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PROGRESS REPORT

SNAP REACTOR IMPROVEMENT PROGRAM

JULY-SEPTEMBER 1965

(Title Unclassified)

AEC Research and Development Report

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SNAP REACTOR IMPROVEMENT PROGRAM
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L. M. MAKI

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I. PROGRAM OBJECTIVES

The primary objective of the SNAP Reactor Improvement Program is to increase the performance capabilities, power density, and temperature of uranium-zirconium hydride reactors.

Engineering analyses will be performed to establish the degree to which increases in reactor performance can be made by extrapolating current reactor designs closer to the limits of existing reactor technology. The development required to achieve these increases in performance will be established.

Parameter studies will be performed to determine the limits of performance of the present SNAP reactors and the gains to be achieved by changes in configuration or extension of the limits of technology. Key problem areas which

limit the performance of the present reactors will be identified and potential increases in performance will be evaluated in terms of increased power, increased operating temperature, increased lifetime, and reduced system weight.

The fuel development program will be directed towards the optimization of the present ceramic barrier nonrefractory metal clad fuel element design. Improved barriers, closures, and clad materials will be studied in order to improve fuel element performance. In addition to the development work, irradiations will be performed to determine the behavior of the fuel and cladding materials.

The Interim SNAP 10A/2 DRM-1 test program will be brought to an orderly termination.*

*Several changes were made in the program guidance at the end of the reporting period. The effects of these changes will be reflected in the next quarterly progress report.

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II. PROGRESS SUMMARY

A. REACTOR ENGINEERING

Capabilities studies were performed as part of a continuing effort to determine the operating limits of uranium-zirconium hydride reactors.

The use of an yttrium getter to reduce the hydrogen loss in static-controlled reactors was investigated and found to be beneficial if the uncertainty in the hydrogen loss rate is large. If, on the other hand, the hydrogen loss rate is predictable with a small uncertainty, the use of a getter actually increases the end-of-life temperature uncertainty.

Studies were also performed to evaluate the materials and component limits for high performance systems. The results of these studies will be used in future analyses aimed at increasing the design margin of SNAP reactors.

B. FUEL DEVELOPMENT

The NAA-82-1 SNAP 2 fuel element irradiation experiment successfully completed its scheduled irradiation and is awaiting hot-cell examination.

The operating parameters for the NAA-83-1, -2, -3 in-pile creep experiments were established. The three S8DS fuel element cladding tubes will be operated at 1400°F for 4000 hours with stresses of 2500, 4000, and 8000 psi, respectively. A total fast-neutron flux exposure of 1.5×10^{21} nvt is expected.

The design of the NAA-120-3, -4 fuel material irradiation experiments has been completed, mockup assemblies have been built, and final fabrication has been initiated.

The SNAP 10A fuel element environmental testing program was terminated during the quarter. None of the 120 fuel elements which underwent endurance testing showed evidence of significant damage.

C. SNAP 2 DRM-1 TEST

The SNAP 2 DRM-1 test program was brought to an orderly termination. The reactor assembly and the associated test equipment were disassembled and placed in storage. The vacuum chambers were "mothballed" and preparation of a final report was initiated.

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III. REACTOR ENGINEERING

A. DESIGN INTEGRATION

A study plan is in preparation for the analysis of reactor and shield parameters relevant to the design of optimized nuclear systems. The parametric design analyses to be performed in this program will be conducted in accordance with this plan.

B. STRUCTURAL ANALYSIS

During the past quarter, the structural analysis of the S2DRM-1 NaK heater was completed and the data from the S2DRM-1 vibration test was analyzed. A preliminary materials application survey and a launch load criteria study for advanced reactor concepts were undertaken.

The structural analysis of the S2DRM-1 NaK heater revealed two critical areas. One was the weld joint between the outer support skirt and the containment vessel. The stresses in this area were reduced by modifying the insulation to minimize the thermal gradients. The other critical area was the line connecting the heater vessel and the expansion tank. This problem was resolved by lengthening the line to accommodate the relative thermal expansions.

The analysis of the S2DRM-1 vibration test data showed the following principal mode characteristics. The transmissibilities shown are between the table input and the reactor top head.

<u>Direction</u>	<u>Transmissibility</u>	<u>Frequency</u>
x-x (Vertical)	8	240
y-y (Lateral)	12	50
z-z (Lateral)	21	70

In general, the reflectors followed the lateral vessel vibrations up to about 200 cps, above that frequency some relative motion is evident. Vertically, the reflectors moved relative to the

vessel at about 115 cps. The accelerometer locations during the vibration test are shown in Figure 1.

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Figure 1 Vibration Instrumentation Locations

C. NUCLEAR ANALYSIS

1. Yttrium Getter Studies

The range of applicability of in-core yttrium getters in static-controlled reactors has undergone extensive investigation. The purpose of the getter is to combine chemically with the hydrogen leaking from the fuel elements, and thereby to effectively eliminate hydrogen leakage as a reactivity loss mode. There is no net reactivity advantage to the use of yttrium since,

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by its presence, it displaces fuel having more reactivity worth than the worth of the hydrogen it can stoichiometrically absorb. The advantage obtained is the reduction or elimination of the end-of-life temperature uncertainty due to uncertainty in hydrogen leakage rate.

Hydrogen leakage, however, exerts a strong temperature feedback upon the temperature uncertainties caused by other effects, such as uncertainties in prepoison loadings and calculations of other reactivity losses. This is due to the strong increase in hydrogen leakage rate with temperature. If reactivity losses under static control are greater than expected, the reactor temperature will tend to be lower than predicted. This lower temperature, in turn, causes the hydrogen leakage rate to be lower and therefore the decrease in temperature will be less than would occur were no hydrogen leakage feedback present.

The presence of an yttrium getter therefore increases that part of temperature uncertainty due to other effects at the same time that it is reducing the hydrogen leakage rate contribution towards temperature uncertainty. The success of yttrium in reducing temperature uncertainties depends on the relative magnitude of the hydrogen leakage contribution vs other contributions. If hydrogen leakage uncertainty is very small, the use of the getter will increase temperature uncertainties. As the assumed uncertainty in hydrogen leakage is increased, a point will be reached where the hydrogen leakage uncertainty contribution is large enough for its elimination to offset the increases in temperature uncertainty due to other effects. Above this point the presence of a getter will decrease temperature uncertainties.

The crossover value is dependent on both power and temperature. In general, it increases with decreasing temperature due to decrease of the relative importance of hydrogen leakage

rate uncertainty at lower temperatures. The crossover value increases with increasing power, since the uncertainties due to effects other than hydrogen leakage increase with power in the absence of leakage feedback, while temperature uncertainties are virtually independent of power level if hydrogen leakage feedback is taken into account.

Figure 2 shows the value of this crossover point as a function of reactor power and temperature for the SNAP 8 reactor. For low power levels (~100 kw), the crossover occurs at a leak rate uncertainty of 40 to 45 percent. For levels in excess of 250 kw, the crossover value is of the order of 100% uncertainty. The

Figure 2. Crossover Values of Hydrogen Leak Rate Uncertainty

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values quoted here are percentages of the predicted value. An uncertainty of 40%, for example, indicates that the leak rate is less than 140% of the predicted value. Results are almost independent of the magnitude of the predicted rate.

2. Fuel-Cladding Thermal Bond Studies

Use of a thermal bond between SNAP reactor fuel and cladding will eliminate or substantially reduce the fuel element gas gap temperature drop. The resultant lower fuel temperatures will lead to lower hydrogen leakage (increasing reactivity and phase-change lives) and increased fuel swelling life.

A series of studies were performed using the S8DS reactor as a model. The fuel swelling life was defined as that time when the fuel touched the cladding in any part of the core. Phase-change life was defined as being reached when fuel in any part of the reactor reached the boundary between the β and the δ phase. Hydrogen leakages to these lifetimes were noted. Although different design criteria will probably be used in future SNAP reactors, it is expected that the results will hold semiquantitatively.

Three effects of the thermal bond on performance were considered, based on the assumption that the use of a bond would reduce the thermal resistance across the gap to 10% of the nominal value.

- 1) Increase in power capabilities at 1300°F coolant outlet temperature for one year:

<u>Thermal Resistance</u>	<u>Max. Power Based on Fuel Swelling</u>	<u>Max. Power Based on Phase Change</u>
Nominal	1020 kw (\$1.90)	1000 kw (\$1.85)
10% of Nominal	1250 kw (\$1.55/2.35)	1400 kw (\$1.65/2.65)

The reactivity figures noted for the nominal cases are leakages at the corresponding power levels for one year. For the reduced resistance cases, the actual leakages are shown at the corresponding power together with the leakages which would occur at the same power level with no thermal bond.

An increase in power capability from 1000 kw to 1250 kw, or about 25%, is obtained by the use of a thermal bond.

- 2) Increase in outlet temperature capabilities at 600 kw for one year:

<u>Thermal Resistance</u>	<u>Max. Outlet Temperature Based on Fuel Swelling</u>	<u>Max. Outlet Temperature Based on Phase Change</u>
Nominal	>1600°F (\$9.50)	1420°F (\$3.50)
10% of Nominal	>1600°F (\$8.35/\$9.50)	1530°F (\$5.50/\$6.60)

An increase in outlet temperature capabilities from 1420°F to 1530°F, or about 100°F, is obtained with the aid of a thermal bond. Limits here are imposed by phase change considerations. Reductions in hydrogen leakage are not significant since at these temperatures the rapid initial loss of hydrogen reduces hydrogen pressures to levels where further leakage is not significant.

- 3) Increase in lifetime capabilities at 1300°F outlet and 600 kw.

<u>Thermal Resistance</u>	<u>Fuel Swelling Life</u>	<u>Phase Change Life</u>
Nominal	30 months (\$2.75)	53 months (\$4.17)
10% of Nominal	38 months (\$2.64/\$3.28)	>5 years (\$3.72/\$4.55)

An increase in lifetime capabilities from 30 to 38 months, or about 25%, is obtained with the aid of the thermal bond.

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In instances where fuel swelling is limiting, further increases in capabilities may be obtained by increasing the gap size. Increasing gap size with no thermal bond leads to only marginal change in capabilities since the resultant higher gap resistance leads to increased fuel temperatures. With a thermal bond, however, no significant increase in fuel temperature is obtained.

In the increased power core, a change in the gap size to 4.0 radial mils will enable the reactor to operate to its phase change limit of 1400 kw. In the increased lifetime case, the life may be increased to its phase-change-limited value of about five years if the gap is increased to 7.5 radial mils. The reactivity penalty associated with increasing the gap is about \$0.50 per radial mil.

It appears that significant energy release capabilities can be realized by use of a thermal bond in conjunction with increasing the radial gap. Such a bond would pose fabrication problems, however. A mechanical bond is not feasible since the coefficient of expansion of the cladding is much greater than that of the fuel. In addition, room must be made available for fuel swelling. If a material bond is to be used, it must be compatible with the fuel and barrier materials and possess a low neutron absorption cross section.

D. THERMAL ANALYSIS

1. Material Limitations of Uranium-Zirconium Hydride Reactors

A preliminary study of the nature of uranium-zirconium hydride reactor performance limitations has been completed. Fuel, barrier, cladding, reflector and structural material limits were investigated and design limit criteria revised. The most serious limitations are fuel

swelling, fuel phase-change and, at elevated temperatures, cladding degradation. Based on these limitations, a cursory examination was made of maximum reactor capabilities with regard to power density, coolant outlet temperature, and design lifetime.

a. Fuel Irradiation Swelling

Experimental data have shown that the density of δ -phase zirconium-hydride fuel decreases during irradiation. This decrease in density, or fractional increase in volume, appears to be a function of the percentage of metal atoms fissioned, the temperature of the sample during irradiation, and possibly the irradiation time. Based on limited data in the 1200 to 1500°F temperature range, several correlations have been proposed; these vary by as much as $\pm 50\%$, which is the approximate uncertainty in any one equation. Even larger discrepancies exist when the correlations are extrapolated above 1500°F.

The most commonly accepted correlation is that of Miller,* who expressed swelling of modified and unmodified Zr + 10 U hydride fuel ($N_H = 6.5$) due to irradiation by the following equation:

$$\frac{\Delta V}{V_{\text{initial}}} = (1.2 \times 10^6) b^2 \exp(-22600/T_{\text{max}}) + 1.0 b$$

where b , the burnup, and T_{max} , the averaged peak fuel temperature, are in units of met. at. % and °R, respectively. The second term attempts to take into consideration volume increase due to the increase in the number of atoms in the fuel. The above equation is an empirical fit to the available data while other proposed equations are based on theoretical models and use only limited portions of the data.

Additional irradiation tests are now underway to investigate fuel swelling and the related problems of cladding creep and hydrogen loss from

*J. R. Miller and D. A. Polak, "Effects of Irradiation on Hydrided Zirconium-Uranium Alloy NAA-53 Experiments," NAA-SR-9633, December 30, 1964

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fuel. The techniques to be employed in measuring volume changes and hydrogen loss are greatly improved so that these tests should define fuel swelling between 1200 and 1500°F to within $\pm 20\%$.

Tests have been made to estimate the effect of fuel swelling on the barrier hydrogen leak rate. Fuel swelling of SNAP 8 elements ($d_0 = 0.56$ in.) was simulated out-of-pile by hydriding zirconium metal which provided an increase in volume as the hydrogen concentration was increased. The test showed that the SCB coating could be strained 4 mils on the diameter with no appreciable increase in hydrogen leak rate. Thus it appears reasonable to set 5 mils as a tentative fuel-barrier diametric interference limit.

b. Fuel Phase Change

The design specification limit excluding any change of phase is based on the uncertainty of the fuel's dimensional stability when a phase change occurs. At the start of reactor operation, all of the fuel will contain sufficient hydrogen to be in the δ -phase. As hydrogen permeates through the ceramic barrier, the hydrogen content of a fuel rod will decrease until the local crystal structure starts to transform from the face-centered-cubic δ -phase to the body-centered-cubic β -phase. During the rearrangement of the fuel structure, the mobility of the inert fission-product gases (xenon and krypton) greatly increases. According to accepted theory, this mobility allows the inert gases to form into larger gas pockets, thereby essentially eliminating the surface tension effect which withstands the expanding tendency of small fission gas pockets. The fuel then swells an unknown amount which is believed dependent on the integrated burnup and the operating temperature. Phase-change swelling superimposed on normal irradiation swelling could result in cladding rupture and loss of hydrogen.

There is no quantitative evidence as to how much the SNAP fuel would swell during the δ - β phase transformation. Only one preliminary set of phase-change tests has been run on SNAP fuel. It was found that occurrence of phase change can cause a burst of released gas. A single transformation did not cause complete gas release and successive transformations produced additional release of fission gas. However, anomalous results were obtained with some samples. In no case was the amount of swelling determined, nor was the amount of fission gas released correlated with the amount remaining in the sample.

Examination of fuel elements from the S8ER reactor core may provide some experimental information. Obviously, any data relating phase-change swelling to burnup and operating temperature would be of great value. Until data is available, it is necessary to set the first appearance of the β -phase as a design limit.

c. Cladding Degradation

Nickel-based alloys (such as Hastelloy-N and Rene 41) should be limited in use to outlet coolant temperatures below 1400°F. Above 1400°F, accelerated nickel mass transfer occurs which could lead to serious material depletion and crud accumulation in the coolant loop. Type-316 stainless steel suffers reduced creep strength at elevated temperatures, but could be used to approximately 1450°F, provided the differential pressure stresses were low.

d. Maximum Reactor Capabilities

Upper performance bounds for U-ZrH reactors have been calculated. A number of assumptions were made with regard to fuel element geometry and initial composition in order to simplify the analysis. The assumptions are listed below and are intended, as far as possible,

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to maximize the limiting power density at a given outlet temperature and lifetime.

Fuel Element: $d_f = 0.450$ in.

$\Delta r_{\text{gas}} = 0.003$ in.

$\Delta r_{\text{clad}} = 0.010$ in.

$\Delta r_{\text{barrier}} = 0.002$ in.

SCB barrier material

Initial $N_H = 6.5$ (H/Zr = 1.85)

Core: 211 elements

$d_{\text{core}} = 7.5$ in.

$L_{\text{core}} = 12$ in.

Coolant $\Delta T = 200^\circ\text{F}$

Figure 3 summarizes reactor performance capabilities based on the three most significant limitations. The use of nickel-based alloys for cladding limits reactor operation to outlet tem-

peratures below 1400°F ; with stainless-steel cladding it may be possible to extend operation to 1450°F . Between approximately 1300°F and the cladding limit, phase change is limiting. Below 1300°F , fuel swelling is limiting. As anticipated, the lifetime requirement strongly affects the power density that can be reached at a given temperature.

The calculated capabilities are based on an actively controlled, nonreactivity-limited reactor. Constraints not considered include static control, reactivity limitations, coolant pressure drop, shielding limitations, and reliability effects.

2. Thermal Conductance of Low-Pressure Gas Gap

In calculating the fuel temperature of a SNAP fuel element, a significant fraction of the thermal resistance between the fuel and the coolant is represented by the annular gas gap between the fuel and the barrier. In a fuel element near the beginning of life and with a radial gas gap of more than 0.001 inch, the gap may be treated as being filled with a continuum of hydrogen whose thermal conductivity is essentially independent of pressure. However, this approach can lead to overestimating the conductance of the gas gap after prolonged power operation when fuel swelling may have reduced the gap width and loss of hydrogen has reduced the gas density in the gap. As the mean free path of the gas increases to approach the order of magnitude of the gas gap, heat transfer changes from the continuum mode to the so-called free-molecular mode. In this latter mode, energy is transferred by direct collision between the gas molecules and the bounding surfaces without intervening intermolecular collisions. Figure 4 illustrates this effect; it shows the conductance of a hydrogen gap as a function of gap width for a gas temperature of 1240°F and a pressure of 68 torr. This state is typical of a SNAP 8ER fuel element at the end

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Figure 3. Reactor Design Capabilities

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of approximately one year of operation. When the gap is closed, the interface conductance becomes a function of contact pressure between the surfaces. This effect is also shown in Figure 4.

Figure 4. Equivalent Gas-Contact Heat Transfer Coefficient vs Radial H_2 Gap and Contact Pressure

E. COMPONENT PERFORMANCE ANALYSIS

1. High-Temperature Bearings

From the basic studies conducted thus far on bearing materials for SNAP reactor applications, an "ideal" friction couple has been established. The material couple is Al_2O_3 against carbon-graphite. The Al_2O_3 is "flame sprayed" onto a metal surface by a detonation gun process and then finish-ground to 8 rms or better surface finish. The carbon-graphite material has been optimized at 40% carbon and 60% graphite.* The surface texture and compressive strength of the graphite material has been improved by addition of metallic salt impregnants.

In a working bearing design the impregnated carbon-graphite material is used as an insert bushing, which mates against an Al_2O_3 flame-sprayed metallic shaft. These journal bearings

are capable of operating at high temperature, in space vacuum, at high radiation levels without added lubrication or maintenance.

Satisfactory operation of these materials in the journal bearing configuration has been demonstrated through development, acceptance, and qualification testing of SNAP 10A, 2, and 8 reactor components. During these studies environmental temperatures have increased from 600°F in SNAP 10A to 1150°F in SNAP 8. These data for control drum bearings are illustrated in Figure 5 as points 1, 2, 3, and 4. Points 5 and 6 represent higher temperature and longer time evaluations but these data are presently incomplete.

Analytical evaluations have been made to predict bearing life as a result of increasing the temperature requirements. Although actual testing has not been performed it appears that the basic bearing couple, Al_2O_3 against carbon-graphite, would be satisfactory to temperatures as high as 2000°F. The substrate material, against which the Al_2O_3 would be plated, would necessarily be a refractory metal or alloy, as would the housing which contains the carbon-graphite bushing. On the other hand, the thermal coefficient of expansion of the refractory metals more closely approximates those of the bearing materials and would, in this sense, represent an improved condition over lower temperature designs.

It should be pointed out that the carbon-graphite materials oxidize rapidly in air or oxygen-rich environments at temperatures above 1000°F, but in vacuums below 10^{-5} torr, oxidation is insignificant. Running seals or molecular restrictions can be designed into the bearing, however, to reduce these effects even at higher pressures.

*L. G. Kellogg, "Snap Reactor Materials Development, Ultrahigh-Vacuum High-Temperature Friction Studies," NAA-SR-9644 (Declassified), June, 1964

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Figure 5. Reactor Control-Drum Bearing Operating Capabilities

All other factors being considered, it would seem that these materials have an apparent capability of successful operation at temperatures near 2000°F for periods of at least 20,000 hr. This point is shown in Figure 5 as Point B.

2. High-Temperature Actuator

The purpose of the actuator in the reactor control loop is to position the control drum reflectors which, in turn, control the reactivity of the reactor. The actuators are electromagnetic stepper motors with solenoid-operated brakes. Several different actuator types have been studied for use on SNAP systems. Each new design represents improvements in the concepts of the previous design. These improvements resulted from development testing, reliability studies, and continuous design review. Figure 6 illustrates the evolution of the actuator development starting with the SNAP 10A design.

The environmental temperature created by conduction and radiation heating from the reactor, plus internal neutron and gamma heating, is approximately 650°F for the SNAP 10A actuator. The succeeding SNAP 2 and 8 reactors have higher power levels, larger reflector assemblies, and higher environmental temperatures. To meet the higher temperature and torque requirements of SNAP 2 and 8 systems, improvements in the 10A actuator were required.

The SNAP 8 Development System (S8DS) actuator is the latest model and incorporates the improved design features gained from previous developments. These improvements are discussed in detail in another report,* but they may be summarized as follows: all the structural parts, including the rotor, are fabricated from the same material (Hiperco 27), thus reducing differential thermal expansion and providing

*S. Giles and A. A. Marcus, "SNAP 8 Control Drum Actuators," NAA-SR-9645 (Uncl.), December 15, 1964

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Figure 6. Actuator Development

more "working" iron. The rotor diameter has been increased, thereby increasing output torque. The rotor and stator tooth alignment is set to provide maximum torque for each tooth at the same instant, rather than in sequential steps, as in the 10A and 2 designs. A chip guard is placed between the encapsulated coils to prevent chipped insulation from collecting between the rotor and stator teeth. The brake surfaces are also Hiperco-27 plates with teeth machined to match during engagement, and the mating surfaces are coated with dry-film lubricant to inhibit self-welding.

The major limiting factors in terms of operating temperature for the S8DS actuator, are the curie point of the iron and self-destruction of the conducting wire. These limits are shown in Figure 7. The conductor material, chrome-plated copper wire, has a melting point of about 1900°F, but can fail due to rapid grain growth at 1500°F. Resultant internal heating becomes a problem at these elevated temperatures which also serve to shorten life.

As designed, the existing actuator could be operated for indefinite periods at 1000°F in vacuum; with slight modifications, the same

unit could be operated at 1400°F for 10,000 hours. These modifications would be replacement of the copper wire with a molybdenum or tungsten wire and change in the dimensional fit of all mating assemblies, including the bearings, to allow for the increased expansion. A cursory analysis of these changes indicates that at temperatures below 1400°F the modified actuator could be operated for 20,000 hours. However, testing would be required to establish material creep and relaxation data before assuming the 20,000-hour capability.

3. Radiation and Temperature Effects on Beryllium Swelling

A project has been initiated to evaluate the upper limits of application of beryllium in a high-temperature, high-dose reactor environment. Study effort will be confined to review and analysis of published and accessible unpublished data. Progress in this quarter has included a review of available information sources and informal contact with active development groups. Preliminary data obtained from Beeston of Phillips Petroleum indicate that swelling due to helium agglomeration may not be serious below 1400°F at doses in excess of 10^{21} nvt. A

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Figure 7. Actuator Operating Capabilities

complete compilation of data preparatory to analysis is planned for the next quarterly period.

4. Shield Degradation from Meteoroid Puncture

The possibility of meteoroid puncture of the shield casing must be considered in the design of lithium hydride fast neutron shields for use in space. A puncture of the shield casing causes degradation of the neutron attenuation properties of the shield, primarily because of hydrogen loss. The present design philosophy for SNAP shields essentially considers a meteoroid puncture of the casing a failure of the SNAP reactor system. The shield casing thickness is determined by a probability of no meteoroid penetration, which, in turn, forms part of the total system reliability. It has been found that SNAP

shields designed by this method are overconservative, resulting in shield casing thicknesses greater than necessary and adding to the system weight.

An analysis has been made to determine the increase in dose due to shield case meteoroid puncture as a function of mission duration, exposed area, lithium hydride temperature, case thickness, and several other factors. Figure 8 shows the increase in dose at the end of 10,000 hours for the S8DS shield for various shield casing thicknesses, with the maximum lithium hydride temperature as a parameter. For comparison, the NASA 417-5 specification would require an armor thickness of 0.259 in. for a

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probability of nonpuncture of 0.999. This analysis has been carried out using the same meteoroid penetration equation and meteoroid cumulative mass distribution as used in the NASA 417-5 specification, but includes the time-dependent effects of hydrogen loss from the shield as a

function of temperature and puncture size. Armor thickness for future shield designs should be based on the probability that the neutron dose at the payload will not increase more than a given amount.

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Figure 8. Effect of Meteoroid Punctures on Shield Performance

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IV. FUEL DEVELOPMENT

A. NAA-82-1 IRRADIATION EXPERIMENT

1. Summary

The NAA-82-1 irradiation experiment was designed to evaluate full-scale-production SNAP 2-type fuel element performance at 125-kwt reactor conditions of temperature and burn-up. These experiments have been described in detail in previous quarterly reports.

2. Status

NAA 82-1 successfully completed its scheduled 23 reactor cycles of irradiation in the MTR† on August 16, 1965 with all of its 16 thermocouples operating satisfactorily. This experiment achieved its design objective of approximately 0.08 metal atom per cent burnup at a calculated peak temperature of 1510°F. The experiment is currently located at the AI hot laboratories pending post-irradiation examination.

B. NAA-83-1, -2, AND -3 IRRADIATION EXPERIMENTS

1. Summary

The NAA-83 irradiation experiments are composed of three separate nonfueled capsules which are designed primarily to determine the effect of irradiation on the creep behavior of the S8DS fuel cladding material, Hastelloy-N. A secondary objective of this irradiation series is to obtain in-pile hydrogen permeation data on S8DS membranes at significantly higher pressures than were studied in previous experiments. These goals will be obtained by internally pressurizing the membranes with hydrogen to obtain three different hoop stresses in the cladding material while keeping the temperature constant. A dynamic vacuum annulus

surrounds each individual membrane and the hydrogen permeation rates will be determined by analyzing the gases drawn from this annulus. Heating of the membrane to the predetermined temperature will be accomplished by utilizing gamma heat and a variable helium-nitrogen gas annulus for heat transfer control.

2. Status

An isometric cutaway view of an individual capsule is shown in Figure 9. The capsules include a Hastelloy-N cladding membrane sub-assembly, a Hastelloy-N gamma heater tube, and an AISI Type-304-stainless-steel outer casing, with the only heat source being the gamma heating. By introducing hydrogen into the membrane at selected pressures, the stress in the cladding subassembly in NAA 83-1 will be maintained at 2500 psi, that in NAA 83-2 at 4000 psi, and that in NAA 83-3 at 8000 psi. The NAA-83-1 and NAA-83-2 capsules will be irradiated for 4000 hours at temperature and should achieve an integrated fast neutron flux exposure of approximately 1.5×10^{21} nvt. The NAA-83-3 experiment will probably fail during its irradiation, thus yielding vital in-pile creep ductility and creep-to-rupture data. The operating parameters for these capsules are summarized in Table 1. The factors contributing to the choice of the above values are summarized as follows

a) Hoop stress — The maximum internal pressure buildup due to a hot-spot temperature and subsequent hydrogen dissociation was calculated for S8DS conditions to cause a hoop stress of approximately 1500 psi, assuming no external restraint. Since the deformation resulting from this stress would be

¹Principal design features were reported in "SNAP 2 Reactor Quarterly Progress Report, July-September 1963," NAA-SR-8990 (SRD), November 15, 1963

†MTR, Materials Testing Reactor

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Figure 9. Design of NAA-83 Irradiation Test Capsules

TABLE 1
OPERATING PARAMETERS FOR NAA-83 EXPERIMENT

Capsule Number	Hoop Stress (psi)	Membrane Irradiation		Exposure (nvt) ($E > 1 \text{ Mev}$)	Creep Rate (%/hr)	Expected Deformation	
		Temperature ($^{\circ}\text{F}$)	Time (hr)			(%)	(in.)
NAA-83-1	2500	1400	4000	1.5×10^{21}	1×10^{-5}	0.04	0.0002
NAA-83-2	4000	1400	4000	1.5×10^{21}	5×10^{-5}	0.20	0.0011
NAA-83-3	8000	1400	?(-1000)	?(- 0.4×10^{21})	6×10^{-4}	0.60	0.0033

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very difficult to measure accurately for any reasonable test time, this stress was nearly doubled for the lowest stress level in this experiment. Also, some creep data have already been obtained at a hoop stress of ~1200 psi on the NAA-77-1 and -3 capsules and these data provide information at this lower stress. Much of the out-of-pile creep data which have been generated at AI on Hastelloy-N have been obtained at stresses that will be used in this experiment.

b) Temperature — 1400°F was selected because this was approximately the S8ER maximum design temperature and much of the out-of-pile creep data were obtained at this temperature. The S8DS maximum design temperature is approximately 1450°F, it is believed that very little will be sacrificed by operating at the slightly lower temperature.

c) Time — The 4000-hour test time was selected so that a measurable amount of deformation or strain could be obtained at the lower stress levels. The NAA-83-3 capsule is designed to rupture at some time during its exposure life. At other laboratories, some data have been generated which showed a decrease in creep rupture life for Hastelloy-N by a factor varying from 4 to 10 at an exposure of $\sim 10^{20}$ nvt. Therefore, a strain to rupture of 0.6% was assumed for this material. The value actually may be low since the irradiation damage is constantly increasing, the total strain to rupture may be closer to 1%. One of the most significant data to be obtained from this experiment is determination of the strain required to cause failure when the stress is held constant and the irradiation damage is constantly increasing.

d) Neutron exposure — A neutron exposure of 1.5×10^{21} nvt ($E > 1$ Mev) is considered to be a reasonable estimate for an S8DS fuel element at the end of its operating life.

e) Creep rate — These values were obtained from a combination of AI's data on Hastelloy-N and Oak Ridge's data on INOR-8 (material equivalent to Hastelloy-N with the exception of slightly higher iron and silicon content resulting in slightly inferior welding properties), both of which were out-of-pile data. Preliminary data from Oak Ridge indicate that we could expect nearly the same creep rate in-pile and out-of-pile, holding all other parameters constant, the basic effect of irradiation being a reduction of the strain required to cause fracture.

f) Expected deformation — Probably the limiting factor which will govern the reliability of the data from this experiment is the accuracy of the hot cell measurements on the diameter of the membrane. Presently the capability of hot-cell instrumentation is ± 0.0003 in., using the existing profilometer. An improved profilometer which will have an accuracy of ± 0.0001 should be installed within a few months, this instrument will provide accuracy adequate for reliable creep data measurements provided interference with oxide surface coatings is not encountered.

The original NAA-83 experiments which were designed under the SNAP 2 program utilized a gamma heating rate of 12 watts/gram, a 50% He/N₂ gas mixture in the gas control annulus, and a membrane temperature of 1400°F to determine the gamma heater and gas annulus sizes. Four capsules were built utilizing out-of-house chromizing techniques, they were designed for 2,000 hours (4 ETR* cycles) testing

*ETR, Engineering Test Reactor

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at operating temperatures of 1400°F and hoop stresses of 2,500, 4,000, 8,000 and 12,000 psi. Since that time, out-of-pile creep tests have been conducted under the SNAP 8 program. These tests indicated that the above stress level of 12,000 psi was probably too high and it was therefore eliminated. This, plus economic considerations, reduced the number of experiments to three, the minimum number necessary to plot a stress-vs-creep rate graph. Therefore, to insure a more accurate measurement of the creep rate for the 2500-psi experiment the irradiation time was increased from 2000 to 4000 hours. In order to meet the 4000-hour total irradiation time and an accumulated fast neutron exposure of 1.5×10^{21} nvt, approximately 8 ETR cycles would be required at a fast-flux level of 1.0×10^{14} n/cm²-sec. A plot of the accumulated fast-neutron exposure vs reactor residence time is given in Figure 10. Since the neutron flux is

now a factor of two lower and the irradiation capsules have to be rebuilt in order to take advantage of current fabrication practices, a different ETR irradiation facility with a lower thermal neutron flux and correspondingly lower gamma heating rate would be economically desirable. Negotiations are presently underway with ETR personnel for a reactor position which corresponds to the new time and flux requirements. After the irradiation facility has been established and the gamma heating fluxes mapped, the capsule gamma heater and control gas annulus size will be determined. It is planned that all three capsules will be inserted simultaneously so that the reactor condition will be as nearly identical as possible for all three capsules. The capsules are scheduled for insertion in the ETR in the first quarter of Fiscal Year 1967.

In conjunction with this experiment, out-of-pile creep tests under the SNAP 8 program have been conducted on nearly equivalent membranes under various environmental conditions; none of the membranes, however, were exactly equivalent to those to be used in S8DS. A program is now being conducted to determine the effect upon creep behavior of using NaK as an external environment of the S8DS-type membranes. The tests (which are to be identical to the NAA-83 experiments with respect to temperature and hoop stresses) are to be performed in four NaK thermal convection loops; these have been constructed (see Section IV-E of this report). Ultimately the effect of irradiation and NaK environments as separate entities will be determined and these results will be used to estimate the effect of the combination of the two parameters.

C. NAA-120-3 AND -4 IRRADIATION EXPERIMENTS

1. Summary

The NAA-120 series of irradiation experiments are statistically designed to evaluate fuel

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Figure 10. Fast-Neutron Exposure vs Reactor Residence Time for NAA-83 Experiments

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material performance as a function of temperature, burnup, and fuel composition variables. The design requirement for these experiments is to operate each fuel specimen at a uniform temperature, both axially and circumferentially, so as to eliminate the effects of temperature gradients. The temperature is to remain constant throughout the irradiation of the experiment. The effect of an in-fuel phase change from γ - to γ - and β -phase fuel material will be evaluated in four special fuel specimens designed for that purpose. These experiments have previously been described in detail.*

The design of the statistical experiments has been completed, design drawings issued, and capsule parts ordered. The NaK expansion compensation bellows, the experiment instrumentation system, and the thermocouples have been received. Although the fuel specimen design has been essentially completed, minor changes are being incorporated as a result of "feedback" information obtained from the fuel specimen mockup fabrication effort.

The phase change experiment design has been established and the detailed analysis is in progress. This experiment will employ a zirconium slug to "getter" the hydrogen from the fuel specimen just prior to the end of the irradiation.

Fabrication development studies for the capsules have been concentrated in the areas of fuel material preparation, fuel specimen assembly, and coating of the cladding assemblies.

Four fuel material sample subassemblies were fabricated, complete with support rod and fuel support thimbles, to demonstrate ability to meet the required tolerances on the various components, to verify the assembly procedures, and to provide specimens for qualification testing prior to assembly of the capsule proper. In

addition, fifteen fuel cladding assemblies were machined for verification of the processes for welding, coating, and assembly of the fuel specimen claddings.

2. Status

The NAA-120 irradiation experiments were initiated under the SNAP 8 program in FY 1965. At the end of FY 1965, the SNAP 8 program was redirected and the responsibility for these experiments came under the SNAP Reactor Improvement program. This is the first Quarterly Progress Report for the 120 experiments under the SRIP project; therefore, this report will attempt to summarize these experiments in detail, explaining the design philosophy, technical objectives, and current status.

The NAA-120 series of irradiation experiments were initially composed of four separate experimental assemblies encompassing a burnup range of 0.3 to 0.9 metal atom percent. These experiments were designated NAA-120-1, -2, -3, and -4, with the NAA-120-3 and -4 experiments being irradiated to the higher burnups of 0.6 and 0.9 metal atom percent, respectively. Because the 120-3 and -4 experiments are to be irradiated for the longest period of time, and because higher burnup data is extremely important in establishing the actual fuel growth of SNAP fuel as a function of burnup and temperature, these experiments are receiving the completed design effort at the present time.

The design efforts for these experiments will be reported under two different sections for clarity. These are the Experiment Design and Materials Evaluation sections.

a. Experiment Design

The NAA-120-3 and -4 experimental assemblies are very nearly identical, differing only slightly

*D. G. Mason, "SNAP 8 Progress Report, May-July 1965," NAA-SR-11492 (CRD), September 30, 1965

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Figure 11. Thermocouple Location Schematic for the NAA-120-3 or 120-4 Experimental Assembly

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in some of their internal dimensions. These differences are associated with the various combinations of temperature and burnup seen by the fuel specimens. Externally, these experiments are very similar to the NAA-115-1, 115-2, and 117-1 irradiation experiments which are now operating in the Hanford reactors.

Each experimental assembly consists of five separate experiments (sometimes called capsules). The first three experiments are devoted to the statistically designed fuel material study

and contain five fuel specimens each. The fourth and fifth experiment each contain one phase-change fuel specimen. The various combinations of fuel composition, temperature, and burnup are shown in tabular form in Table 2.

Each fuel specimen will have at least three thermocouples. The locations of these thermocouples are shown in Figure 11. One fuel specimen in each of experiments 1, 2, and 3 and the phase-change experiments will have an in-fuel thermocouple and a hydrogen pressure sensor

TABLE 2
NAA-120 EXPERIMENT PARAMETERS
(Nominal Values)

Experimental Assembly Number	Experiment Number	Fuel Specimen Number	Burnup (Metal at. %)	Temperature (°F)	H/Zr	Additive (wt %)
120-3	1	120-3A-1*	0.6	1300	1.85	0.10
		120-3A-2			1.70	0.02
		120-3A-3			1.70	0.10
		120-3A-4			1.85	0.02
		120-3A-5			1.85	0.10
	2	120-3B-1*		1500	1.70	0.10
		120-3B-2			1.70	0.02
		120-3B-3			1.70	0.10
		120-3B-4			1.85	0.02
		120-3B-5			1.85	0.10
	3	120-3C-1*		1600	1.70	0.10
		120-3C-2			1.70	0.02
		120-3C-3			1.70	0.10
		120-3C-4			1.85	0.02
		120-3C-5			1.85	0.10
120-4	4	120-3D-1*			1.70	0.02
		120-3D-2*			1.70	0.15
	1	120-4E-1*		1300	1.85	0.10
		120-4E-2			1.70	0.02
		120-4E-3			1.70	0.10
		120-4E-4			1.85	0.02
		120-4E-5			1.85	0.10
	2	120-4H-1*		1500	1.70	0.10
		120-4H-2			1.70	0.02
		120-4H-3			1.70	0.10
		120-4H-4			1.85	0.02
		120-4H-5			1.85	0.10
	3	120-4M-1*		1600	1.70	0.10
		120-4M-2			1.70	0.02
		120-4M-3			1.70	0.10
		120-4M-4			1.85	0.02
		120-4M-5			1.85	0.10
	4	120-4V-1*		1300	1.70	0.02
		120-4V-2*			1.70	0.15

*Instrumented fuel specimens

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in addition to three cladding thermocouples. These extremely well instrumented specimens will be referred to as instrumented fuel specimens in the following portions of this report.

The fuel specimen design has been called a "shish kebab". The criterion for the design is to maintain the fuel at a uniform temperature for the duration of this experiment. This has made it necessary to support the fuel symmetrically within the cladding with radial clearance to allow for swelling and with guard rings to eliminate end effects.

The fuel sample (center fuel sleeve) is a cylindrical sleeve of uranium-zirconium hydride. The uranium is 93% enriched and totals 10 wt% of the unhydrided alloy. At both ends of the center fuel sleeve are first, a guard ring of the same material (intermediate fuel sleeves) and second, a guard ring of fuel material containing 20 wt% enriched uranium (end fuel sleeve); see Figure 6. These five sleeves are fitted on a support rod made of fuel material containing 10 wt% of natural uranium (no enrichment). The support rod is held at either end by two Hastelloy-X fuel support thimbles which center the "shish kebab" fuel assembly within the cladding. The dimensions of the various parts of the fuel specimens are listed in Table 3.

All the fuel specimens in both NAA 120-3 and 120-4 are identical with three exceptions:

1) Instrumented fuel specimens have an axial hole drilled in the support rod to accommodate an in-fuel thermocouple and have a pressure sensor attached.

2) The phase-change experiments have a zirconium metal "getter" which will be employed to absorb hydrogen and cause phase change to occur in the fuel.

3) Since variations in burnup and temperature will lead to variations in fuel swelling, the radial clearance has been designed accordingly. Radial clearance has been adjusted by changing the OD of the fuel sleeves. The radial clearance has also been designed to maintain the cladding below 1250°F.

The locations of dynamic flux monitors are shown schematically in Figure 12. This drawing also shows the location of fuel specimens within the experiments. These experiments will be irradiated in side-to-side test facilities at Hanford.

Nuclear analysis for this experiment has established that a type-304 stainless-steel shroud

TABLE 3
NAA-120-3 AND -4 FUEL SPECIMEN DIMENSIONS

Fuel Specimen Number	End Fuel Sleeve			Intermediate Fuel Sleeve			Center Fuel Sleeve			Support Rod		
	OD (in.)	Length (in.)	Volume (in. ³)	OD (in.)	Length (in.)	Volume (in. ³)	OD (in.)	Length (in.)	Volume (in. ³)	OD (in.)	Length (in.)	Volume (in. ³)
120-3A-1, 2, 3, 4, 5	0.330	0.200	0.0073	0.4090	0.200	0.0164	0.4090	2.250	0.1851	0.250	3.790	0.1857
120-3B-1, 2, 3, 4, 5	↓	↓	↓	0.4030	↓	0.0157	0.4030	↓	0.1766	↓	↓	↓
120-3C-1, 2, 3, 4, 5	↓	↓	↓	0.3970	↓	0.0149	0.3970	↓	0.1680	↓	↓	↓
120-3D-1, 2, 3, 4, 5	↓	↓	↓	0.3970	↓	0.0149	0.3970	↓	0.1680	↓	↓	↓
120-4E-1, 2, 3, 4, 5	0.330	0.200	0.0073	0.4090	0.200	0.0164	0.4090	2.250	0.1851	0.250	3.790	0.1857
120-4H-1, 2, 3, 4, 5	↓	↓	↓	0.4010	↓	0.0154	0.4010	↓	0.1737	↓	↓	↓
120-4M-1, 2, 3, 4, 5	↓	↓	↓	0.3920	↓	0.0143	0.3920	↓	0.1611	↓	↓	↓
120-4V-1, 2, 3, 4, 5	↓	↓	↓	0.4010	↓	0.0154	0.4010	↓	0.1737	↓	↓	↓

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Figure 12. Flux Monitor Location Schematic for the NAA-120-3
or 120-4 Experimental Assembly

tube will produce the required neutron flux depression. The shroud tube will be similar in design to those used in the NAA-115 irradiation experiments.

Tolerance buildup in the various parts of the fuel specimen will inevitably lead to a certain degree of asymmetry of the fuel within the cladding. An extensive heat transfer analysis using the HEATING computer code has been performed to determine the effect of off-center fuel conditions on the circumferential temperature profile. Asymmetry will have the greatest effect on the lower temperature specimens because they have a minimum fuel-to-cladding separation.

In the worst case, when the tolerances accumulate to the maximum off-center condition, the circumferential temperature difference will be 37 and 20°F, respectively, in the 1300 and 1600°F fuel specimens.

Both NAA-120-3 and 120-4 will use the same instrumentation console even though they will be in-pile simultaneously. This console has been received at AI and is being operated to "burn out" the defects before it is shipped to the Hanford reactor site.

The ADIT pressure sensor, which was developed at AI, has been adapted for use in the NAA-120 experiments to measure the hydrogen dissociation pressure in-pile. These devices

have been tested extensively, two units having survived over 14,000 cycles at 500 psig pressure, conditions which greatly exceed those expected in-pile. The units used in this experiment will also incorporate a thermocouple, which will monitor the temperature of the pressure sensor.

The thermocouples for this experiment have been received and subjected to extensive testing to evaluate their performance and reliability. Tests include dye penetrant, helium leak, thermal shock, and insulation breakdown. After passing these tests, the thermocouples were calibrated and their electrical resistance measured. At the present time, the thermocouples are ready for use.

Sixteen dynamic flux monitors have been built and tested in the STIR reactor at thermal neutron fluxes up to 1×10^{13} nvt. These flux monitors are being used to evaluate the relative efficiency of rhodium, vanadium, and boron carbide electrodes. These tests after evaluation will determine which type will be used in the NAA-120 experiments.

The design of the statistical fuel experiment has been completed, refinement of the design is being incorporated as indicated by the fuel mockups. As previously mentioned, many of the long lead-time items such as NaK expansion compensation bellows, thermocouples, and the

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instrumentation console have been received and are ready for use.

The design of the phase-change experiments is in progress. The zirconium "getter" will be heated by gamma heating rather than by electrical heaters because of reliability considerations. The zirconium specimen temperature will be controlled by a separate gas system which will be used to induce phase change when the experiment has achieved about 90% of design burnup.

b. Materials Evaluation

A redirection of the fuel support rod fabrication was required due to the unexpected low mechanical strength of the modified zirconium hydride material. This material broke during handling when it was being assembled into the test specimen subassembly. In order to circumvent this problem a decision was made to replace the modified zirconium hydride with zirconium hydride containing normal uranium because of the superior mechanical strength properties of the uranium-containing material. This material was the original choice and was initially replaced because of the possibility of swelling during low-temperature irradiation. Since then, continued studies have led to an explanation of the unusual swelling of hydride fuel during low-temperature irradiation such as that observed in the NAA-116 capsules under the SNAP 8 program. In brief, thermal cycling tests performed under that program have indicated that alpha uranium growth is probably responsible for the dimensional changes observed in the NAA-116 capsule specimens. Since alpha uranium growth, under conditions of low temperature irradiation, is proportional to the concentration of fissile atoms present, the use of normal uranium in the support rods should minimize this problem.

With the exception of this replacement material — zirconium hydride containing normal uranium — isochore testing and chemical analy-

sis have been essentially completed on all fuel specimen materials, including fuel support rod materials. Testing of the replacement material, however, is still continuing to assure an adequate supply of carefully characterized fuel materials for the test.

Two fuel alloy melts were made during the quarter. These melts, containing 10 and 20 wt % uranium, respectively, were made to augment the reserve material being depleted due to fuel breakage during fabrication in excess of the breakage anticipated.

A photograph of a fuel material specimen mockup assembly that was completed to verify the fabrication and assembly procedures is presented in Figure 13. In the assembly procedures the enriched sleeves and fuel supports were shrink-fitted over the support rod. The sleeves were initially left oversize (before assembling onto the support rod) and final grinding to size was performed after the assembly in order to assure high precision in the finished parts. However, after the Hastelloy end support pieces were installed it was found that the assembly had excessive total indicated runout. The final assembly procedure has therefore been changed

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Figure 13. NAA-120 Fuel Assembly Mockup

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and the final grinding operation will be performed by indexing from the Hastelloy end-support thimbles.

An additional problem was encountered with the Hastelloy end-support pieces which were coated with alumina to prevent bonding to the permeation barrier material during irradiation. It was found that the alumina coating tended to spall during the grinding operation. As a consequence, alternate support coating materials are being tested for this application. Based on preliminary results of the tests, vapor-deposited tungsten appears to be an attractive material for this application and samples are being prepared for fabrication, final machining, and assembly tests.

Hastelloy-X has been selected for the fuel-pin cladding material and is being used in the final cladding assembly fabrication and mockup tests. The original plan was to use Hastelloy-N because of the extensive experience with this material during the SNAP fuel element programs. However, it was found that excessive distortion of the 0.030-in. wall tubes occurred during the chromizing operation. Low-cobalt Hastelloy-X-285 was next evaluated but serious welding problems in the form of cracking were encountered. The decision was then made to use Hastelloy-X as the cladding material in spite of its cobalt content since previous experience indicated that it can be satisfactorily coated with the barrier material, it can be routinely welded, and it is readily available. The increased macroscopic absorption cross section due to the cobalt content does not represent a major problem for this application.

Studies of the coating problems related to the NAA-120 capsule were performed to verify that the short thick-walled claddings required for the capsule can be routinely coated and the closure ends sealed while maintaining required tolerances and low hydrogen leak rates. The coating

studies consisted of coating both the closure end plugs, coating the heavy-wall blind end tubes, and performing the fusion seal at the closure end. For the closure end a parametric study of fusion times and temperatures was performed and the fused end seals were subsequently permeation tested. Results are shown in Table 4.

Six fuel-pin closure end plugs were coated as a part of the assembly technique development work. Four of these plugs were Hastelloy-X-285 material and two were Hastelloy-X material. The Hastelloy-X-285 parts were used for convenience since (1) the purpose of the tests was primarily to verify that required dimensional tolerances can be met and the tolerances achieved are essentially independent of the details of the substrate composition, and (2) the parts were available from previous tests. The results of the tests are presented in Table 5. It was found that, with the exception of plugs numbered 32 and 33, all of the coated plugs were within the required dimensional tolerances. It is therefore anticipated that no problems will be encountered in coating of the end plugs for the actual capsule fuel pins.

In addition to coating tests of the end plugs, a number of blind-end heavy-wall tubes were coated to establish the dimensional tolerances that could be achieved. Inside diameter measurement data for three of the tubes are presented in Table 6. As a result of these tests, the inside diameter requirement was fixed at $0.4120 \pm \begin{smallmatrix} 0.0007 \\ 0.0002 \end{smallmatrix}$ inches for the NAA-120 coated fuel cladding tubes.

D. NAA-XXX-1 IRRADIATION EXPERIMENT

1. Summary

The objective of the NAA-XXX series of experiments is to establish the high-temperature capability of the present SNAP type fuel elements with respect to burnup and fuel composition. In general, the series will test both fuel

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TABLE 4
LEAK RATE DATA FROM THE NAA-120
FUSED-END PROCESS PARAMETRIC
STUDY

Fusion Time (min)	Leak Rate [cc(STP)/hr at 1500°F, 3 atm.] Fusion Temperature (°F)		
	1900	1950	2000
1	0.07	0.64	0.08
2	in test	1.48	1.37
3	0.06	in test	0.11

TABLE 5
COATING OF NAA-120 PLUGS

Plug Number	Alloy	OD (in.)		Average Enamel Thickness (0.001 in.)
		Before Coating	After Coating	
32	Hastelloy-X-285	0.4150 ± 0.0010	0.4197 ± 0.0005	2.18
33	Hastelloy-X-285	0.4165 ± 0.0005	0.4211 ± 0.0005	1.85
34	Hastelloy-X-285	0.4158 ± 0.0003	0.4211 ± 0.0003	2.65
35	Hastelloy-X-285	0.4170 ± 0.0010	0.4210 ± 0.0002	2.00
36	Hastelloy-X	0.4170 ± 0.0004	0.4206 ± 0.0003	1.80
37	Hastelloy-X	0.4173 ± 0.0001	0.4213 ± 0.0001	2.00

Specifications

OD before coating 0.4170 ± 0.0005 in.

OD after coating 0.4210 ± 0.0001 in.

Enamel thickness 0.001 in. min.

TABLE 6
DIMENSIONAL DATA FOR COATED HEAVY-WALL NAA-120 FUEL
SPECIMEN CLADDING TUBES

Tube Number	Alloy	ID (in.)						TIR (in.)
		Distance From Blind End (in.)						
		1	2	3	4	5	6	
1B	Hastelloy-X-285	0.4183	0.4178	0.4178	0.4171	0.4171	0.4168	0.0005
2B	Hastelloy-X-285	0.4180	0.4180	0.4171	0.4175	0.4175	0.4172	0.0015
18	Hastelloy-X	0.4180	0.4175	0.4174	0.4174	0.4175	0.4170	0.002

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and cladding, however, the NAA-XXX-1 will be only a fuel test, with its primary objective the establishment of upper limits of peak fuel temperatures as a function of metal atom percent burnup.

2. Status

The NAA-XXX-1 experiment is a high-temperature, relatively low-burnup fuel material irradiation experiment. This experiment will be similar to the NAA-120 irradiations in that quantitative values of fuel volume increase will be determined, covering temperature and burnup ranges (1600 to ~1900°F and 0.2 to ~0.7 metal atom percent) which are not covered by the 120-3 and -4 experiments. By combining the data from the NAA-120 and the NAA-XXX-1 capsules and other SNAP irradiation experiments, a complete mapping of the anticipated temperature range for SNAP reactors will be covered.

Based on the previously stated temperatures and burnups and utilizing an extrapolation of the "Miller" correlation, the volume increase of the fuel in the XXX-1 experiment should be about 6 to 14% ($\Delta V/V$). Figure 14 is a diagram which indicates the percent of fuel volume increase for this experiment as a function of burnup and temperature. Included in this figure is a summary of the AI irradiation experiments for the past, present, and future showing how the NAA-XXX-1, -2, and -3 experiments will investigate a completely new temperature regime and why they are necessary if the peak temperature capabilities of the existing SNAP-type fuel elements are to be investigated.

During the past quarter a parametric study was initiated to establish the NAA-XXX-1 design criteria and limits from the standpoint of heat

transfer, fuel composition, and nuclear and geometric characteristics. The results of this analytical study will be incorporated into the experimental fuel element design from which elements will be fabricated for irradiation and evaluation.

E. ENVIRONMENTAL TESTING PROGRAM

1. Summary

In May 1963 the fuel environmental testing program was initiated to determine the reliability of the SNAP 10A fuel elements as a function of the severity of the environment. The program was designed to define the levels of vibration and shock and ramp heating beyond which failure will occur.

During this report period the Fuel Environmental Testing Program was terminated and a topical report summarizing the work written. The present report covers those aspects of this program not covered in the previous two quarterly reports.

2. Status

As previously stated, the Fuel Environmental Test Program was terminated during this report period. Prior results of this test program are discussed in the previous quarterly. All 120 elements in this program received a dimensional check and a permeation test after endurance. No significant change in dimensions was found. K_{BLD} data[†] for the elements are summarized in Table 7.

The results of 1400°F tests of ten fuel elements which had completed the Fuel Environmental Test Program are discussed below. All ten of these elements were at 1400°F for at least 2500 hours. Eight of the ten completed

R. J. Gimera and J. Susnir, "Progress Report, SNAP Reactor Improvement Program, April-June 1965," NAA-SR-11490 (CRD), August 31, 1965

[†]Op.cit., p. 15

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Figure 14. Predicted Fuel Swelling vs Burnup and Temperature

(For SNAP-Fueled Reactors and Experiments, Based Upon Miller's Correlation:

$$\frac{\Delta V}{V} = B + 1.2 \times 10^6 B^2 e^{-[22600/T(R^\circ)]}$$

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TABLE 7
PRE- AND POST-ENDURANCE K_{BLD} VALUES BY PHASE

Phase	Number of Elements	Endurance Temperature (°F)	Average K_{BLD} Based On 1200°F Permeation Test		Percentage of Elements With Increased K_{BLD}
			Before Endurance	After Endurance	
I (High level V & S)	12	1200	329.5	266.2	33
	12	1300	1003.0	944.9	58
	6	1350	612.1	1008.5	83
II (High ramp heat)	12	1200	506.4	431.1	17
	12	1300	566.8	772.7	67
	6	1350	715.5	1609.5	100
III (Interaction)	12	1200	1607.1	945.6	8
	12	1300	1269.8	986.8	25
	6	1350	1417.6	1476.8	67
IV (Reference pre-endurance)	12	1200	926.0	754.8	0
	12	1300	1211.6	1070.9	8
	6	1350	1074.1	1173.1	67
All	48	1200	842.2	599.4	15
	48	1300	1012.8	943.8	40
	24	1350	954.8	1342.0	79

over 3200 hours with only slight barrier damage. The other two elements, N-88 and N-101, began 1400°F test in a single retort. After 200 hours at 1400°F, the hydrogen flow from the retort increased about 40%. The elements were unloaded and then permeation tested in individual retorts at 1200°F. The 1200°F permeation of N-88 had increased from 0.24 to 0.86 cc(STP)/hr, while that of N-101 had decreased from 0.24 to 0.11 cc(STP)/hr. The temperature of these two elements, still in individual retorts, was increased to 1400°F. After 1200 additional hours at 1400°F, the temperature of these elements was cycled to 1200°F and the permeation rates again measured. The permeation of N-88 was unchanged, but the permeation of N-101 increased to 0.17 cc(STP)/hr. The elements were again heated to 1400°F and remained there until they had accumulated about 1650 hours at this temperature.

The temperature of N-101 was then increased to 1450°F for 6 days, while N-88 remained at 1400°F. When N-101 was again tested at 1400°F, its permeation rate had doubled, indicating significant damage. Both elements were then cycled from 1400°F to room temperature and back to 1400°F. The permeation rate for N-88 showed no effect, but the rate for N-101 increased about a factor of 7, indicating severe barrier damage. The data for these 10 post-endurance elements are shown in Figure 15 and Table 8.

From these data, we conclude that 1400°F, per se, causes increased permeation rates even though the internal hydrogen pressure decreases. This indicates a less efficient hydrogen barrier. In addition, it appears that extended time at 1450°F, combined with 1400°F testing and thermal cycling, can cause catastrophic damage.

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Figure 15. K_{BLD} vs Time for Ten Post-Endurance Environmental Testing Elements at High Temperature

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HIGH TEMPERATURE TEST DATA FOR
10 POST-ENDURANCE ENVIRONMENTAL
TESTING ELEMENTS

Element Number	Composition				Length (in.)		Dimensions (Maximum/Minimum) (in.)						Fuel Element Weight (gm)		Time at 1400°F (hr)	Test Data			K _{BLD} at 1400°F High/Low
							OD-1/2 in. From Closure		OD-Center		OD-1/2 in. from Blind End					Internal Pressure (High/ Low) at 1400°F (atm)	K _{BLD} at 1200°F		
	wt % H ₂		wt % Add.	wt % U	Before	After	Before	After	Before	After	Before	After	Before 1400°F	After 1400°F					
	Before	After														Before	After	Before	
N-88*	1.790	1.659	0.20	9.98	12.449	12.454	1.253 1.252	1.2550 1.2540	1.254 1.251	1.2590 1.2545	1.253 1.253	1.2560 1.2550	N. A. [†]	1536.944	*	18.1 1.6	255.9	719.2	4076.7 656.1
N-97	1.768	1.697	0.24	10.00	12.451	12.456	1.253 1.251	1.2545 1.2535	1.252 1.251	1.2620 1.2590	1.252 1.251	1.2595 1.2585	N. A.	1537.933	3840	12.9 3.6	345.4	2712.3	1640.7 352.6
N-101*	1.798	1.623	0.24	10.00	12.452	12.456	1.254 1.253	1.2585 1.2555	1.251 1.249	1.2830 1.2780	1.252 1.250	1.2900 1.2880	N. A.	1536.260	*	25.6 1.1	260.4	8083.0	13274.8 667.5
N-104	1.770	1.701	0.21	10.03	12.450	12.459	1.251 1.249	1.2560 1.2545	1.253 1.250	1.2750 1.2620	1.254 1.252	1.2615 1.2610	N. A.	1535.332	3624	12.7 3.4	251.6	1252.9	1668.3 433.6
N-105	1.825	1.758	0.21	10.03	12.453	12.457	1.254 1.252	1.2560 1.2550	1.254 1.251	1.2620 1.2570	1.255 1.253	1.2630 1.2590	N. A.	1538.141	3624	42.8 9.8	77.0	785.5	683.8 348.2
N-137	1.792	1.747	0.21	10.03	12.447	12.450	1.252 1.249	1.2530 1.2520	1.252 1.248	1.2530 1.2510	1.253 1.251	1.2525 1.2515	N. A.	1537.796	3624	20.2 7.6	89.1	497.8	773.4 181.3
N-146	1.794	1.759	0.225	10.00	12.449	12.451	1.250 1.249	1.2535 1.2520	1.251 1.248	1.2515 1.2505	1.252 1.250	1.2515 1.2505	N. A.	1536.882	3624	22.3 9.1	1.0	540.6	626.8 59.6
N-155	1.795	1.758	0.145	9.98	12.451	12.455	1.251 1.249	1.2550 1.2525	1.251 1.249	1.2640 1.2590	1.253 1.249	1.2610 1.2580	N. A.	1534.654	3624	16.4 7.8	72.0	726.8	477.6 124.1
N-159	1.776	1.717	0.21	10.00	12.449	12.450	1.250 1.248	1.2535 1.2520	1.250 1.249	1.2525 1.2515	1.252 1.251	1.2515 1.2500	N. A.	1535.750	3624	14.3 4.6	169.3	1268.5	1189.1 309.1
N-165	1.800	1.717	0.135	10.10	12.447	12.456	1.250 1.249	1.2540 1.2530	1.250 1.249	1.2570 1.2540	1.253 1.251	1.2550 1.2540	N. A.	1534.393	3840	16.8 3.8	381.4	944.6	1634.2 726.2
Average	—	—	—	—	12.450	12.454	1.252	1.2542	1.251	1.2598	1.257	1.2594	—	—	—	—	190.3	1753.1	—

*Elements N-88 and N-101 received a series of tests from 1200°F to 1450°F (see text).

†Not Applicable

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The results of individual tests in the Fuel Environmental Testing Program are as discussed above and in the preceding two quarterly progress reports. Final updated values of assessed reliability for reactor operation are shown in Table 9.

TABLE 9
ASSESSED RELIABILITY FOR SNAP 10A
REACTOR OPERATION
(50% Confidence Level)

Input	Reliability	
	One Fuel Element	37-Element Core
Thermal soak	>0.99999	>0.99963
Vibration and shock (100%)	>0.99999	>0.99963
Ramp heating	0.99979	0.99926
All pre-endurance inputs	0.99979	0.99926
Endurance:		
90 days	>0.99999	>0.99963
1 year (estimated)	>0.99999	>0.99963
All inputs	0.99979	0.99926

A report of the Fuel Environmental Test Program, now terminated, is being issued.*

F. BURNABLE POISON APPLICATION STUDIES

1. Summary

A burnable poison in the form of Sm_2O_3 is added to the ceramic hydrogen barrier coating contained on the inside surface of SNAP reactor fuel cladding tubes. The burnable poison provides for passive control of the reactor by burning out at a rate that compensates for reactivity losses resulting from hydrogen loss, fuel depletion, and fission product accumulation. Neutron activation techniques have been developed to permit 100% nondestructive inspection of coated cladding tubes. While this method provides much higher reliability than destructive chemical analysis of a small sample of tubes, the coated tube activation method has encountered

several problems. One of these is that when a coated cladding tube is irradiated in a reactor, not only the burnable poison, but one of the alloying elements in the cladding tube (molybdenum) becomes radioactive and forms an isotope, Mo^{99} , which emits gamma photons as an undesirable background when counting the radiation emitted by the activated poison. A proposed solution is irradiation of the poison-containing barrier coating prior to its application to the cladding tube. The anticipated advantages are as follows:

1) The pre-irradiated slip method would permit poison measurements to be made immediately after coating, which would aid production personnel in more accurate control of the coating process.

2) The new method will produce more accurate poison measurements because there will be no undesirable background radiation emitted by the activated cladding tube.

3) The cost of poison analyses should be reduced.

Major efforts during this reporting period were concerned with: (1) the procurement and checkout of special scintillation crystals for measuring gamma radiation inside tubes containing activated ceramic coating, (2) devising handling procedures and obtaining approval of these procedures for handling irradiated slips, and (3) investigating the available reactor facilities necessary to activate poisoned slip.

All further work on this project is being terminated as of the end of this reporting period. This termination is in accordance with a re-direction of the SRIP project objectives.

2. Status

Improved design scintillation probes were received for use in 1/2- and 1-1/8-in.-ID tubing.

*J. L. Isaacs, "Environmental Testing of SNAP 10A Fuel Elements," NAA-SR-11547 (CRD), to be published

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The probes contained 3/4-in.-diameter by 1-in.-long and 1/2-in.-diameter by 1-in.-long NaI (Tl)* crystals. The 3/4-in.-diameter crystals were directly coupled to a 3/4-in.-diameter photomultiplier tube, while the 1/2-in.-diameter crystals were coupled to a photomultiplier through a 20-in.-long lucite light pipe. The portion of the probes designed for insertion in tubing are contained in 20-in.-long light-tight housings. The end surfaces of the NaI (Tl) crystals are shielded from gamma radiation to permit only that radiation emanating from 360° directly over the 1-in. crystal length to activate the counting system. Previously used detectors were considerably larger and less efficient than the new miniaturized probes based on design requirements. However, checkout of the new probes found them to be insensitive to the low-level (70 kev) gamma radiation from Sm^{153} . The probes have been returned to the manufacturer for repair or replacement.

Health and Safety requirements, as well as irradiation reactivity requirements for preliminary studies, were based on activating 30 cc of poison containing SCB-1 slip. This was determined to be the minimum volume required for coating the inside surface of a 4-in. length of SNAP 8 (0.540-in. ID by 0.010-in. wall) Hastelloy-N alloy fuel cladding tubing. SCB-1 slip contains 3.7 wt % Sm_2O_3 on a dry weight basis or 0.04 gm/cc of Sm_2O_3 in the liquid slip. The procedure approved for handling this quantity of activated slip is as follows:

1) Double-encapsulate the slip in glass and aluminum tubes.

2) Immediately after irradiation, place and hold the capsule containing the slip in a 1/2-in.-thick lead container for 12 hours before further processing. Capsule activity

immediately after irradiation is to be no greater than 200 mrem/hr at its surface.

3) Coat and dry the tube by hand, using rubber gloves in a standard laboratory hood with a minimum air flow of 120 cfm. After drying, fire the sample in a normally vented gas furnace and gamma scan. The samples are kept in the "blue tag" area for 14 days or seven Sm^{153} half-lives.

4) All slip spillage is removed, placed in plastic bags, held 14 days (seven Sm^{153} half-lives), and disposed of as clean waste. The Sm^{153} is the only constituent of the slip that exhibits any significant radioactivity.

Reactivity worth determination of the poisoned SCB-1 slip made in the Atomics International 10-watt L-77 reactor, has shown its negative reactivity to be 0.051% ρ/gm . Based on this value, the total negative reactivity for 30 cc of slip is 2.78% ρ . The allowable excess negative reactivity permitted by the L-77 operating license is 0.5% ρ . Therefore, based on this limitation, in-core irradiation could not be performed in this reactor. Similarly, the 1.0% limitation of the 3-kwt AE-6 reactor located at the Atomics International Santa Susana facility prevents the use of this reactor for in-core slip irradiation. However, it was established that irradiation could be performed at the 1-Mwt STIR, also located at the Atomics International Santa Susana facility. Here irradiation can be performed out-of-core in a thermal neutron flux of $\sim 10^{11}$ neutrons/cm²/sec. A specially designed containment capsule was built and loaded with slip for irradiation. This irradiation has not been performed.

All further work on this project, as previously stated, is being terminated as of the end of this report period in accordance with a redirection of the SRIP project objectives.

*NaI (thallium activated).

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G. CREEP STUDIES (OUT-OF-PILE)

1. Summary

A series of creep studies were conducted in FY 65 under the SNAP 8 program on SCB-coated SNAP cladding. These studies were designed to determine the low stress-long term creep behavior of ceramic-coated Hastelloy-N. External environments of both vacuum and NaK were investigated. Results indicated a slightly accelerated creep rate in NaK, although more tests are required to substantiate these preliminary conclusions.

Under the SRIP program in FY 66 these studies are to be continued; however, the basic program has been modified to provide data in support of the NAA-83 irradiation experiments. Test parameters have been altered to be identical to those proposed for the in-pile experiment. Four NaK thermal convection loops have been fabricated and samples are being prepared to meet this goal.

2. Status

Studies of the creep behavior of SNAP fuel element cladding were initiated in FY 65 under the SNAP 8 program. Although the experimental work was completed in FY 65, the data had not been analyzed. During the first quarter of FY 66 the SRIP program has analyzed these tests, thus enabling future out-of-pile creep studies to be performed in the most efficient manner.

This work demonstrated the effect of the various SNAP process parameters on the creep behavior of the cladding. The results of this work have been recently published.

The second phase of this work, also funded by SNAP 8 in FY 65, studied the long-term low-stress effects on the cladding. Ten SCB coated cladding tubes were internally pressurized with

hydrogen to provide a hoop stress of 1250 psi. The tubes were tested at 1450°F with vacuum as the external environment. Two additional cladding tubes were also tested in NaK thermal convection loops to observe the effect of NaK external environment on the creep behavior. The results of both the vacuum and the NaK tests are summarized in Table 10.

An attempt to obtain a composite strain-time creep curve by measuring the creep deformation of these samples with varying amounts of accumulated testing times (2000, 3000, 4000, and 5000 hours) proved unsuccessful. A comparison of the pre- and post-diametral measurements obtained for the 2000-, 3000-, and 4000-hour samples in vacuum indicated the creep deformation to be zero or very close to the minimum limits of the measuring system. The results obtained for three of the four 5000-hour samples were in agreement and an average diametral creep rate of $1.2 \times 10^{-5}\%$ /hr was representative. This value does not indicate any acceleration of creep due to the low-stress long-term test condition. The creep rate appears to be comparable to that obtained by Oak Ridge for INOR-8 (Hastelloy-N) in molten salt and air (ORNL-2780).

One of the two samples tested in NaK loops failed after 2000 hours. Examination revealed that failure occurred in the region where the SCB hydrogen barrier had been machined off to allow insertion of the end cap assembly. This area of the tube differed from SNAP 8 fuel element assembly specification in that the ceramic blending operation was deleted and approximately one-half inch of the hydrogen barrier ceramic was removed from the top portion of the tube by a lathe operation. Metallographic examination of the failed area revealed the following

- 1) Work hardening and subsequent recrystallization of approximately 25% of the

¹G. E. Culley and J. T. Berling, "Effects of Processing and Operating Parameters on Creep Rates of SNAP Reactor Fuel Element Cladding," NAA-SR-11316 (CRD), October 5, 1965

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TABLE 10
BIAXIAL CREEP TESTS ON SNAP FUEL CLADDING - SCB-COATED
(Hoop stress 1250 psi, Temperature 1450°F)

Sample	Environment		Test Duration (hr)	Total Diametral Deformation (%)	Average Diametral Creep Rate (%/hr)
	Internal	External			
2925	H ₂ ↑	Vacuum	2000	No change	—
2915		↑	2000	No change	—
2582			3000	No change	—
2801			3000	0.030	1.0×10^{-5}
2752			4000	No change	—
2945			4000	0.030	7.5×10^{-6}
2950			5000	0.062	1.2×10^{-5}
2897	↓	↓	5000	0.060	1.2×10^{-5}
2979			5000	0.058	1.1×10^{-5}
2975		Vacuum	5000	0.020	4×10^{-6}
2953		NaK	2000	0.16	8×10^{-5}
2956		NaK	4000	0.15	3.7×10^{-5}

tube wall was observed in the area of interest. This hardening probably resulted from the lathe operation that was used to remove the hydrogen barrier ceramic. The ASTM grain size for the recrystallized area was 8 to 9 as compared to 2 to 4 which represented the remaining 75% of the tube wall.

2) Large amounts of grain boundary precipitates were observed in the area of the small grains as compared to areas of large grains.

3) The tube wall thickness was slightly reduced (approximately 0.001 to 0.002 inches) by the lathe operation and the transition point from the machined to unmachined resulted in a step change in wall thickness.

From these observations, it appears that failure occurred due to local stress risers and wall thinning introduced in the cladding during the machining operation. Such conditions are not likely to be present in a fuel element because the end seal fusion would remove the hydrogen pressure from this region and also provide mechanical support.

The average diametral creep rates obtained for tests conducted in NaK, presented in Table 10, are higher by a factor of 3 to 6 than identical tests conducted in vacuum. A test plan to attain more quantitative creep data in NaK to substantiate these findings has been redirected. The new program objective is to conduct biaxial creep tests in NaK thermal convection loops under conditions identical to the NAA-83 irradiation experiments. These tests will support the NAA-83 experiments and show the effect of flowing NaK on the creep behavior of S8DS-type membranes. Four samples will be tested at 1400°F under hoop stresses of 8000, 4000, and 2500 psi (two samples). The 8000-psi sample will be tested to failure, whereas the other samples will be tested for 4000 hours.

Four NaK thermal convection loops have been constructed for these tests and are presently available for use in this study. Sample fabrication is underway.

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V SNAP 2 DRM-1

A. TEST SUMMARY

1. Test Objectives

The primary objective of the SNAP 2 DRM-1 program was to establish the adequacy of the Interim SNAP 10A/2 reactor components integrated into a complete reactor-reflector shield system and subjected to the conditions required by the SNAP Reactor Improvement Program for Mercury-Rankine and Thermoelectric Power Conversion Systems. A secondary objective of the program was to design, fabricate, and operate an ultrahigh-vacuum system capable of subjecting a complete reactor system to conditions more nearly simulating space environment than those obtained by conventional methods, i.e., diffusion pump and/or ion pump systems.

2. Test Operations (Figure 16)

The SNAP 2 DRM-1 system test was deleted from the SNAP Reactor Improvement Program during this quarter, all testing was terminated and the system, including test hardware, was disassembled and stored. A summary of all S2DRM-1 tests to date follows

The reflector assembly was subjected to acceptance level and two simulated hot critical thermal-vacuum tests as a separate unit in December 1964 (Figure 17). The shield assembly was subjected to five thermal-vacuum cycles as a separate unit to simulate hot critical testing that would be performed on a flight system. The shield tests were conducted between September 10, 1964 and October 12, 1964 in Vacuum System "B" at the Santa Susana Building 062 facility (Figure 18). NaK loading tests and the thermal-vacuum operational tests of the complete reactor system were performed in

Vacuum System 'B' at the Santa Susana Building 062 facility between March 10, 1965 and April 28, 1965 (Figure 19). In the interim the reactor system was assembled and subjected to launch simulation vibration tests at the Santa Susana Building 027 facility (Figure 20). A performance record test consisting of mechanical, electrical, and operational measurements was performed prior to and after each phase of testing to determine the integrity of the reactor system. A helium leak check, dimensional check, and x-ray inspections were made of the shield assembly prior to and after the simulated hot critical thermal-vacuum tests to determine the effects of thermal cycling on the shield structure. Further inspection of the shield is not scheduled at this time.

A vacuum system was designed (Figure 21) by expanding the existing state-of-the-art in titanium-sublimation supplemented ion pumping methods to meet the large gas loads anticipated from the S2DRM-1 reactor system. This method of achieving a "hard" vacuum uses a periodically replenished surface film of active titanium deposited on a liquid-nitrogen cooled surface as a primary pump. An electron-beam gun (Figure 22) is used to heat a module of titanium which vaporizes and deposits a thin film on the cooled surface. The pumping action is accomplished by chemical combination of the active gases which strike the titanium film and form low-vapor-pressure solids. Shields are mounted around each titanium module to prevent the vaporized titanium from striking the test item. Ion pumps supplement the getter pumping of the titanium film and are primarily used to pump the noble gases. Mechanical and oil diffusion pumps are used to rough the chamber

R. J. Piper and J. A. Craig, "Ultra-High Vacuum With Extremely High Pumping Speeds," 1965 Proceedings, Institute of Environmental Sciences

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TEST SEQUENCE

SZDRM-1 ASSEMBLY PRIOR TO LAUNCH SIMULATION VIBRATION AND SHOCK TESTS

Figure 16. SNAP 2 DRM-1 Reactor System Nonnuclear Test Program

*TEST TERMINATED AFTER 18 DAYS DUE TO FAILURE OF SYSTEM NaK HEATER
MODIFICATION INITIATED TEST TO RE-START 9-1-65 ALL WORK TERMINATED 7-1-65

STATUS 7-19-65

7623-0561E

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Figure 17. S2DRM-1 Reactor Assembly

10-22-65

7635-55129A

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7635-5422A

Figure 18. S2DRM-1 Shield and Test Fixtures
Assembled for Simulated Hot Critical
Thermal-Vacuum Test

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10-22-65

7635-1232A

Figure 19. S2DRM-1 Reactor System Installed in Vacuum System "B", Building 062, Santa Susana

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Figure 20. S2DRM-1 Reactor System and Test Fixtures for Vibration Tests

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7647-0153

Figure 21. Interim SNAP 10A/2 DRM-1 High Vacuum Test Chamber

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7635-55137A

Figure 22. Electron-Beam Gun

down to the 10^{-6} torr range and are valved off from the chamber after the ion pumps are started. A method of sealing the numerous flanges was devised using indium instead of elastomer O-rings, indium being a soft metal which flows under light force but has a very low vapor pressure. This decreased the chamber gas load caused by outgassing of elastomers.

3. Test Results

The acceptance level thermal-vacuum, simulated hot critical, shock, vibration, NaK loading, and thermal-vacuum operational test demonstrated that the S2DRM-1 reactor assembly can survive the nonnuclear environments that would be imposed upon it from final assembly through startup and 20 days of operation at temperature in a vacuum (simulating an earth orbit) without failure. The thermal-vacuum operational test was terminated due to the failure of the NaK heater assembly which was not part of the reactor system. Comparison of the performance

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record test data taken prior to and after each phase of testing showed no signs of degradation of the reactor assembly components. Due to the unavailability of SNAP 2 fine control drum motors, SNAP 10A flight-type motors (with mounting adapters) and controller were used to rotate the fine control drums during the tests conducted on the S2DRM-1 reactor assembly. The SNAP 10A fine control drum motors and controller demonstrated their compatibility with the SNAP 2 reflector assembly and operating requirements. A preliminary pumpdown of the modified vacuum chamber was conducted prior to installing the reactor system. An ultimate pressure of 5×10^{-10} torr was obtained on the clean, dry, and empty chamber without bakeout. The qualitative and approximate quantitative information obtained from a residual gas analyzer showed a substantial decrease in residual gases present in the chamber after the diffusion pumps were isolated from the chamber (by their high vacuum valves) and all of the pumping was being done by the ion pumps and titanium sublimation systems.

Electron-beam gun problems encountered during the thermal-vacuum operational test of the S2DRM-1 reactor assembly have been resolved, additional operational testing of the ultrahigh-vacuum system, however, would be required to demonstrate endurance capabilities.

B. FABRICATION

Following the failure of the NaK heater assembly during thermal-vacuum testing of the S2DRM-1 Reactor System, a 30-element NaK heater and reservoir assembly was designed

and fabricated for testing the S2DRM-1 Reactor System at elevated temperatures up to 1450°F. The heater and reservoir assemblies were not installed in the reactor system heat simulator, due to program curtailment. The new heater was designed to be installed vertically in the chamber base of Vacuum System "B" at the Santa Susana facility, with provisions for replacing the heater elements from the bottom without interrupting the test. The major consideration in the heater design was to provide an ullage area above all of the heaters to decrease the possibility of hot spots on the heaters resulting from accumulation of vapors in the event of film boiling. The heaters were designed to be inserted into thimbles which allow for easy replacement. An expansion reservoir was designed and fabricated to allow for expansion of the NaK at elevated temperatures (up to 1500°F).

C. HARDWARE DISPOSITION

Due to program curtailment, the S2DRM-1 Reactor System was disassembled and temporarily stored at the Santa Susana Building 062 facility along with the associated test fixtures and hardware. The test fixtures and hardware include the vibration fixture, radiator-condenser heat simulator, handling fixtures, wiring harness, redesigned NaK heater and reservoir assemblies, and the associated NaK plumbing for the reactor system heat simulator.

D. REPORTS

A rough draft of the Topical Report on the S2DRM-1 testing has been completed and is being reviewed.

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