions.

An additional TEM-15A series module, designated S/N-4G, was fabricated during this reporting period. This module, which is identical to TEM-15A S/N-4F, will be tested at $\bar{T}_H = 1025^\circ\text{F}$ and $\bar{T}_C = 480^\circ\text{F}$ operating temperatures as opposed to 1125/570°F conditions maintained for all previous TEM-15 series modules. The reduced hot cladding temperature will allow a determination of the temperature dependence of the degradation rate for this design. $\bar{T}_C = 480^\circ\text{F}$ was specified to duplicate the axial cladding expansion mismatch existing at the standard TEM-15 cladding temperatures (0.017 in.). Testing of TEM-15A S/N-4G will be initiated in early November.

Figure 30. A Comparison of Power Output Data from TEM-15A Series Modules

Degradation rate data from all TEM-15A series modules, listed in Table 16, remained essentially unchanged during the past reporting period, i. e., no major changes in previously established performance stability trends were observed. These data indicate that degradation continues to follow a linear relationship with time for extended testing periods and that performance stability is relatively insensitive to the processing and dimensional variables under investigation in the TEM-15A series of modules.

b. TEM-15B Series Modules

The TEM-15B designation applies to all sublength reactor technology modules incorporating tungsten foil diffusion barriers positioned between two mica electrical insulators. Two TEM-15B series modules, S/N-4A and S/N-4B, continued on test throughout this reporting period. A third module, TEM-15 S/N-4C was placed on test in late September.

Electrical power output from the tungsten foil barrier modules is presented in Figure 31. The individual design features of each of the modules are listed in Table 15. Although the power output and efficiency levels of the TEM-15B series modules are somewhat lower than those of the nonbarrier TEM-15A series modules as a result of the tungsten foil shunt heat losses, greatly improved performance stability has been achieved as can be seen in Table 16.

c. Additional Modules

In addition to the tungsten foil barrier and nonbarrier modules, five one-of-a-kind modules were tested during the past quarter. These are the last five modules listed in Table 16. TEM-15C S/N-8, TEM-15D S/N-4, and TEM-15H S/N-4, whose individual design features are listed in Table 15, all have accrued substantial testing times and all continued to operate without change during the past quarter. These modules indicate that various degrees of performance stability improvements can be achieved through the use of EBVD molybdenum barriers, tungsten N-washer contacts, and increased mica washer thicknesses. Power output data from these modules are presented in Figure 32.

TEM-15C S/N-4A, a diffusion barrier module incorporating 500 to 1500Å of electron beam vapor-deposited tungsten on each mica insulator, was placed on test during the past month. Because of the thickness of the tungsten barrier,

Figure 31. A Comparison of Power Output Data from TEM-15B, Tungsten Foil Barrier Modules

Figure 32. A Comparison of Power Output Data from Improved Performance Stability
 TEM-15 Series Modules

Figure 33. Power Output Data from TEM-15F S/N-4

shunt heat losses are negligible and performance predictions for TEM-15C S/N-4A are the same as those for the TEM-15A series modules.

TEM-15F S/N-4, the first module to incorporate the recently developed lead telluride base ternary alloy Pb-Te-GeTe, N-type thermoelectric material (designated ANX-6), was tested continuously throughout the past quarter. This module is identical to TEM-15A S/N-4F except for the N-type material incorporated. Power output data from this module, shown in Figure 33, indicate a rate of degradation higher than that of other TEM-15 modules (see Table 16). The increased degradation is thought to be a result of contamination of the ternary material. Identification of the contamination mode will require a detailed destructive examination of the module.

2. Glass Seal Development

a. Borate Glass

Over the past reporting period, various seals were made on single-pin mockups using preforms of the borate glass. Fourteen different specimens were made. Four were formed by furnace heating and ten by use of a hand torch. None of the seals made in the furnace were helium leak tight (HLT). Of the ten torch-formed seals, eight were cycled at a rate of $\sim 17^{\circ}\text{C}/\text{min}$ between ~ 530 and 210°C . Only two seals which had seven and nine cycles were helium leak tight after cycling. The remainder leaked after as few as two cycles. Pressure checking the non-helium leak tight seals with argon showed no detectable leaks to argon using a liquid leak detector. The leaks in the others showed up near the vertical glass-to-stainless-steel interface. The glass was deemed not suitable for the application and further work using this glass was cancelled.

b. Leaded Glass

Another glass was selected for experimentation, again starting with single-pin mockups. The glass, a leaded type, had a lower thermal coefficient of expansion than the borate glass, but possessed a longer working temperature range. Seven single-pin seals were made during these preliminary tests. The results, shown in Table 17, were sufficiently encouraging to proceed to four-pin mockups.

TABLE 17
RESULTS OF SINGLE PIN SEALS

Seal	Cycles	HLT	Disposition
NM-L7	50	Yes	On hold
NM-L8	15	Yes	On hold
NM-L9	0	No	Examined metallographically
NM-S9	36	Yes	On hold
NM-S11	40	Yes	On hold
NM-L10	5	Yes	Put on life test
NM-L12	0	Yes	Put on life test

All seals made with glass paste and hand torch.

NM-L10 and NM-L12 were placed on life test at 500° C. The former was instrumented for a resistance measurement between the pin and plate, and the latter was set up for helium leak checking at temperature. NM-L12 was helium leak checked every 24 hours, at which time the procedure produced a thermal shock, i. e., the cooler helium abruptly cooled the seal by about 40 to 50C°. The test resulted in a helium leak at 768 hours in this seal. NM-L10 has maintained a resistance in excess of 2×10^6 ohms for 1340 hours at 500° C. At 989 hours the seal was cooled and found to be helium leak tight.

Four-pin mockups were fabricated using preformed glass beads as well as glass paste, and radial frequency (RF) heating experiments also supplemented the hand torch method. The experimental transition from one-pin seals to simultaneous sealing of four seals in a single plate was accomplished with more non-successful seals, at first, than were encountered during the single-pin program. Further process development, using RF heating and glass beads then produced seals which exhibited cycling capability and are helium leak tight. Table 18 shows the results obtained on both four- and eight-pin mockups that were cycled as described above.

Module end mockups have been prepared for sealing experiments. These autoclaved mockups incorporate a redesigned pin bolt circle and a two-piece BN sleeve which, after processing, results in a recessed annulus for the glass which is to be applied.

TABLE 18
SUMMARY OF FOUR- AND EIGHT-PIN SEALS

Seal	HLT Cycles	Non-HLT Cycles	Disposition
4P-1S	4	--	On hold
4P-2S	45	50	Storage
4P-3L	5	--	On hold
4P-8L	6	24	Storage
4P-9L	6	24	Storage
4P-10L	18	30	Storage
4P-13L	36	42	Storage
4P-14L			Life test 500°C
4P-15L			Life test 500°C
8P-3L	41	53	Storage
8P-4L			Life test 500°C
8P-5L			Life test 500°C
8P-6L	54	--	On hold
8P-7L	50	--	On hold
8P-8L	30	--	On test

BLANK

VI. REACTOR-THERMOELECTRIC SYSTEMS ENGINEERING

During this reporting period, the Systems Engineering effort was devoted to the design and analysis of SNAP reactor systems for two Department of Defense (DOD) missions. One study consisted of reviewing the various types of nuclear power systems, selecting the preferred system for the stated mission objectives and constraints, and performing a conceptual design on the selected system. The second study involved the design of a reactor-thermoelectric system using available SNAP 10A hardware.

For the first mission, the complete spectrum of nuclear space power systems was evaluated. Both reactor and radioisotope heat sources were considered as well as Brayton, organic Rankine, and thermoelectric power conversion systems. All combinations of heat sources and power conversion systems were characterized over a power range from 1 to 10 kwe. The characterization included system weight, size, cost (development, unit, and launch), and other qualitative considerations. The various combinations were then compared at four discrete power levels (ranging from 1.5 to 4.5 kwe) defined by the spacecraft requirements. The results of the study showed that the reactor-thermoelectric system was the lowest cost power system. Other qualitative considerations such as availability, safety, and development risk also favored the reactor-thermoelectric system.

For the second DOD mission, a system design was prepared using as much as possible of the available SNAP 10A FS-3 hardware, including the reactor. TEM 13-G modules were used for the power conversion system. A detailed shielding analysis was performed, since low payload dose levels were required. An optimum split in shield placement between the reactor and payload was defined in the study.

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