9/6/72
89 app
Distributed in
C-92 b.

Dr. 2274 UC-80

AI-AEC-13032

SNAP REACTOR PROGRAMS

PROGRESS REPORT

FEBRUARY-APRIL 1972

AEC Research and Development Report

Atomics International
North American Rockwell

P.O. Box 309
Canoga Park, California 91304

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy \$3.00; Microfiche \$0.95

SNAP REACTOR PROGRAMS
PROGRESS REPORT
FEBRUARY-APRIL 1972

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Atoms International
North American Rockwell

P.O. Box 309
Canoga Park, California 91304

CONTRACT: AT(04-3)-701
ISSUED: AUGUST 15, 1972

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

py

DISTRIBUTION

This report has been distributed according to the category "Systems for Nuclear Auxiliary Power (SNAP) Reactor — SNAP Program," as given in the Standard Distribution for Classified Scientific and Technical Reports, M-3679.

CONTENTS

	Page
I. Introduction.	9
A. Program Scope and Objectives	9
B. Summary of Progress	9
II. Zirconium Hydride Reactor Engineering	11
A. Introduction and Summary	11
B. Fuel Element Design.	13
1. Cladding Selection	13
2. Fuel and Fuel Cladding Interaction Test.	16
3. Fuel Segment Bow	16
C. Thermal-Hydraulic Analysis	17
1. Coolant Mixing Model	17
2. Coolant Pressure Drop Correlation.	19
3. Core Thermal Characteristics	21
4. Reflector Component Temperature	24
D. Nuclear Analysis	25
E. Mechanical Design	25
1. Reactor Assembly	25
2. Reflector and Drive System	29
3. Control System Reliability.	29
III. Reactor Development	33
A. Introduction and Summary	33
B. Core Heat Transfer Studies	35
1. Finned Element Tests	35
2. Prototype Heater Development	35
C. Core Hydraulic Studies (Mixing Tests).	35
D. Reflector Drive System.	41
1. Evaluation and Recommendation of Components	41
2. Testing.	49
E. Actuator.	51
1. Design Analysis	51
2. Tests	51

CONTENTS

	Page
F. Electrical Components	53
1. Hard Cable	53
2. Limit Switch	53
G. NaK EM Pump	53
1. Development Pump	53
2. System Pump	55
H. NaK Volume Accumulator	55
IV. Fuel Development	57
A. Introduction and Summary	57
B. Fuel Element Materials Evaluation	57
1. Fuel Element Assembly and Test	57
2. Burnable Poison	63
C. Cladding Properties	65
V. Thermoelectric Converter Development	75
A. Reactor Power Module	75
1. Introduction	75
2. TEM-X S/N-3 Series Modules	75
3. High Voltage Refractory-Clad Module (TEM-X S/N-4)	85
4. Module Parametric Analysis and Preliminary Design	85
B. Pump Tubular Power Modules	87
1. Performance Summary of Prototype Pump Modules	87
2. Pump Module Parametric Analysis	87
C. Supporting Technology Development	91
1. Sublength Modules	91
2. Glass Seal Development	105
References	108

TABLES

	Page
1. As-Fabricated Stresses	14
2. As-Assembled Stresses	14
3. Short-Term Operational Stresses	15
4. Core Thermal Performance Summary (Symmetric Core).	23
5. Reflector Assembly Temperature	24
6. Dimensions of Mixing Test Bundle Configurations	39
7. High Temperature Ball Screw Experience	44
8. Control Drum Gear History, MoS ₂ -Coated Gears.	45
9. MoS ₂ Friction Characteristics	50
10. Effects of Processing Atmosphere on Tensile Properties	58
11. Hobbed Tube Wall Thickness Variation Batch to Batch	60
12. Hastelloy X Element Permeation Data	61
13. Vibration/Permeation Test Data	61
14. Hastelloy X Tensile Data, Heat E9504	66
15. NAA 120-4 Hastelloy X Cladding Chemistry	69
16. NAA 120-4 Hastelloy X Creep Data, Heat C-6505.	69
17. Refractory-Clad Reactor Module Performance Stability Summary	80
18. Pump Module Performance Stability Summary.	89
19. 0.2-volt, 700-amp Pump Module Parametric Results, $\bar{T}_H = 1168^\circ\text{F}$, $\bar{T}_C = 522^\circ\text{F}$	90
20. Process and Design Features of TEM-15 Series Modules	92
21. TEM-15 Performance Analysis Summary	94

FIGURES

	Page
1. Reactor Assembly — 5-kwe Thermoelectric System	12
2. Fuel and Cladding Interaction	16
3. Salt Concentration Hydraulic Experiment	18
4. Hobbed Fin Hydraulic Test Friction Factor Correlation	20
5. Random Asymmetry Case 1	22
6. Local-to-Average Power, Segments Closed	26
7. Local-to-Average Power, Segments 2 in. Open, Homogenized Beryllium	27
8. Peak Power per Element Relative to Core Average	28
9. Five-Year Reliability for Control Drive System.	30
10. Heater Cladding Thermocouple Calibration in Water with 1.7-kw Heater Power	34
11. Test Cladding Temperature Profile, R-L-N Finned Element P/D = 1.05	34
12. Corrected Cladding Temperature Profile, R-L-N Finned Element P/D = 1.05	36
13. Maximum ΔT (Cladding-Coolant) Variation	36
14. Typical Arrangement of Test Bundle Elements (As Viewed from Outlet End)	37
15. Tracer Salt Distribution, Mixing Model with 0.012-in. Fins	38
16. Tracer Salt Distribution, Mixing Model with 0.026-in. Fins	40
17. Pressure Drop Through Mixing Model	42
18. Sliding Reflector Control Translating Nut Ball Screw Drive	43
19. Alumina vs Carbon-Graphite Friction (Static)	46
20. Sector Anti-Rotation Method (Nut Reaction)	47
21. Room Temperature Spectral Emittance of Anodized Beryllium	48
22. Dual-Throat NaK EM Pump Assembly	52
23. EM Pump and Thermoelectric Converter	54
24. Load Relaxation Test Rig.	62
25. Hastelloy X Load Relaxation Test	64
26. Creep Rate Comparison of High and Low Cobalt Hastelloy X	68
27. Uniaxial Creep of Low Cobalt Hastelloy X (E9504), Aged and Unaged	70
28. Uniaxial Creep of Irradiated Hastelloy X, Heat C6505, NAA 120-4	71

FIGURES

	Page
29. Uniaxial Creep Comparison of Irradiated vs Aged Hastelloy X at 1200° F	72
30. Biaxial Stress Rupture of Hastelloy X (C9591)	73
31. Overall Efficiency vs Elapsed Time for Refractory-Clad Reactor Power Modules.	76
32. Power Output vs Elapsed Time for Refractory-Clad Reactor Power Modules	77
33. Open Circuit Voltage vs Elapsed Time for Refractory-Clad Reactor Power Modules.	78
34. Internal Resistance vs Elapsed Time for Refractory-Clad Reactor Power Modules.	79
35. A Comparison of Calculated and Experimental Effective Seebeck Coefficient Data Recorded During Open Circuit Heatup of TEM-X Modules.	82
36. A Comparison of Calculated and Experimental Internal Resistance Data Recorded During Open Circuit Heatup of TEM-X Modules. . . .	82
37. A Comparison of Calculated and Experimental Thermal Impedance Data Recorded During Open Circuit Heatup of TEM-X Modules. . . .	82
38. Electrical Current Data from Three Prototype Pump Power Modules	86
39. Power Output Data from Three Prototype Pump Power Modules . . .	88
40. A Comparison of Power Output Data from TEM-15A Series Modules	96
41. A Comparison of Overall Efficiency Data from TEM-15A Series Modules	97
42. A Comparison of Power Output Data from TEM-15B, Tungsten Foil Barrier Modules	98
43. A Comparison of Efficiency Data from TEM-15B, Tungsten Foil Barrier Modules	100
44. TEM-15B S/N-4D Couple Design.	102
45. A Comparison of Power Output Data from Improved Performance Stability TEM-15 Series Modules	103
46. A Comparison of Overall Efficiency Data from Improved Performance Stability TEM-15 Series Modules	104
47. Helium Leak Tight Hours vs 1/T.	106

BLANK

I. INTRODUCTION

A. PROGRAM SCOPE AND OBJECTIVES

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

- 1) Zirconium Hydride Reactor Engineering
- 2) Reactor Development
- 3) Fuel Development
- 4) Thermoelectric Converter Development.

A new activity, 5-kwe System Engineering, was added to the SNAP Program during this report period. The task is devoted to system engineering of a 5-kwe, reactor-thermoelectric system. The system is to be designed to meet both flight and ground test requirements and to be ground-tested for up to 5 years. The design is presently in the early conceptual design phase and, consequently, no significant milestones have been realized as yet. Therefore, this task is not a part of this report.

The Zirconium Hydride (ZrH) Reactor Engineering effort was primarily directed toward conceptual design of the reactor for the 5-kwe system.

The Reactor Development tasks were devoted to (1) reactor hydraulic studies and (2) component evaluation and development for the 5-kwe system reflector control subsystem and for the liquid metal heat transfer subsystem.

The Fuel Development tasks establish process parameters for fuel element fabrication and evaluate fuel element performance.

The Thermoelectric Converter Development activities are being carried out by the Astronuclear Laboratory of Westinghouse Electric Corporation, under sub-contract to Atomics International. The activities are directed toward development of compact, thermoelectric power converters for utilization in the 5-kwe system.

B. SUMMARY OF PROGRESS

Although Incoloy 800 is easier to machine, more ductile, and has better long-term thermal stability, Hastelloy X was selected, because of its superior strength,

as the reference fuel element cladding material for the current generation of ZrH reactors. Test work to determine the stress distribution resulting from fuel bearing on the cladding wall was completed. A mathematical model was prepared so that these results could be applied to fuel elements of different diameters.

Coolant mixing tests for the right-left-neutral fuel element array were completed and the data evaluated.

Conceptual design of the reactor for the 5-kwe system was initiated. Although the reactor design point is a function of the system design point selected, its thermal power requirement is approximately 100 kw at a reactor outlet temperature of 1200°F for 5 years.

A ball screw, with a translating nut, was selected as the reference technique for the reflector segment drive. Test fixtures have been designed to test the ball screw mechanism to temperatures of 1000°F and 10^{-6} torr vacuum at design loads.

Satisfactory SCB-coated butt welds were obtained on Hastelloy X fuel element cladding. In addition, helical fins were machined in 19 Hastelloy X cladding tubes by hobbing. Vibration and permeation tests indicate that fuel elements are unaffected by loads resulting from a Saturn V or Titan III-C launch. Hastelloy X cladding property testing continued.

Reactor power modules continue to operate with degradation rates ranging from 1.0 to 3.5% per 10,000 hours for operating times to 3500 hours, temperatures to 1140°F (average), and up to 58 thermal cycles per module.

II. ZIRCONIUM HYDRIDE REACTOR ENGINEERING

A. INTRODUCTION AND SUMMARY

The engineering effort during the past quarter was devoted to cladding material selection, fuel and cladding interaction tests, fuel bow analysis, and hydraulic analysis; as well as the initiation of the design of a reactor for application in a 5-kwe unmanned system.

A significant milestone was the selection of Hastelloy X as the reference cladding material for the current generation of ZrH reactors. Development tests showed that the SCB permeation barrier adheres well to Hastelloy X. Hastelloy X also exhibits a strength advantage over the alternate cladding, Incoloy 800. The higher strength results in increased pre-operational and operational stress margins. Those properties in which Incoloy 800 showed some superiority — machinability, ductility, and long-term thermal stability — were considered insufficient to outweigh the advantages of Hastelloy X.

Other fuel element engineering activities included the evaluation of test data on the magnitude and distribution of stresses in the cladding wall resulting from the bearing force of an internal fuel slug. A mathematical model was formulated to enable application of the test results to other tube diameters and wall thicknesses.

The engineering evaluation of coolant mixing test data for the right-left-neutral finned element array was completed. A computer solution of the simultaneous differential equations used to model the test section provided a good match of the experimental data. A correlation of mixing rate as a function of geometry was generated for use in core design and analysis. The mixing test also provided pressure drop data which were used to update the correlation used in current core design.

The nuclear analysis activity during the quarter included two-dimensional transport theory calculations (DOT code) to determine power distribution in cores having a tapered reflector. The "open window" effect, due to a partially inserted reflector segment was also determined.

Figure 1. Reactor Assembly — 5-kwe Thermoelectric System

Design and analysis needed to establish the conceptual reactor design for a 5-kwe thermoelectric system application was initiated. The reactor power, temperature, and lifetime requirements are approximately 100 kw and 1200° F outlet temperature for a period of 5 years. Thermal, nuclear, structural, weight, and configuration studies were performed and tradeoffs evaluated. The initial conceptual configuration is shown in Figure 1. Control is achieved through the use of two sliding segments. This configuration represents a highly compact configuration and is optimum from the overall system weight standpoint since it minimizes the volume required within the shadow shield envelope.

B. FUEL ELEMENT DESIGN

1. Cladding Selection

A study was undertaken to compare the two candidate cladding materials for the SPF reactor fuel elements. Interchangeable elements of Incoloy 800 and Hastelloy X were compared for both the pre-operational and operational environment.

The static stress levels in the cladding are evaluated at three conditions: as-fabricated, as-assembled, and operational. The as-fabricated stresses are induced in the cladding from the firing cycle of the SCB coating. The SCB coating has a lower coefficient of expansion than the candidate cladding material and, in cooling from the firing temperature to room temperature, tensile stress is induced into the cladding. In addition to these stresses, there are assembly stresses caused by placing the as-fabricated element, which, per print, has a maximum TIR of 0.012 in., into the core cluster of elements. This causes bending and local contact stresses in the "as-assembled" condition. It is important that this stress be below the elastic limit of the material in order to avoid thermal ratcheting upon reactor shutdown.

Comparisons of the two cladding materials are shown in Tables 1 and 2. The cladding wall thicknesses for the two designs were chosen to provide equal core reactivity. As can be seen, the higher strength Hastelloy X cladding provides substantially greater margins of safety.

TABLE 1
AS-FABRICATED STRESSES

	Hastelloy X	Incoloy 800
Geometry (in.)		
OD	0.735	0.735
Thickness	0.027	0.032
SCB Thickness	0.0025	0.0025
Total Induced Stress (psi)		
Hoop	7,715	8,105
Axial	6,990	7,340
Equivalent	7,380	7,750
Margin of Safety	3.82	1.06

TABLE 2
AS-ASSEMBLED STRESSES

	Hastelloy X	Incoloy 800
Bending Stresses (psi)	-2,225	-1,990
Contact Stress (psi)		
Hoop	61,110	4,430
Axial	2,880	1,970
As-Fabricated (psi)		
Hoop	7,715	8,105
Axial	6,990	7,340
Equivalent Stress (psi)	12,000	10,000
Margin of Safety	1.96	0.47

During reactor operation the short-term operational stresses are those which occur during reactor operation without regard to the time effects. The failure mode for this condition is the glass (SCB) barrier tensile strain. The stresses considered include the pre-operational stresses plus the induced, six-theta, bending and load contact stresses from the thermal environment. The superiority of Hastelloy X is again evident in Table 3 which shows a margin of safety against gross hydrogen permeation of 1.74 for the Hastelloy X element vs 0.22 for the Incoloy element.

TABLE 3
SHORT-TERM OPERATIONAL STRESSES

	Hastelloy X	Incoloy 800
Bending (psi)	-4,356	-4,877
Six-Theta (psi)	-1,706	-2,006
Local Contact (psi)		
Hoop	11,900	11,950
Axial	4,710	4,970
Equivalent Stress (psi)	12,394	12,676
Allowable Stress for Onset of H ₂ Permeation (psi)	34,000	15,500
Margin of Safety	1.74	0.22

The long-term environment includes the effects of creep and for the multi-startup condition, fatigue. The maximum creep strain predicted is 0.0238% for the Hastelloy X element and 0.035% for the Incoloy 800 element. Both of these are well below the capability of the materials and the associated damage is nil. The same is true for fatigue.

In summary, Hastelloy X is superior over Incoloy 800 because of its higher strength and was chosen as the reference cladding material. The pre-operational margins are higher in Hastelloy because of the increased elastic limit and the lower coefficient of expansion. During operation, the higher strength of Hastelloy X means less strain induced into the SCB barrier for overstress conditions or a higher margin of safety. Because of the nature of this application, the superior short term ductility, and somewhat superior irradiated creep ductility of Incoloy 800 could not be utilized to advantage.

2. Fuel and Cladding Interaction Test

The test to establish the stress distribution of fuel bearing on the cladding wall was completed. The test was run utilizing 0.015-in. gage length strain rosettes measuring the induced strain due to a simulated fuel slug side load (Figure 2). Stresses on the outer surface were derived from the measured strain data. Data analysis consisted of determining stress components under the load and at 90° from the load.

6531-51109

Figure 2. Fuel and Cladding Interaction

The components of stress under the load include: (1) Those circumferential bending stresses due to the cladding conforming to the fuel, (2) the tensile circumferential stresses developed after the cladding has conformed, and (3) the beam-on-elastic-foundation mathematical model of meridional stresses. Using these models and extrapolating from the 0.540-in. ID by 0.020-in. -wall test specimen, a 1.0-in. OD by 0.025-in. -wall-thickness element has an outer surface stress varying from 11,500-psi tension with 5-lb load to 21,500-psi tension with 50-lb load. On the inner surface these stresses become 9,800 and 14,000 compression, respectively.

At 90° from the applied load, the maximum stress are approximately 1310 psi/lb tension on the inner surface and compression on the outer surface.

After the vibration suppression device maximum design load is determined, the side load due to an offset fuel slug can be determined. Then the actual stress levels can be established.

3. Fuel Segment Bow

In a segmented fuel element, the length per segment is selected to limit the maximum bow exhibited by any segment to a small fraction of the total available diametral gap. The number of segments per element should, however, be minimized since it relates directly to the fabrication cost of the core. Segment bowing

calculations were performed on the "worst-case" segment for the operating conditions of the 5-kwe reactor application. Four-segment and five-segment elements were analyzed. Each exhibited a maximum bow at end of life as shown below.

<u>Number of Segments</u>	<u>Segment Length (in.)</u>	<u>Maximum Bow (mils)</u>	<u>Time for Maximum Bow (hr)</u>
4	4.0	1.36	44,000
5	3.2	0.94	44,000

Further studies are being performed to determine the effects of these results on the segment length selection.

C. THERMAL-HYDRAULIC ANALYSIS

1. Coolant Mixing Model

An advanced model has been developed to better predict the channel-to-channel mixing using the 19-element, salt-injection mixing test data. The model was developed as a result of an extensive study of the local mechanics of the mixing phenomenon for a rotationally additive flow system (R-L-N fins). Previously developed mixing models were mathematically inconsistent with the observed data, in that the models were not capable of correctly calculating the peak values of saline concentration. The advanced diffusion-dilution mixing model was developed to correctly perform this calculation.

In the advanced diffusion-dilution mixing test model, it is assumed that two distinct flows exist within the tricus: (1) a rotational flow which spirals in an annulus about the finned elements; and (2) an axial diffusional flow which has no transverse (mixing) component, and only serves to dilute the rotational flow through diffusional processes. The model defines a distinct boundary between the rotational and diffusional flows, and in actuality this boundary does not exist as such, however, the boundary serves to establish a diffusional surface and a mechanism for calculation of the saline concentration peak values. Thus the mathematics of the model agree with the physics of the flow. As a result of the symmetry which was achieved in the test section, adequate representation was achieved by considering only four channels in the model. Each flow channel

Figure 3. Salt Concentration Hydraulic Experiment

is represented by 60 coolant nodes, hence 240 linear differential equations are solved simultaneously to determine the best data fit. A decaying exponential term was used to represent the buildup of the rotational pattern from a zero rotation entrance condition.

The model was used to predict saline concentrations vs axial distance in the 19-element bundle for the four instrumented tricuspid gaps. Figure 3 illustrates the theoretical vs experimental fit obtained for the 30-mil-high fin bundle. Additional theoretical predictions were made for 15- and 20-mil bundles, which exhibited good agreement between the theory and the data. These calculations were then used to formulate the current mixing correlation which appears below.

$$h_{\text{mix}} = \frac{236 P_{\text{elem}} H_{\text{fin}}}{P_{\text{fin}} \left(0.433 P_{\text{elem}}^2 - 0.393 D_{\text{elem}}^2 - W_{\text{fin}} H_{\text{fin}} \right) \left(1 + 0.91 e^{-45.2 Z H_{\text{fin}}} \right)}$$

where

h_{mix} = mixing parameter (%/in.)

P_{elem} = element-element pitch (in.)

D_{elem} = cladding OD (in.)

H_{fin} = fin average height (in.)

W_{fin} = fin width (horizontal projection, in.)

P_{fin} = fin axial pitch (in.)

Z = axial distance into bundle (in.)

2. Coolant Pressure Drop Correlation

In the early hydraulic tests, conducted with plastic fins on S8DR dummy elements, Reynolds numbers above 5000 were of prime interest. As a result, the test data were maximized in this range and the computed friction factors fit fairly well to a straight line. It was decided to use the Blasius correlation:

$$f = \frac{0.316}{\text{Re}^{0.25}}$$

Figure 4. Hobbed Fin Hydraulic Test Friction Factor Correlation

as a base (since this fit bare rod data well) modified to account for fin geometry and P/D. The work of Sanger in this field was used modified to suit the test data:

$$f = \frac{0.316 YZ}{Re^{0.25}}$$

where

Y = factor to account for P/D (element pitch diameter ratio)

Z = factor to account for fin geometry

$$f = \frac{4.76}{\left(\frac{P_f}{D} \frac{0.5 S}{N_f h} \right)^{0.5}}$$

where

P_f = axial fin pitch

D = element diameter

N_f = number of fins/channel

S = element spacing

h = effective fin height.

Y vs P/D is essentially a logarithmic curve.

Since this early correlation, tests have been run for hobbled elements on 15- and 30-mil spacings using a 0.672-in. element pitch. The average fin heights were 11.5 and 26.5 mils, respectively. Test runs were made at lower Reynolds numbers with the resulting data as shown in Figure 4. It can be seen that the old correlation is good at $Re > 8000$ but degrades rapidly at lower Re . The current correlation fits the data quite well for $Re > 2000$. It then appears that a transition zone exists below this value. Further tests are planned to more clearly define the transition and laminar zone limits.

3. Core Thermal Characteristics

Table 4 summarizes the nominal beginning-of-life (BOL) characteristics of the 5-kwe core and compares them to the SPF reactor. Figure 5 shows a typical

Figure 5. Random Asymmetry Case 1

TABLE 4
CORE THERMAL PERFORMANCE SUMMARY
(Symmetric Core)

	SPF	5 kwe
Nominal Thermal Power (kwt)	300	94.4
Number of Elements	199	85
Inlet Temperature (°F)	1125	1064
Outlet Temperature (°F)	1200	1141
Lattice Pitch (in.)	0.737	1.0255
Element-to-Element Spacing (in.)	0.030/0.015 [†]	0.020
Fin Height (Before Scallop, in.)	0.035/0.020 [†]	0.050
Fin Height (Shaved at Closest Approach)	0.027/0.012 [†]	0.010
Peak Fuel Temperature (Center Element, °F)		
BOL	1353	1223
End of Offset	1333	1212
Radially Averaged Peak Fuel Temperature (Center Element, °F)		
BOL	1331	1209
End of Offset	1311	1200
Heat Transfer Coefficient (Btu/hr-ft ² -°F)	7200/3850 [†]	3300/2030*
Reynolds Number	20680/5890 [†]	7700/3900*
Coolant Velocity (fps)	4.92/3.33 [†]	1.82/1.49*
Mixing Rate (%/in.)	29.3/18.1 [†]	27
Maximum Cross Element ΔT (°F)	12	12
Pressure Drop, Grid-to-Grid (psia)	1.21	0.24

*Denotes nominal and edge channel values.

†Denotes inner and outer zones for SPF.

asymmetric run made using a full 85-element core model and permitting the elements to bow to contact or to the extent of free bow imposed by the thermal gradient. The clearance between elements is 0.010 in. and, in most instances, contact occurs.

It is of interest to note that asymmetry in this reactor is not as powerful in forcing cross element ΔT direction as it has been on previous reactor designs. This is due in large part to the larger diameter of the elements and to some extent on a steeper radial power gradient associated with a one-zone core. It is also due to a lower power density than used heretofore. In general, the cross element ΔT follows the asymmetry, however, in some instances it is diametrically opposed.

Random asymmetry runs have not shown ΔT 's in excess of the 36° shown in Figure 4. It is necessary to impose highly improbable distributions in order to reach higher ΔT 's. The model assumes a 6-mil manufacturing bow always in the same direction as the thermal gradient, thus the solution is somewhat conservative.

4. Reflector Component Temperature

Table 5 presents the results of a preliminary study to determine various reflector system component temperatures for the 5-kwe application. These temperatures are based on the end-of-life reactor power and thermal conditions. The beryllium emissivities used in this study were 0.3 for inner reflector surface and 0.6 for the outer reflector surface.

TABLE 5
REFLECTOR ASSEMBLY TEMPERATURE

	Temperature (° F)
Stationary Reflector	
Above Sliding Segment	775
Beneath NaK Line	875
Sliding Segment	
Beryllium	775
Ball Nut	775
Bearings	775
Tungsten Gamma Shield	1160
Electrically Actuated Band Release Device	735

D. NUCLEAR ANALYSIS

Two-dimensional DOT calculations in R,Z geometry were made to evaluate the effect that sliding reflector segments have on the core power distribution. Two sliding segments, each subtending 120° of the reactor diameter, were considered. The beryllium between the two segments was stationary. The sliding segments have a full travel of 4 in., from 2 in. above the axial centerline to 2 in. below. The reflector is tapered with the movable end being thicker than the stationary end. The reflector and the core are both 16 in. in length. The core contains 85 fuel elements.

Calculations were made with the segments fully closed and 2 in. open. Segments fully closed is representative of the end-of-life position and segments 2 in. open is representative of the beginning-of-life position.

Because azimuthal variation could not be described in R,Z geometry calculations, two approaches were used to calculate the power distribution with the segments 2 in. open. In the first, the fixed beryllium was homogenized with the void in the opening. In the second, the opening was assumed to subtend the full 360° and the power distribution was synthesized from this calculation and the calculation with the segments fully closed. Power contours with segments closed are shown in Figure 6. Contours for segments 2 in. open with homogenized beryllium in the opening are shown in Figure 7. The peak power in each fuel element relative to the core average is shown in Figure 8.

E. MECHANICAL DESIGN

1. Reactor Assembly

The 5-kwe system reactor core is a unitized assembly of the fuel elements, internal reflectors, a core liner, and the inlet and outlet grid plates. The inlet grid plate, a brazed sandwich structure for strength and rigidity with light weight, is the basic support frame for the core.

The reactor vessel is a stainless-steel shell, 10.5 in. ID by 0.040-in. wall over the core section, configured to provide shallow, approximately 2 in. deep, inlet and outlet plenums for minimal reactor height. The coolant inlet pipes connect to the vessel at two balcony-like extensions of the inlet plenum. The outlet nozzles exit radially outward on an axis offset 15° from the inlet pipes. The

Figure 6. Local-to-Average Power, Segments Closed

6531-4602

Figure 7. Local-to-Average Power, Segments 2 in. Open, Homogenized Beryllium

6531-4603

Figure 8. Peak Power per Element Relative to Core Average

reactor is supported by a flanged skirt for bolting to the top of the neutron shield. The skirt flange includes the fixed portion of the reflector mounting hinges and also provides support for any gamma shielding required beyond the reactor vessel diameter.

The neutron shielding is lithium-hydride cast into a stainless-steel casing containing a perforated stainless-steel honeycomb matrix. The four reactor coolant pipes pass straight through the shield in conduits arranged at angles such that they do not align the reactor with the payload. The reactor support loads are carried from the reactor mounting ring through an internal structural cone to the casing sidewall and then down to the shield mounting ring.

2. Reflector and Drive System

Evaluation of control system designs for the 5-kwe thermoelectric application was initiated. The objective of a minimum weight system requires minimum shielded reactor diameter. To this goal, the use of a reflector which is tapered to fit within the shield cone angle is very attractive. Control drums which pivot about a longitudinal axis involve a weight penalty over an axially sliding reflector segment. This results from the need to shield the drums when in the "out" position. The mechanical design evaluations have therefore concentrated on a design concept using tapered, axially sliding segments for the control function.

Each half-reflector is mounted on pivots attached to the reactor vessel support skirt. During operation, the reflector halves are held in contact with the vessel by means of the EABRD/spring assembly. At end-of-life, the EABRD can be electrically "fired" to provide a positive backup shutdown method.

The control segments are driven by a screw and nut assembly. The nut is affixed to the movable segment. Backlash of the drive train is eliminated by the use of a compression spring between the fixed and sliding segments. Rotation of the segment about the drive screw axis is prevented by engagement of a rolling element in a guide track.

3. Control System Reliability

As a contribution toward defining the reliability goals for the control system, a parameter study was made of reliability as a function of drive unit mean failure rate. The study was limited to a flight system where the control segments

6531-5504

Figure 9. Five-Year Reliability for Control Drive System

are only required to start up and move through their travel once during the 5-year reactor life. Figure 9 shows the 5-year reliability for two and for four unit control systems with mean failure rates from 0.01 per million hours to 10 per million hours.

The reactivity characteristics of the 5-kwe reactor were assumed for these calculations. The initial shutdown margin was \$0.50, the startup reactivity is \$2.43 and the long-term loss rate (includes burnable poison) was \$0.31/yr. The curves shown assume that all control segments function until the reactor is at operating conditions (100% startup probability). For the two-segment, \$0.50 end-of-life margin case the reactor cannot reach operating conditions if one of the two segments fails to start. The control segments were assumed to move in together during the reactor life unless one failed. A failed segment was assumed to remain in its failed position.

BLANK

III. REACTOR DEVELOPMENT

A. INTRODUCTION AND SUMMARY

The current reactor development efforts comprise the fuel element and reactor core hydraulic studies to support the reactor design and evaluation of components for the sliding reflector reactor control system and the liquid metal heat transfer system. These components include the linear drive mechanism, drive actuator, support bearings, position switch, end of life (EOL) shutdown device, metallic sheathed electrical cables, NaK pump, NaK volume accumulators, and space radiator.

Based on the conceptual design study of the sliding reflector control system, the following components are being evaluated.

- 1) Reconfigured S8DR actuator
- 2) Ball screw assembly and spur gear reduction to convert the rotary actuator output to linear reflector movement
- 3) S8DR-type bearings to support the linear drive mechanism and guide the sliding reflector
- 4) S8DR-type position switch to signal reflector position
- 5) Metallic sheathed electrical cables
- 6) SNAP-10A-type electrically actuated release device to release the reflector at EOL
- 7) Anodizing selected surfaces of the beryllium reflectors to minimize temperature.

The design of test units and samples and facilities to evaluate the above items for this reactor application are in progress.

Fabrication of a NaK EM pump previously designed for another reactor system was resumed. The general configuration and expected performance of this two-throat (primary and secondary loops) unit are similar to the pump required for the 5-kwe system. Evaluation of this pump will provide data to verify the basic design and performance predictions.

6531-5408

Figure 10. Heater Cladding Thermocouple Calibration in Water with 1.7-kw Heater Power

6531-5409

Figure 11. Test Cladding Temperature Profile, R-L-N Finned Element P/D = 1.05

Trade studies on the volume accumulator for the primary and secondary NaK heat transfer loops were initiated. Studies include the type of bellows design and fabrication methods, and means to back load the bellows to prevent voids in the NaK loop.

B. CORE HEAT TRANSFER STUDIES

1. Finned Element Tests

Testing on a simulated 19-element fuel bundle made up of 12 finned elements and seven smooth elements, has been completed. As reported previously,⁽¹⁾ the presence of the fins improved the NaK-to-sheath heat transfer coefficient over the $P/D = 1.00$ case and the ring spacer configuration by factors of approximately 800% and 60%, respectively. Because of data scatter, it was not conclusively determined whether or not a small vestigial circumferential temperature gradient remained.

To separate out thermocouple reading bias, water calibration tests were performed on the elements. The results of these tests are plotted in Figure 10. When this plot is compared with that of the NaK test raw data (Figure 11), it is quite evident that any apparent circumferential temperature gradient follows the same pattern as the thermocouple calibration scatter. The difference between each thermocouple calibration temperature and mean of H-1 data was applied to the raw test data of Figure 11. The best fit straight horizontal line of the corrected test data of Figure 12 would indicate the nonexistence of circumferential temperature variation for the finned elements. This conclusion is further supported by the nondependence of the maximum circumferential temperature difference on element power, as shown on Figure 13, for finned and spaced elements.

2. Prototype Heater Development

Development prototype heaters of a size approximating the fuel element geometry are nearing completion after some swaging difficulties.

C. CORE HYDRAULIC STUDIES (MIXING TESTS)

Flow mixing tests were completed on two different fuel element bundles. These fuel element bundles were of the R-L-N configuration previously described (see Figure 14 and Table 6). The flow test assembly used for these mixing tests

Figure 12. Corrected Cladding Temperature Profile,
R-L-N Finned Element P/D = 1.05

Figure 13. Maximum ΔT (Cladding-Coolant) Variation

- LEGEND**
- (12) = ELEMENT NUMBER
 - = SENSOR ORIENTATION
 - ↻ = FLOW DIRECTION
 - = INJECTED CHANNEL
 - 8A = CHANNEL NUMBER

6531-5412

Figure 14. Typical Arrangement of Test Bundle Elements
(As Viewed from Outlet End)

Figure 15. Tracer Salt Distribution, Mixing Model with 0.012-in. Fins

TABLE 6
DIMENSIONS OF MIXING TEST BUNDLE CONFIGURATIONS

Element-to-Element Pitch (in.)	Element Fin Pitch (in.)	Number of Fins per Element	Element Diameter (in.)	Fin Height (in.)
0.670	5.50	3	0.642	0.026
0.670	5.50	3	0.657	0.012

incorporated the modifications previously described to prevent sensitivity loss by the "in-element" sensors and to improve the agreement of injection shroud tube sensor readings for equal injection rates. Sensitivity loss was prevented by improving the sealing of the instrumented elements. Agreement of readings of sensors in each injection tube was improved by increasing the shroud tube length from injection point to sensor location to allow more uniform flow mixing to occur in each shroud tube.

Flow mixing data were recorded by injecting salt tracer solutions into the flows entering three selected flow channels and then determining the salt distribution in the bundle. This distribution was determined from conductivity measurements made using sensors installed in elements surrounding one of the injected channels (see Figure 14).

These salt distribution measurements were determined as fractions of the salt concentration entering the injected channel (i.e., the salt concentration upstream of the bundle resulting from the injection of salt tracer into flow in the inlet grid plate shroud tube). Measurements were made for Reynolds numbers of 10,000, 15,000, and 20,000 and showed negligible effect of Reynolds number. The effects of reducing the fin height and element gap were quite pronounced, increasing the distance the flow traveled through the bundle before the lateral inter-channel flow was fully developed and also making the bundle salt distribution more symmetrical (see Figures 15 and 16).

These mixing test data were utilized to develop a diffusion-dilution mixing model to predict the coolant interchannel flow on a bulk average basis as a function of flow travel. Based on this model the mixing of 13.7%/in. for the low fins and 24.8%/in. for the high fins are predicted. These values are in percent of the total axial flow through the coolant channels.

Figure 16. Tracer Salt Distribution, Mixing Model with 0.026-in. Fins

A series of pressure drop measurements were made for the 0.012-in. fin height configuration. These measurements were made for Reynolds numbers from 1000 to 20,000. The pressure loss characteristics, Figure 17, were similar to those of the 0.026-in. fin height configuration. The entrance section pressure drop per foot was lower than the drop per foot measured further downstream. The total pressure drop per foot for the 0.012-in. fin height configuration was higher than that measured for the 0.026-in. fins.

D. REFLECTOR DRIVE SYSTEM

1. Evaluation and Recommendation of Components

Following selection of the sliding reflector as the reactor control mode, a conceptual design study of the reflector control and support mechanisms was initiated. Studies included the method of reflector support, the type of drive mechanism, gearing and bearing requirements, and reflector anti-rotation devices. Studies were conducted to determine the preferred location of drive motors and other components on the basis of minimum envelope and weight constraints. A key restraint was the requirement that backup end-of-life shutdown be accomplished by swinging the reflector halves away from the core.

Reflector drive techniques were narrowed to a ball screw, plain screw, or rack and pinion. The traveling rack and traveling screw options were eliminated as they required penetrations into the shield and, consequently, complicated EOL design. The plain screw was dropped because of high friction. A ball screw with a translating nut was selected as the final drive design (Figure 18), because it is efficient and simple. The major advantages are low drive torque, minimum selfwelding potential, simple and reliable design, and availability of supporting test data at operational conditions.

The actuator was located near the shield instead of at the opposite end of the reactor because of greater design flexibility and more space is available in which to locate the drive components. In addition, this layout shortens the screw and locates it near the center of gravity of the segment, locates the actuator adjacent to the load carrying structure, and shortens the electrical cables.

Expected operational conditions of the ball-screw and drive mechanism are 800°F and 1×10^{-5} torr (air) or less for 5 years. A detailed study of ball-screw

6531-5415

Figure 17. Pressure Drop Through Mixing Model

Figure 18. Sliding Reflector Control Translating Nut Ball Screw Drive

design state of the art was conducted and visits were made to the engineering departments of the two major manufacturers of special application ball-screws. Table 7 lists experience with ball screws operated under similar conditions. The key criteria, currently under study, is the selection of ball nut, screw, and ball materials. A variety of high temperature, high strength alloys are available with test experience. Experiments also show no problem with application of MoS₂ dry-film lubricants which are planned to further lower friction and reduce any self-welding potential.

TABLE 7
HIGH TEMPERATURE BALL SCREW EXPERIENCE

Test	Design	Experience
NASA Lithium Valve Loop	17-4 PH Nut, Hipercut Screw, WC-6Co Ball, 3/8-in.-diameter by 1/8-in. Lead, No Lubrication	5,000 hr at ~800° F in 10 ⁻⁷ torr, 500 to 1000-hr dwell under ~1000 axial load, 95 cycles (3 turns) on 1 unit, 600 cycles (1 to 3 turns) on 1 unit
NASA Nb - 1 Zr	17-4 PH Nut, Hipercut Screw, WC-6Co Ball, 3/8-in.-diameter by 1/8-in. Lead, No Lubrication	2 units used to control throttle valve during 10,000-hr test, >500° F, 10 ⁻⁷ torr
Concorde Jet Silencer Drive (Air-Research)	Nickel Plated H-11, Nut and Screw, WC-6Co Ball, 3/4-in.-diameter by 1/16-in. Lead, X-15 Dry Lubrication	4,000 cycles (~48 turns) under 1700-lb load (~90 hr), 450° F in air, efficiency ~93% (1 unit) - 88% (final)
FFTF IVHM Drive (LMEC)	Inconel-718 Nut, and Screw, Stellite Star-J Ball, 2-7/8-in.-diameter by 0.66 Lead	Soaked in 1100° F sodium and operated at 450° F under 1,450-lb and higher loads

The drive support bearings and gears are based on state-of-the-art technology generated and proven in the SNAP 8 and Advanced ZrH reactor programs. The drive support bearings utilize a self-aligning spherical bearing with a carbon-graphite-against-alumina friction couple. This combination has accumulated in excess of 70,000 hours testing under elevated temperature-vacuum conditions and individual bearing sets have completed 18,000 hours at 1150° F and 1 x 10⁻⁵

torr pressure. No incidents of bearing failure have been encountered. Figure 19 shows the design limits and expected operating friction levels for the bearing couple as a function of temperature. The drive gears are also of proven design with a MoS_2 dry-film lubricant on the gear teeth for low friction operation. The coating has performed well in repeated long-term operational tests, both nuclear and non-nuclear. Table 8 gives the test background of gear performance for previous reactor programs.

TABLE 8
CONTROL DRUM GEAR HISTORY, MoS_2 -COATED GEARS

System	Material	Tests*
SNAP 10A	Stellite 6B (P), Titanium + MoS_2 , 13.6:1 Ratio	1) 90 days at 700° F, 10^{-9} torr 2) 500 hr at 800° F, 10^{-5} torr 3) Shock and vibration 4) SNAP 10A nuclear ground test 5) SNAP 10A space operation
SNAP 8	Inconel 750 (P), Inconel 750 + MoS_2 , 13.8:1 Ratio	1) 15,000 hr at 800° F, 10^{-5} torr (scram kit tests) 2) S8DR half reflector acceptance test 3) S8DRM shock and vibration 4) S8DR 7,000-hr nuclear ground test

*No significant wear, no selfwelding, were noted for either material combination.

Two techniques (Figure 20) were studied to prevent rotation of the reflector segment as a result of ball-screw reaction. One scheme consists of a torsion bellows rigidly attached to the reflector segment and the base plate on the ball screw centerline. Calculations indicate the resulting bellows is practical, but the drive screw support and gear area becomes highly congested.

The selected approach uses a single guide track and translating spherical roller on one edge of the sliding reflector. The guide is in an uncongested area and the reaction force is small due to the large moment arm. The spherical ball and track design is immune to thermal misalignment between the sliding and fixed reflectors. It is easily fabricated from a variety of sources and is extremely

Figure 19. Alumina vs Carbon-Graphite Friction (Static)

6531-5506

Figure 20. Sector Anti-Rotation Method (Nut Reaction)

6531-5507

Figure 21. Room Temperature Spectral Emittance of Anodized Beryllium

simple in design. Use of a MoS_2 dry-film coated ball, or a carbon-graphite ball against an alumina coated track utilizes a proven low friction couple for the reactor environment.

2. Testing

a. Ball Screw Evaluation

Based on the above conceptual sliding reflector design and requirements, thermal-vacuum and launch vibration test fixtures have been designed for preliminary testing of the proposed ball-screw reflector drive mechanism. The thermal-vacuum fixtures will allow testing of the ball-screw mechanism to 1000°F at pressures of 1×10^{-6} torr (air) or less at design loads. Friction torque required to operate the drive will be studied as a function of time, temperature, and operational cycles. This will include long-term (>200 hr) non-operational periods and thermal cycling to operating temperatures.

b. Anodize Beryllium Coating

The beryllium reflector segments will be anodized to enhance thermal emittance. This technique was successfully proven in the S8DR and SNAP 10A reactor programs and has operated up to 7,000 hours in nuclear tests. The supplier who applied the original anodized coating no longer performs the service, but the same coating is applied by another supplier. Testing has been initiated to determine the emissivity, the thermal stability, and the oxidation resistance of the new coating and to compare the above properties with the anodized coating used on the S8DR reflectors. Emissivity measurements are in progress and the room temperature spectral emissivity of the two coatings has been measured and is shown as Figure 21. Eighty percent of the total energy at 1150°F is between approximately 3 and 8 microns wave length. The 80% band shifts to between 4 and 10.5 microns at 500°F . Although the emissivity of the new ABC coating is more stable, the total emissivity of both averages about 82%. Total emissivity tests at 800 and 1150°F are currently being conducted, and will be followed by tests to define oxidation and thermal stability limits.

c. MoS_2 Dry-Film Lubricant

The previous quarterly report⁽¹⁾ discussed the sliding friction evaluation of sodium-silicate-bonded, molydisulfide dry-film lubricant on a René 41 pin and

TABLE 9
MoS₂ FRICTION CHARACTERISTICS
(10⁻³ torr Environment)

Sample	Temperature (° F)	Starting Friction	Running Friction	Time at Temperature (hr)	Time at Temperature Since Last Cycling (hr)	Total Accumulated Cycles
A	600	0.148 to 0.153	0.065 to 0.087	0	0	120
		0.205 to 0.165	0.116 to 0.091	129	129	220
		0.22 to 0.168	0.145 to 0.096	290	161	320
		0.21 to 0.165	0.138 to 0.095	411	121	424
B	800	0.122 to 0.162	0.080 to 0.096	0	0	120
		0.133 to 0.140	0.076 to 0.081	105	105	220
		0.136 to 0.136	0.086 to 0.085	220	115	320
		0.140 to 0.137	0.088 to 0.086	317	97	420
		0.140 to 0.138	0.096 to 0.096	433	116	520
B	900	0.225 to 0.157	0.128 to 0.112	60	60	620
		0.21 to 0.43	0.140 to 0.105	108	48	720
B	1000	0.170	0.102	0	0	730
		0.308 to 0.210		50	50	830
		0.305 to 0.225	0.183 to 0.198	117	67	855

plate system under expected operating temperature, loads and pressure. Using the same procedures and equipment, an investigation of the performance under the slightly oxidizing ground test environment was conducted. The samples were subjected to 400-hour tests at pressures of 1×10^{-3} torr (air) which provides dry-film lubricant oxidizing potential equivalent to 40,000-hour test at pressures of 1×10^{-5} torr (air). The test temperatures investigated were: 600, 800, 900, and 1000°F. Testing at the last two temperatures was limited to 100 hours actual test time, since these temperatures are in excess of the predicted operating temperature. All tests were conducted with a contact stress of 35,500 psi. Table 9 is a summary of the test results. Frictional characteristics at 900 and 1000°F normally began at the higher values indicated in the table and stabilized to the lower values after a few cycles. The increased higher friction at the latter temperature is attributed to the formation of a slightly oxidized layer of MoS_2 during the static dwell. Reduction of friction after a few cycles is probably due to loss of this oxidized layer which exposes fresh MoS_2 . The results of this exposure test demonstrated that MoS_2 should provide adequate lubrication for metallic contact devices such as gears, ball screws, and locking cams.

E. ACTUATOR

A direct current, variable reluctance, stepping actuator(energized in discrete pulses from a controller) is to be used in the reflector drive system to position the reflector sectors to control core reactivity.

1. Design Analysis

Studies were conducted to predict performance of S8DS configuration actuators as a function of various parameters. The analytical results were correlated with test data from S8DS developmental tests for tooth sizes up to 0.060 in. in width. Preliminary results indicate the most desirable actuator performance, within the envelope constraints of the system arrangement, can be obtained with a tooth width of approximately 1/8 in.

2. Tests

Components have been designed for a test to measure the output of S8DR-type actuators with 0.090- and 0.120-in. wide teeth. The results of this testing will be used to verify the analytical predictions of actuator performance for teeth with a width larger than 0.060 in.

Figure 22. Dual-Throat NaK EM Pump Assembly

7759-40250A

F. ELECTRICAL COMPONENTS

1. Hard Cable

Cables are required to supply power to the actuator, limit switch, and backup shutdown device, to provide high mechanical integrity and avoid outgassing of organic materials; inorganic cables were selected which consist of a metal sheath assembly swaged from metal tubing, ceramic insulation, and an electrical conductor. Both single-conductor and multiple-conductor cables were initially evaluated. Because of termination difficulties, as well as cable stiffness, the multiple conductor cable concept was dropped. Cable assemblies approximately 1/8-in. in diameter, are now under test to evaluate sealing techniques and the effect of pre-flight humidity on performance. Preliminary designs for termination of the hard cable to the limit switch and the actuators have been selected.

2. Limit Switch

The limit switch developed and tested for the SNAP 8 program will be utilized, with some modifications, in the 5-kwe Reactor Thermoelectric System. Preliminary design sketches have been made to verify the feasibility of incorporating the desired design changes in the limit switch to improve switch quality (resistance to insulation degradation) and facilitate connection to the hard cable.

G. NaK EM PUMP

1. Development Pump

A dual-throat d-c conduction electromagnetic NaK pump, Figure 22, was designed, and sufficient parts to assemble two units were partially fabricated several years ago. Welding and brazing processes were also developed and qualified, although no welded or brazed subassemblies were produced. Because of the general similarity of the configuration and performance requirements, tests on this pump will provide information useful to the design of the system pump.

The fabrication and assembly of one pump assembly for development evaluation was recently reactivated and the following activities were completed:

- 1) Inventory of completed and partially completed parts
- 2) Review of the design analyses and drawings

6531-5416

Figure 23. EM Pump and Thermoelectric Converter

- 3) Review of Manufacturing and Quality Assurance processes and documentations
- 4) Updating of drawings and other engineering documentations
- 5) Planning for the fabrication completion and assembly.

Testing of this development pump will be performed in a vacuum chamber which will be mated to an existing NaK test loop. Initially, the pump will be operated with an externally controlled power supply. Later, the pump will be mated to a thermoelectric pump converter assembly. Planning for these tests was started and a conceptual test configuration was designed. Design of a modification to the PCA NaK test loop was performed. Equipment required to modify the NaK loop and an existing vacuum chamber was designed.

2. System Pump

On the basis of a trade study, a dual-throat, d-c permanent magnet pump was selected over other type pumps for supplying system primary and secondary head. The dual-throat permanent magnet electromagnetic type pump was selected for use on the system because it is significantly lighter than a-c pumps which could produce present system hydraulic pumping power requirements and somewhat lighter than two individual d-c pumps. Figure 23 shows a conceptual isometric drawing of the dual-throat pump assembly mated to three thermoelectric pump modules.

H. NaK VOLUME ACCUMULATOR

The liquid metal, NaK, used in the primary and secondary coolant systems for the reactor thermoelectric power generator requires an expansion compensator. This device provides an expansion volume during heating up of the system and maintains a pressure on the system within specific limits to preclude cavitation or excessive pressure in the system. The critical item in an expansion compensator is the bellows.

Trade studies have been started for the volume accumulator based on system requirements of primary and secondary loop NaK volume, operating temperatures, and NaK loop pressure requirements. Conceptual designs of volume accumulator based on welded or formed bellows and spring or gas pressurization to maintain void free condition are being assessed for size, weight, NaK volume, fabrication, and quality assurance processes and operational reliability.

BLANK

IV. FUEL DEVELOPMENT

A. INTRODUCTION AND SUMMARY

The fuel development effort is directed toward establishing the process parameters for reproducibly fabricating fuel elements, and evaluating fuel element performance under simulated reactor conditions. Evaluation of cladding properties is a part of this task.

During this report period, coated butt-weld studies on Hastelloy X resulted in joints with acceptable metal and SCB coating quality. Hobbing studies were initiated on Hastelloy X cladding. Thermal and mechanical testing of fuel elements continued. Vibration tests indicate that fuel elements are unaffected by anticipated launch loads from either Saturn V or the Titan III-C. Hastelloy X cladding property testing continued. Some tensile data on 5000-hour-aged tubes was obtained and stress rupture testing in NaK was initiated.

B. FUEL ELEMENT MATERIALS EVALUATION

1. Fuel Element Assembly and Test

Preliminary parameters were established for EB butt-welding 28-mil-wall, Hastelloy X coated, end caps to coated tubes. Metallographic and visual examination of the welds indicate full penetration of the base metal and complete fusion of the SCB over the weld joint. No spalling has been observed after second coating and firing. The current parameters, while not fully optimized, are considered adequate for developmental elements of the 0.540-in.-ID, 28-mil-wall configuration. Adaptation of these parameters to the interim size 0.651-in.-ID, 28-mil cladding was initiated.

In the previous report, difficulties in achieving acceptable coated butt welds with Incoloy 800 were described. Process parameters were varied but did not improve the SCB coating adherence over the weld zone. It must be concluded that a good quality, coated butt weld is not achievable with Incoloy 800, based on the results of the extensive weld study.

Tensile test data on Hastelloy X cladding from Heat E9504 (0.10% cobalt) has shown a reduction in ductility between simulated thermal treatment (STT) and simulated process (SP) specimens from 36% to 19%, respectively.⁽²⁾ The

TABLE 10
EFFECTS OF PROCESSING ATMOSPHERE ON TENSILE PROPERTIES

Group Identification	Number of Specimens	Oxidation	Firing Atmosphere	Objective	Yield Strength 0.2% Offset (ksi)	Ultimate Strength (ksi)	Elongation (% in 2-in. gage)
STT	3		Encapsulated in 0.1 atm He	Control group	51	117	35
SP-1	2	Air - ID He - OD	In air in a shroud tube	"Old" simulated process to show effects of ppt's	51	113	24
SP-2	2	O ₂ /Ar mixture - ID O ₂ /Ar mixture - OD	In air with no shroud	Prevent OD ppt's	50	114	24
SP-3	3	O ₂ /Ar mixture - ID He - OD	He on OD	Prevent OD ppt's	49	114	26
X-1	3	O ₂ /Ar mixture - ID O ₂ /Ar mixture - OD	Encapsulated in 0.1 atm He	Evaluate effect of "protective" OD oxide	49	112	29
X-2	3	O ₂ /Ar mixture - ID He - OD	Encapsulated in 0.1 atm He	Evaluate effect of ID oxide layer	49	113	29
X-3	3	O ₂ /Ar mixture - ID O ₂ /Ar mixture - OD	In air* with SCB coating and no shroud	Evaluate effect of SCB coating during firing	50	113	27

*SCB coating removed with HF prior to tensile testing.

reduced ductility was attributed to a narrow band of precipitates observed at the OD of the SP specimens. Conditions thought to be potentially responsible for the reduced ductility (precipitates) were: (1) the oxidation of cladding ID required for SCB coating, (2) the formation of precipitates on the cladding OD during simulated coating firing in air, or (3) a combination of the first two items.

A series of tensile tests was conducted to study processing effects on ductility. Hastelloy X cladding from Heat C9591 (0.9% cobalt) was used because all of Heat E9504 had been used. Two processing approaches were used in an attempt to prevent or minimize precipitate formation, i. e., (1) maintaining an inert atmosphere at the cladding OD during all process steps, or (2) intentional oxidation of the cladding OD to form a "protective coating" prior to the 2150° F coating firing step. The test groups consisted of three specimens each, which were aged for 100 hours in helium at 1300° F after the indicated processing. The tests, objectives, and tensile test results from the series are shown in Table 10.

The data indicate that the formation of the ID oxidation layer alone reduces the ductility from 35 to 29% (Test STT vs X-1 and X-2). Maintaining an inert atmosphere at the OD (SP-3) resulted in a ductility of 26%. Intentional oxidation of the OD prior to firing in air with SCB-coated ID resulted in 27% ductility, i. e., about the same ductility as with inert atmosphere at the OD. Intentional oxidation of the OD prior to firing the SCB coating was selected as the reference coating process. This decision was made based on (1) ductility being similar to inert gas processing, and (2) the fact that inert gas processing is more complex, difficult to control, and yields an SCB coating which is different in appearance and performance than the direct air-fired SCB coating.

Helical fins were machined on 19 Hastelloy X cladding tubes by hobbing. A tool steel hob was used with special tooling to support the tubing. The original support tooling was constructed such that shims were needed each time the machine was adjusted. New tooling was designed and fabricated which reduced the time and skill required for machine setup and hobbing operations. A comparison of wall thickness variation in the cladding before and after hobbing is presented in Table 11. Groups A and B were machined with the original tooling and Group C was machined with the new tooling. All wall thickness data were

TABLE 11
HOBBED TUBE WALL THICKNESS VARIATION BATCH TO BATCH

Batch	Pre-Hob	A	B	C	All (A,B,C)
Number of Tubes	10	6	4	9	19
Mean Wall (in.)*	0.507	0.0282	0.0269	0.0292	0.0284
Standard Deviation (in.)	0.0006	0.0007	0.0008	0.0006	0.0022

*Target wall thickness was 0.0280 ± 0.001 in.

obtained with a Vidigage. Wall thickness variation within each batch of hobbled tubes approached that of the feed stock. The standard deviation of wall thickness within individual tubes was ~ 0.0005 in. Little, if any, improvement beyond this can be expected due to the limitations imposed by variations in the feed stock. Improvement in controlling nominal wall thickness from batch to batch is possible and tooling aids for the machinist are being designed to this end.

Thermal testing of a group of Hastelloy X developmental elements continued. These elements have 20-mil-wall cladding, segmented fuel with an H/Zr of 1.81 and a 0.1-atm helium backfill. Permeation data are summarized in Table 12.

A series of vibration tests was conducted to demonstrate the ability of the fuel elements to withstand the loads imposed during launch. The fuel elements were assembled with various fuel-rod-to-vibration-suppressor axial gaps. The fuel elements were vibrated in the vertical and horizontal axes with both the simulated Saturn V and Titan III-C vibration spectra. The tests imposed 100 and 150% of the anticipated Saturn V loads. In the case of Titan III-C, one test was made at the anticipated loads and an additional test was conducted at 10 times the anticipated loads. A peak acceleration spectral density of $0.0085 \text{ g}^2/\text{Hz}$ occurs between 80 and 160 Hz for the Saturn V. For the Titan III-C, the peak acceleration spectral density of $0.015 \text{ g}^2/\text{Hz}$ occurs between 550 and 1000 Hz. The Titan III-C peak occurs much higher than the natural frequency of the fuel element (i. e., 150 Hz) and for this reason is a more benign test. No damage to the fuel elements was indicated by permeation data taken before and after each vibration input. Permeation data are shown in Table 13.

TABLE 12
HASTELLOY X ELEMENT PERMEATION DATA

Element Number	As-Measured Permeation [cc(STP)/hr at 1300° F]					
	Initial Test	Post-Scan	Post-Ramp	1000-hr Aging	2000-hr Aging	3000-hr Aging
X-7	0.13	-	0.16	0.25	0.40	0.26
X-8	0.17	-	0.22	0.23	0.26	0.31
X-11	0.13	-	0.18	0.24	0.34	
X-15	0.14	0.13	0.16	0.11	0.23	
X-17	0.20	0.19	0.24	0.20	0.29	
X-19	0.25	0.19	0.21	0.19	0.25	
X-10*	0.18	-	0.18	0.08	0.14	
X-13*	0.17	-	0.18	-	0.16	

*Elements X-10 and X-13 are aged at 1050° F; all others aged at 1300° F.

TABLE 13
VIBRATION/PERMEATION TEST DATA

Element Number	Fuel Rod to Suppressor Gap (in.)	Permeation Rate [cc(STP)/hr at 1300° F]				
		As-Built	Saturn V 100%	Saturn V 150%	Titan III-C 100%	Titan III-C 1000%
X-35	0.005	0.37	0.20	0.28	0.31	*
X-41	0.015	0.18	0.36	0.28	0.31	*

*Permeation testing in progress.

A series of tests was conducted to demonstrate the integrity of the fuel element under loading conditions simulating reactor operation. The test system was designed to simulate an ~50 mils deflection (bending) with the bending load imposed at ~0.7 Z/L at a peak temperature of about 1200° F. A quartz rod was used to apply the load, the end of the rod having the geometry of a cladding fin. The load required to achieve the desired deflection was measured

6531-51110

Figure 24. Load Relaxation Test Rig

with a load cell. A schematic of the test system is shown in Figure 24. Effects of deflection due to changes in geometry when the load was applied (i. e., changes in thermal stress with changes in cross element ΔT) were corrected by pre-test calibrations.

Two elements were tested: (1) an Incoloy 800 element with 0.032-in. cladding wall, and (2) a 0.020-in.-wall Hastelloy X fuel element. Each was a fully processed element. On the first test, the Incoloy 800 element was loaded to 20 lb and a deflection of approximately 0.050 in. The load relaxed to approximately 14-1/2 lb at the end of 200 hours while maintaining the preset deflection. There was no degradation in permeation rate of the element at the end of approximately 350 hours and the test was terminated. There was concern with the high rate of load relaxation and possibility of slippage of the element in the test rig. A subsequent test was initiated with a 17-lb load which relaxed to 12.2 lb at the end of 80 hours. Again, there was no degradation in hydrogen permeation, but the load relaxation was repeated. Then the Hastelloy X element was put on test with an approximate 0.050-in. deflection which required an 11-lb load. This load relaxed to approximately 6 lb in 100 hours. The first test was terminated at the end of 220 hours with no barrier degradation; a second test was conducted with almost identical results. A final Hastelloy X relaxation test was conducted with more care in measuring the axial and radial temperature gradients. The deflection-load history is shown in Figure 25. The element was deflected 0.046 in. requiring a load of 12-1/2 lb which decayed to 7.6 lb in 120 hours. The load was then reduced to zero while maintaining temperature. There remained a 0.020-in. permanent set in the element which correlates well with the load relaxation. The element was then deflected an additional 0.050 in., with a load of 14 lb which decayed rapidly repeating the trend established. The same Hastelloy X element was used in these and prior tests; permeation rate was unchanged indicating no barrier damage. These tests have demonstrated that the hydrogen barrier, on either Incoloy 800 or Hastelloy X, is undamaged by severe bending loads.

2. Burnable Poison

Four fuel elements were removed from test after 10,000 hours at temperature. These fuel elements, two each on test at 1300 and 1400°F, were replicates of the 1500- and 3000-hour test elements discussed in the previous

6531-51111

Figure 25. Hastelloy X Load Relaxation Test

quarterly.⁽¹⁾ One set contained fuel rods coated with Gd_2O_3 or Y_2O_3 ; the other set contained fuel rods coated with Gd_2O_3 , Y_2O_3 , and/or Eu_2O_3 . The results were virtually the same as for the shorter tests with only trace transfer of Y or Gd to the adjacent fuel or to other fuel rods and massive transfer of the Eu to the adjacent SCB, where the Eu remained.

The 1430° F compatibility test of Eu in SCB terminated after 1500 hours. No movement of Eu from or in the SCB was detected.

Natural erbium and Er^{167} exhibit negative temperature coefficients of nuclear reactivity in SNAP-type reactor spectra and therefore offer promise for use, in place of europium, to balance the positive temperature coefficient contribution of large quantities of gadolinium. During the current report period, Er_2O_3 and Er_2O_3/Gd_2O_3 coatings (0.7 to 1.5 in. thick) were vapor deposited on fuel rod segments. These were assembled into four fuel elements, each containing eight 1-in. fuel segments. Two elements each were placed on test at 1400 and 1500° F. One of the 1500° F test elements was removed from test after 72 hours and sectioned and checked by X-ray fluorescence (XRF) to make certain that there was no early movement of erbium. (Early movement of europium has been observed.) There was no movement of erbium in this or in the second 1500° F test element, which was examined after 1101 hours. The 1400° F tests are scheduled for 5000- and 10,000-hour exposure prior to examination.

C. CLADDING PROPERTIES

Tensile tests were conducted on Hastelloy-X cladding from Heat E9504 (0.1% cobalt) which had been thermally aged for 4544 hours at 1200° F. One set of data is from cladding that was heat treated for the time and temperature of the coating process cycle in an inert environment. This treatment is designated STT. Another set of specimens was heat treated in the coating environment but not actually coated. This treatment has flowing air on the ID of the tube and flowing helium on the OD during the oxidation cycle. The specimens were then fired in shrouds in an air furnace. This heat treatment is designated SP. All specimens were given a simulated permeation cycle of 100 hours at 1300° F, followed by 4544 hours at 1200° F. The results of the tensile tests are presented in Table 14, in conjunction with prior test data.

TABLE 14
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
	RT	4544	68	126	25
	1100	-	37	90	39
	1200	-	36	84	46
	1200	300	37	84	53
	1200	1000	44	87	48
	1200	4544	52	91	42
Simulated Process (SP) Condition	RT	-	54	110	19
	1100	-	40	91	28
	1200	-	37	84	36
	1200	300	38	84	40
	1200	1000	41	86	32
	1200	4544	49	92	36

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

The strength of the alloy in both conditions continues to increase with aging time. The ductility of the STT material continues to degrade but it is still quite high, i. e. , 25% and 41.7% at room temperature and 1200° F, respectively. The initial ductility of the SP material at 1200° F was approximately 10% lower than that of the STT material. After 4544 hours thermal aging at 1200° F, this difference has been reduced to 6.0%, with STT still being the more ductile. The yield strength and ultimate strength for both SP and STT heat treated materials are increasing as a function of thermal aging time and are essentially the same.

Secondary creep rates have been determined at 1100, 1200, and 1300° F. The specimens were from both a high and a low cobalt heat and were tested in

the unaged condition or after thermal aging 1000 hours at 1200 or 1300°F. The minimum creep rates at 1200°F for the aged and unaged specimen are presented in Figure 26. It was previously reported that the high cobalt heat was anomalously weaker after aging. Post-test measurements have shown that the wall thickness was thinner on this specimen and the stress was actually higher. The complete data set shows that aged specimens are usually stronger. However, at low creep rates the aged and unaged material from both heats have nearly the same creep rate. The minimum creep rate data for the low cobalt heat alone is given in Figure 27. The tendency for the aged and unaged data to approach each other is clear in this figure.

Nine creep tests were conducted on Hastelloy-X irradiated cladding from the NAA-120-4 experiment. The cobalt content of this cladding is higher than that of the high cobalt heat of the unirradiated tensile and creep test series. The heat chemistry is given in Table 15. Irradiation temperature and data from the creep tests are given in Table 16. The stresses were calculated using the actual load applied by the 3:1 magnification test machine and the areas are determined from pre-test measurement of the wall thickness and width of the gage section. The irradiation temperatures were determined from the average thermocouple readings obtained during the irradiation. Five of the nine irradiated tests broke at the extensometer attachment point, invalidating the ductility data. However, all nine tests were of long enough duration to obtain a minimum creep rate. Four of the irradiated tests broke in the gage section and gave meaningful ductility values of 1.0% at 1300°F and 2.0 to 2.1% at 1200°F. Minimum creep rate curves for the tests are given in Figure 28. A comparison of the minimum creep rates of irradiated Heat C6505 and unirradiated but aged C9591 is given in Figure 29. The line through the data in Figure 29 is the best fit for the irradiated data set. At creep rates in the 10^{-5} to 10^{-6} in./in./hr range, the irradiated material is somewhat stronger than the aged material.

Biaxial stress rupture tests are in progress in helium and flowing NaK environments. The data from the completed tests are given in Figure 30. There is considerable scatter in the rupture times for the helium controls. The two NaK data points at 1300°F are closely grouped and fall between the two data points obtained in helium at the same stress.

6531-4719

Figure 26. Creep Rate Comparison of High and Low Cobalt Hastelloy X

TABLE 15
NAA 120-4 HASTELLOY X CLADDING CHEMISTRY

Element	Chemical Analyses (wt %)
	C-6505*
Cr	21.36
Fe	17.80
Mo	8.47
Co	1.46
W	0.54
Si	0.79
Mn	0.64
P	0.015
S	0.008
C	0.12

*Universal Cyclops Specialty Steel Division

TABLE 16
NAA 120-4 HASTELLOY X CREEP DATA, HEAT C-6505

Test Temperature (° F)	Irradiation Temperature (° F)	Stress (ksi)	Rupture Time (hr)	Rupture Ductility (%)	Specimen Number	Minimum Creep Rate (in. /in. /hr)
1300	1160	24.2	5.9	*	4V1-2	6.7×10^{-4}
1300	1200	25.5	15.6	0.99	4M1-2	5.8×10^{-4}
1300	1190	19.0	83.8	*	4M2-1	5.6×10^{-5}
1300	1240	14.7	607	0.97	4M4-2	1.1×10^{-5}
1200	1140	28.7	90	2.0	4M5-1	1.8×10^{-4}
1200	1140	24.5	266	2.1	4M5-2	6.1×10^{-5}
1200	1200	18.8	573	*	4M1-1	7.7×10^{-6}
1200	1160	19.1	511	*	4V1-1	3.1×10^{-6}

*Broke at extensometer attachment.

6531-4720

Figure 27. Uniaxial Creep of Low Cobalt Hastelloy X (E9504), Aged and Unaged

6531-4721

Figure 28. Uniaxial Creep of Irradiated Hastelloy X, Heat C6505, NAA 120-4

6531-4722

Figure 29. Uniaxial Creep Comparison of Irradiated vs Aged Hastelloy X at 1200°F

Figure 30. Biaxial Stress Rupture of Hastelloy X (C9591)

6531-4723

BLANK

V. THERMOELECTRIC CONVERTER DEVELOPMENT

A. REACTOR POWER MODULES

1. Introduction

During the past quarter, major emphasis was placed on the continued evaluation of performance data from four refractory-clad power modules tested throughout the reporting period, on the application of glass seals and preparations for testing of three additional modules, and on the generation of module parametric calculations in support of reactor-thermoelectric systems studies.

All reactor power modules continued to operate with power output degradation rates ranging from 1.0 to 3.5% per 10,000 hours for operating times up to 8500 hours, operating temperatures as high as 1140° F average (1190° F peak), and including up to 58 thermal cycles per module. The first reactor power module incorporating glass-to-metal pin base seals was placed on test and performance essentially duplicated that of the previous TEM-X S/N-3 reactor power modules.

2. TEM-X S/N-3 Series Modules

Four refractory-clad TEM-X S/N-3 series modules were tested continuously throughout the past quarter. TEM-X S/N-3C and S/N-3E have accrued 8487 and 6667 hours of static testing at WANL, respectively, at $\bar{T}_H = 1085^\circ\text{F}$ and $\bar{T}_C = 507^\circ\text{F}$ test conditions with the former being thermally cycled 57 times. Overall efficiency, electrical power output, open circuit voltage, and internal resistance life test data from these modules are presented in Figures 31 through 34, respectively, along with similar data from all other TEM-X S/N-3 series modules currently on test. The excellent degree of performance stability of all refractory-inner-clad modules tested to date is quite evident from these data.

Testing of two refractory-clad modules in the liquid metal loop system also continued throughout the past quarter. TEM-X S/N-3D has accrued 5049 hours at average operating temperatures identical to those of the static tested modules discussed above. TEM-X S/N-3F has been operating for 1011 hours at $\bar{T}_H = 1085^\circ\text{F}$, $\bar{T}_C = 507^\circ\text{F}$, and 2829 hours at elevated $\bar{T}_H = 1140^\circ\text{F}$, $\bar{T}_C = 523^\circ\text{F}$, test conditions. The high temperature data are clearly distinguishable in Figures 31 through 34. Both reactor power modules on loop test are being operated with 100° F axial temperature drops.

Figure 31. Overall Efficiency Versus Elapsed Time for Refractory-Clad Reactor Power Modules

Figure 32. Power Output Versus Elapsed Time for Refractory-Clad Reactor Power Modules

Figure 33. Open Circuit Voltage Versus Elapsed Time for Refractory-Clad Reactor Power Modules

Figure 34. Internal Resistance Versus Elapsed Time for Refractory-Clad Reactor Power Modules

TABLE 17
REFRACTORY-CLAD REACTOR MODULE PERFORMANCE STABILITY SUMMARY

Module	$\bar{T}_H/\bar{T}_C$ (°F)	Testing Time (Hours)	Rate of Change: % Per 10, 000 Hours				Thermal Cycles	
			R_g	V_{oc}	P_e	η_o	Rapid	30°F/Min.
TEM-X S/N-3A*	1085/507-S	1166	4.0	1.1	-1.9	-2.0	1	49
TEM-X S/N-3B*	1085/507-S	2584	2.8	0.1	-2.5	+3.0	0	0
TEM-X S/N-3C	1085/507-S	8487	2.9	0.7	-1.5	-0.2	4	53
TEM-X S/N-3D	1085/507-L	5049	5.5	0.9	-3.5	+0.3	5	53
TEM-X S/N-3E	1085/507-S		2.5	0.8	-0.8	-0.4	3	0
TEM-X S/N-3F	1085/507-L	1011	6.0	2.5	-1.0	+4.1	5	1
TEM-X S/N-3F	1140/523-L	2829	8.7	3.2	-2.3	(A)		
TEM-X S/N-3G	1085/507-S	236	(B)	(B)	(B)	(B)	0	0

*Testing Terminated

L Indicates Loop Test

S Indicates Static Test

(A) Not calculated due to flow meter fluctuations.

(B) Insufficient data for meaningful analysis.

All static test data included through 4/17/72

All loop test data included through 4/17/72

A statistical analysis (linear curve fit) of all life test data presented in Figures 31 through 34 is presented in Table 17. The reproducibility of the excellent long-term stability of the TEM-X design is evident from these data. The calculated power output degradation rate of TEM-X S/N-3D has increased from 2.1%/10,000 to 3.5%/10,000 during the past 2000 hours of test. It is not known at this time whether this is a true increase in module degradation or is due to instrumentation degradation. The increased degradation rate infers a power output decrease of 0.9 watt (about 0.3%) greater than would have occurred with the initial 2.1%/10,000 degradation rate. The data from this module are being examined in an attempt to ascertain the cause of the apparent increase.

An additional reactor power module, TEM-X S/N-3G, was placed on test during the past quarter. This module is identical to the four earlier modules with the exception that glass-to-metal seals were incorporated at each pin base to allow testing in an air environment. Previous modules incorporate hermetically sealed caps installed over the ends of the module which are back-filled with argon.

Effective Seebeck Coefficient (defined as the ratio of the open circuit voltage to the average module radial temperature drop), internal resistance, and thermal impedance data as functions of average module temperature during the open circuit heatup of TEM-X S/N-3G are compared to calculated values as well as similar data from other modules in Figures 35, 36, and 37, respectively. Modules on loop test are heated up under load and no direct comparison of the data from these modules can be made. TEM-X S/N-3G heatup data can be seen to correlate excellently with both the calculated lines and previous data from other modules over the entire temperature range. The deviation between measured and calculated thermal impedance near operating conditions is felt to be a result of using higher than actual high temperature lead telluride thermal conductivity data in the calculational code TEMOD.

Two other TEM-X series modules, S/N-3H and 3J, were being prepared (instrumented) for static test at the writing of this report. Both of these modules will be tested at elevated temperatures. Assembly of TEM-X S/N-3K was initiated this quarter. This module will be tested in a two-loop liquid metal system with a space environment (vacuum). Operating conditions (primary and

Figure 35. A Comparison of Calculated and Experimental Effective Seebeck Coefficient Data Recorded During Open Circuit Heatup of TEM-X Modules

Figure 36. A Comparison of Calculated and Experimental Internal Resistance Data Recorded During Open Circuit Heatup of TEM-X Modules

Figure 37. A Comparison of Calculated and Experimental Thermal Impedance Data Recorded During Open Circuit Heatup of TEM-X Modules

secondary NaK temperatures and flow rates) for this module will be selected on the basis of the projected operating conditions of the 5-kwe demonstration test.

3. High Voltage Refractory-Clad Module (TEM-X S/N-4)

The next generation reactor power module is directed at demonstrating that tubular converters can deliver power levels equivalent to that of the TEM-X S/N-3 series (300 watts) at substantially higher voltage levels while maintaining an acceptable power degradation rate. To accomplish this, TEM-X S/N-4 was designed with a reduction in thermoelectric washer thickness from 0.188 in. (as in TEM-X S/N-3 series modules) to 0.090 in. This provides approximately a factor of two increase in voltage. Rationale behind this selection is discussed in the previous quarterly progress report.⁽¹⁾

All engineering design and drafting work on TEM-X S/N-4 has been completed; the module is released for fabrication, and assembly has been initiated.

4. Module Parametric Analysis and Preliminary Design

During the past quarter, an effort was initiated to provide parametric performance information as required in support of the 5-kwe system design.

The initial phase of this work was involved with determining the effects of module power degradation. The system is to be designed to deliver 5 kwe for a 5-year mission. The optimum technique for providing a constant power output is to allow the operating temperatures to increase at a rate that will compensate for module degradation. The system, then, will reach its design (maximum) temperatures at the end of the 5-year mission. Since the module is always operated below the design operating temperatures, degradation will be decreased.

Temperature parametrics (i. e., calculated performance parameters as functions of operating temperatures) are being generated for three candidate reactor power module designs for the 5-kwe system.

Figure 38. Electrical Current Data from Three Prototype Pump Power Modules

B. PUMP TUBULAR POWER MODULES

1. Performance Summary of Prototype Pump Modules

Two tubular modules designed to deliver power to an electromagnetic pump continued on test during the past quarter. These prototype pump modules, designated as TEM-14A S/N-1 and TEM-14A S/N-2, are being endurance tested at nominal average cladding temperatures of $\bar{T}_H = 1125^\circ\text{F}$ and $\bar{T}_C = 570^\circ\text{F}$.

Electrical current and electrical power output data from the pump modules are presented in Figures 38 and 39, respectively. Performance levels achieved in S/N-2, 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 18 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.

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, a previously tested pump module, and the various uncontrolled shutdowns of all three modules indicate that the TEM-14 design is insensitive to thermal cycling.

2. Pump Module Parametric Analysis

During the past quarter, work was initiated to provide redesigned pump module parametric performance data in support of the 5-kwe system design.

Initial systems work indicates that an optimum EM pump configuration requires between 550 and 700 amp at approximately 0.2 volt. The only module design parameter which can be changed independently while meeting the above specified performance requirements and operating conditions, is the number of thermoelectric washers in the module. Table 19 presents the results of the parametric analysis. The analysis shows that a module with eight thermoelectric washers (four couples), provides the maximum efficiency. It is interesting to note, however, that a nine-washer module is somewhat shorter, hence lighter.

Figure 39. Power Output Data from Three Prototype Pump Power Modules

TABLE 18
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 ($^{\circ}$ F/Min)
TEM-14A S/N-1	17847	*	*	-4.8	-6.2	9	0 -
	10204**	1.2	-1.0	-3.2	-3.2	2	2 30
TEM-14A S/N-2	20362	3.6	-0.6	-4.3	-3.9	10	2 30

*No open circuit data taken for S/N-1 during first 17847 hrs of testing. All modules tested at $\bar{T}_H = 1125^{\circ}$ F, $\bar{T}_C = 570^{\circ}$ F
All data included through April 17, 1972.

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

TABLE 19
0.2-volt, 700-amp PUMP MODULE PARAMETRIC RESULTS
 $\bar{T}_H = 1168^\circ\text{F}$, $\bar{T}_C = 522^\circ\text{F}$

No. of T/E Washers *	Circuit Length (Inches)	V Load (Volts)	Current (Amps)	Power Out (Watts)	Heat Input (Watts)	Efficiency (Pct.)
7	8.68	0.20	700	140	5015	2.79
8	6.46	0.20	700	140	4603	3.04
9	6.22	0.20	700	140	4614	3.03
10	6.24	0.20	700	140	4743	2.95
12	6.56	0.20	700	140	5138	2.72
14	7.05	0.20	700	140	5610	2.50

*A standard "couple" consists of one pair of thermoelectric washers.

Such a module, however, would have an uneven number of washers and would require a new collector ring design since the circuit would terminate with a hot junction on one end and a cold junction on the other. All previous modules terminate with cold junctions on both ends of the circuit.

The relatively low efficiencies shown in Table 19 result from the large Joule power losses in the module conductor ring and access pins located at the ends of the module. For example, over 22% of the total power produced in the thermoelectric circuit of the module is dissipated as Joule heat in the electrical access pins. A temperature performance parametric for the eight-washer (4 couple) pump module listed in Table 19 is being prepared.

C. SUPPORTING TECHNOLOGY DEVELOPMENT

1. Sublength Modules

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 in static test facilities.

A list of all the process and design features of each TEM-15 series module fabricated, or scheduled for fabrication, to date is presented in Table 20. Twelve TEM-15 series modules were tested continuously throughout the past quarter.

Table 21 lists the operating conditions of all TEM-15 series modules tested during this reporting period and summarizes the results of a degradation analysis. Degradation rates were determined using all data recorded through April 17, 1972.

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

TABLE 20
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. (Tested at $\bar{T}_H = 1025^\circ\text{F}$, $\bar{T}_C = 480^\circ\text{F}$)
15A S/N-4H	Same as 15A-4F except leak-tight glass seals incorporated.
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 15B-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 prevented from extending beyond mica at O. D. and tungsten insert clearance eliminated.
15B S/N-4C	Same as 15B-4B except W foil reduced from full to 1/2 radius.
15B S/N-4D	Same as 15B-4B except W foil extended through inner conductor ring.

TABLE 20
PROCESS AND DESIGN FEATURES OF TEM-15 SERIES MODULES
(Sheet 2 of 2)

Module	Process and Design Innovation
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.
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.

TABLE 21
TEM-15 PERFORMANCE ANALYSIS SUMMARY

Module	Test Conditions $\bar{T}_H/\bar{T}_C$ (°F)	Testing Time (Hours)	Rate of Change: % Per 10,000 Hrs.				Thermal Cycles	
			R_g	V_{oc}	P_e	η_o	Rapid	30°F/
TEM-15A S/N-1	1125/570	19,538	32.0	1.6	-17.9	-15.3	9	1
TEM-15A S/N-4D	1125/570	19,789	32.1	1.7	-17.4	-15.0	8	1
TEM-15A S/N-4F	1125/570	8,946	28.0	1.7	-19.9	-15.3	3	1
TEM-15A S/N-4G	1025/480	3,740	8.8	2.2	-4.2	-2.1	3	0
TEM-15A S/N-5	1125/570	19,066	30.6	2.1	-16.7	-14.0	11	51
TEM-15B S/N-4A	1125/570	14,038	14.1	2.7	-7.2	-6.3	5	1
TEM-15B S/N-4B	1125/570	8,947	17.7	1.6	-12.7	-10.5	5	1
TEM-15B S/N-4C	1125/570	5,210	21.4	3.3	-13.1	-9.0	3	0
TEM-15C S/N-4A	1125/570	5,534	21.1	3.8	-12.2	-8.6	2	1
TEM-15C S/N-8	1125/570	17,228	24.5	2.1	-14.6	-11.9	8	1
TEM-15D S/N-4	1125/570	15,256	19.2	2.4	-11.1	-8.4	7	1
TEM-15H S/N-4	1125/570	9,334	22.0	2.7	-14.0	-12.6	3	1

All data included through April 17, 1972.

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 and overall efficiency data from the first four of these modules are presented in Figures 40 and 41, respectively, as functions of elapsed time. All of these modules are being tested at $\bar{T}_H = 1125^\circ\text{F}$, $\bar{T}_C = 570^\circ\text{F}$ operating conditions and have testing times ranging from 9000 to 19,700 hours. As can be seen in the figures, excellent long-term performance reproducibility has been achieved in all TEM-15A series modules, indicating that none of the minor design, processing, or fabrication modifications investigated in this series of experiments, has had a gross effect on module performance. To date, none of the modules has indicated a tendency to deviate from the characteristic linear power degradation despite the long testing times.

TEM-15A S/N-5 has completed 4900 hours of steady-state operation since completion of 50 controlled thermal cycles. The time at which the cycles were performed is indicated in Figures 40 and 41. Analysis of the data shows that the degradation rates have remained essentially unchanged from the pre-cyclic test levels.

A more recent TEM-15A series module, designated S/N-4G, remained on test during this reporting period. This module, which is identical to TEM-15A S/N-4F, is being steady-state endurance tested at $\bar{T}_H = 1025^\circ\text{F}$ and $\bar{T}_C = 480^\circ\text{F}$ operating conditions, thus duplicating the inner-to-outer cladding thermal expansion mismatch existing in all other TEM-15 modules at the standard 1125/570° F operating conditions. Performance data from this module are discussed in a later section of this report.

An additional TEM-15A series module, S/N-4H, was fabricated during the past month of this quarter. This module includes all end closure design modifications required to incorporate glass-to-metal seals at the electrical pin bases. The module will be tested to determine if the elimination of air leakage at the pin bases has a measurable effect on module performance. TEM-15A S/N-4H will be instrumented and placed on test during the next quarter.

Figure 40. A Comparison of Power Output Data from TEM-15A Series Modules

Figure 41. A Comparison of Overall Efficiency Data from TEM-15 A Series Modules

Figure 42. A Comparison of Power Output Data from TEM-15B, Tungsten Foil Barrier 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. Three TEM-15B series modules continued on test throughout this reporting period.

Electrical power output and overall efficiency data from the tungsten foil barrier modules are presented in Figures 42 and 43, respectively. The individual design features of each of the modules are listed in Table 20. 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 21.

Both the data plots and the degradation analysis show that the performance stability of the 0.010-in. mica module (TEM-15B S/N-4A) is substantially better than that of either module incorporating 0.005-in. mica. In addition, a comparison between the two more recent TEM-15B series modules verifies that substantial performance improvements can be achieved, with little or no power degradation penalty by terminating the foil barriers midway in the thermoelectric washers. Temperatures beyond the midpoint are too low to sustain tellurium diffusion.

Degradation rate data and destructive examination of tungsten foil barrier modules indicate that the primary residual mode of degradation in the TEM-15B series modules is tellurium transport around the ends of the tungsten foil. To eliminate this phenomenon, various techniques for sealing the tungsten foil diffusion barriers to the inner conductor rings have been investigated.

The only means for insuring an absolute seal is through a bonded connection, using a weld or braze. However, to date, all attempts at bonding have resulted in distortion of the foil where it extends beyond (outside) the conductor ring components. In addition, to reduce the bonding temperature, nickel filler must be used for this joint. Since nickel could be a contaminant to the lead telluride, the module performance could be affected.

During the past quarter an improved barrier seal module, TEM-15B S/N-4D, was identified as a proof-of-principle experiment to test the existing

Figure 43. A Comparison of Efficiency Data from TEM-15B, Tungsten Foil Barrier Modules

degradation mechanism theory. Although the module will be fabricated without a bonded joint seal, isolation of adjacent washers will be obtained by fabricating the inner conductors as two separate rings and extending the tungsten foil radially inward to the inner boron nitride insulator. Figure 44 presents a cross-section schematic of the couple design to be incorporated in TEM-15B S/N-4D.

Fabrication and testing of TEM-15B S/N-4D will occur during the next reporting period.

c. Additional Modules

In addition to the tungsten foil barrier and nonbarrier modules, four one-of-a-kind modules were tested continuously during the past quarter. These are the last four modules listed in Table 21. TEM-15C S/N-4A, TEM-15C S/N-8, TEM-15D S/N-4, and TEM-15H S/N-4, whose individual design features are listed in Table 20, 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 electron beam vapor deposited barriers, tungsten n-washer contacts, and increased mica washer thicknesses. Power output and overall efficiency data from these modules are presented in Figures 45 and 46, respectively.

Of particular interest in this group of modules is the fact that TEM-15C S/N-4A, a tungsten EBVD barrier module, continues to exhibit performance stability as low as has been observed for tungsten foil barrier modules (see Table 21). This is of significance due to the substantially improved performance associated with elimination of the tungsten foil barriers and their associated shunt heat losses. TEM-15C S/N-4A has a beginning-of-life efficiency level 18% higher than TEM-15B S/N-4C and 37% higher than TEM-15B S/N-4B.

Also shown in Figures 45 and 46 are data from TEM-15A S/N-4G, a low temperature nonbarrier module, discussed above. Notice that the power output and efficiency levels of this module are higher than all previously tested modules despite its being operated at lower temperatures and at a reduced radial temperature drop. Although the Carnot efficiency for 1025° F/480° F is 1% higher than for 1125/570 operation, a measured overall efficiency improvement of 5% was observed. This phenomenon is predicted by the performance code

Figure 44. TEM-15B S/N-4D Couple Design

Figure 45. A Comparison of Power Output Data from Improved Performance Stability TEM-15 Series Modules

Figure 46. A Comparison of Overall Efficiency Data from Improved Performance Stability
TEM-15 Series Modules

and results from the fact that the lower operating temperatures span a more favorable band of the lead telluride figure-of-merit curves. Power and efficiency degradation rates recorded for this nonbarrier module can be seen in Figures 45 and 46 as well as Table 21, to be substantially lower at the lower temperature levels, as expected.

2. Glass Seal Development

A series of experiments was performed during the last quarter in order to estimate glass seal capability during operation in air. Life test operation of the glass seal has been carried out at 500, 540, 570, and 600°C. Failure of the glass was defined as the point where the seal was no longer helium leak tight.

The failure mechanism proposed to explain the results is essentially one caused by devitrification of the glass. This process, where amorphous glass (V_1) transforms to crystalline solid (V_2), apparently causes a volume change ($V_2 < V_1$), pinholes, and the resulting inability to provide a helium leak tight seal. A model of the process has been made by assuming the crystallization follows first order kinetics. If we let n equal the fraction of glass crystallized,

$$\frac{dn}{dt} = A e^{-Q/RT}.$$

The time to failure, t_f , is assumed to occur when n equals unity, or some critical fraction n_c . In either case,

$$t_f = A' e^{Q/RT}$$

where $A' = 1/A$ or m/A .

Hence, a plot of the log of failure time versus $1/T$ should allow an estimate of seal life at any specific operating conditions. Such a curve is shown in Figure 47 where the data for the four previously discussed temperatures are plotted. The data were obtained on six eight pin mockups tested at temperatures $\geq 500^\circ\text{C}$, and one single pin mockup at 500°C . The time spread indicates the interval over which the seal failed, i. e., went from a helium leak-tight

Figure 47. Helium Leak Tight Hours vs $1/T$

condition to a state which permitted the passage of helium. The data allow a reasonable straight line curve fit over the temperature range studied. The longer time data range at 500° C corresponds to the failure of three eight-pin mockups and one single-pin mockup over virtually an identical time interval. The lower temperature data correspond to a possible helium leak indication of a single pin on an eight-pin mockup after 2373 hours. However, it was not detected after the next check at 2642 hours.

Also shown on the plot is a point labeled goal. This point represented the 5-year desired life of a glass seal during air operation at the maximum operating temperature at the pin seal bolt circle (420° C) for a module operating at $\bar{T}_H = 585^\circ \text{C}$ (1085° F). It is apparent that extrapolation of the data to the lower operating temperature assures a more than adequate operating time, by a factor of ten, for these glass seals. In addition, recent measurements of temperature on a module whose $\bar{T}_H = 635^\circ \text{C}$ (1175° F), show a temperature of 387° C at the pin seal. Hence, regardless of how conservatively the data are handled, the prognosis for successful thermal capability of the glass seal appears good.

Four additional mockups are on test at 500° C, three of which have demonstrated more than 3000 hours of helium leak tight capability.

A series of experiments was carried out on various mockups and modules to completely define the processes and parameters necessary for the successful application of glass-to-metal seals at each pin base of TEM-X S/N-3 modules. Using the procedures defined by the above tests, three TEM-X S/N-3 modules had glass seals applied prior to being placed on static test. In addition proper RF coil modifications have been defined and carried out so that seals can be applied to shrouded TEM-X S/N-3 modules destined for loop test operation.

REFERENCES

1. "SNAP Reactor Programs Progress Report, November 1971-January 1972," AI-AEC-13023, April 15, 1972
2. "SNAP Reactor Programs Progress Report, August-October 1971," AI-AEC-13017, December 23, 1971