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QUARTERLY PROGRESS REPORT
JANUARY-MARCH 1962
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I. PROGRAM OBJECTIVES

SNAP 8 is a nuclear power supply system designed to provide 30 or 60 kw of electrical power on space vehicles. It is based on the SNAP 2 concept of a NaK-cooled uranium-zirconium hydride reactor coupled to a mercury Rankine cycle power conversion system. The SNAP 8 system is being developed jointly by the Atomic Energy Commission (AEC) and the National Aeronautics and Space Administration (NASA).

The objectives of the SNAP 8 development work under this contract are to (1) develop the SNAP 8 reactor, including its associated controls, shield, and safety features, and (2) test the complete SNAP 8 system, i.e., reactor coupled to power-conversion equipment. The power-conversion equipment is being developed by the Aerojet General Corporation under contract to NASA.

The general objectives and operational constraints are shown in Table I.

TABLE I
SNAP 8 GENERAL OBJECTIVES

Planned application	Electrical propulsion
Test vehicle	Atlas-Centaur
Safety criteria	Subcritical during launch Orbital startup Orbital shutdown at end of power-conversion system (PCS) life backed up by re-entry shutdown mechanism
Life	10,000 hr
Power conversion	Mercury Rankine cycle Turboalternator being developed by Aerojet under contract to NASA
Electrical power output	30 kw (capable of extension to 60 kw)
Reactor thermal power	300 kw (capable of operation at 600 kw)
Control mode of operation	System operates at full power. Electrical power not used by payload is dissipated in parasitic load. The reactor control system maintains the reactor coolant outlet temperature around 1300°F by stepping reflector control elements in or out as required.
Reactor weight objective	450 lb
Shadow shield weight objective	1000 lb

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The shadow shield is designed to reduce the direct dose at the payload to 5×10^{12} nvt from fast neutrons and 9×10^6 rad from gammas. The dose due to scattering from the radiators or other structures is assumed to bring the total dose at the payload to 10^{13} nvt and 10^7 rad. The payload is assumed to be 8 ft in diameter and located 15 ft from the reactor.

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II. PROGRESS SUMMARY

A. SYSTEM DESIGN

In view of the anticipated operation of the flight reactor at the 300 kw power level, the shadow shield has been redesigned for this power level. The TiH thermal shield has been replaced by a 1-in. thick tungsten gamma shield. The LiH neutron shield is a two-region shield containing 99.99% Li^7H at the hot end (end nearest reactor) and natural LiH at the cold end.

Control and stability analyses of the S8ER and the flight reactor are in progress.

The current design weight of the entire nuclear system is 1440 lb.

B. FUEL DEVELOPMENT

The addition of 0.5 wt % carbon to fueled zirconium hydride has been found to make the material much less susceptible to cracking during hydriding. The current yield of uncracked hydrided rods is in excess of 90%. Initial experiments indicate that the carbon addition results in an increase in the hydrogen dissociation pressure of the fuel.

Experiments were performed aimed toward determining changes in the behavior of the AI 8763D coating as a function of time and temperature in a hydrogen environment.

Hydrogen permeation rates of assembled fuel elements were found to be significantly reduced by welding the end caps prior to blending.

Examination of the irradiated fuel specimens from capsules NAA-53-1 and -2 showed a diametral increase of 1.3% to 1.9% at a burnup of about 0.4 metal at. %.

C. COMPONENT DEVELOPMENT

A series of tests was run on developmental control drum bearings. These tests included thermal cycles, shock vibration and temperature-vacuum. The condition of the bearings after testing appeared to be quite satisfactory.

A conceptual design of the control and safety components was prepared and detailed drawings are being made.

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The logic for the SNAP 8 controller has been worked out and a "bread-board" model set up to check out the scheme.

A method has been devised for casting the two-region LiH shield.

D. S8ER

The critical loading of the S8ER was calculated to be 160 fuel rods in the critical assembly filled with lucite rods.

The control drum support and bearing system was redesigned to eliminate the upper bearing adjacent to the hot (outlet) end of the core. Developmental test data indicated that performance of this bearing would be marginal at 1300°F.

A program is underway to develop a suitable coating to protect the beryllium reflectors from destructive oxidation in the S8ER environment. An anodized coating prepared by Brush Beryllium Co. has shown the best oxidation resistance to date.

The experimental program for the testing of the S8ER has been detailed.

E. S8DS

The test program for the SNAP 8 Ground Prototype Test (S8DS) has been outlined and the basic system descriptions written.

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III. DESIGN DESCRIPTION

A. SUMMARY

The SNAP 8 Nuclear System consists of a shielded compact nuclear reactor moderated by zirconium hydride and cooled by liquid NaK-78. The reactor is controlled by movable beryllium reflectors whose position controls the amount of leakage flux. The current design weight objective of the nuclear system is 450 lb not including the shield. The shield weight is strongly dependent on the payload location and size. Each major component is described below.

B. FUEL ELEMENTS

The SNAP 8 reactor core contains 211 fuel elements made of Zr - 10 wt % U alloy hydrided to a hydrogen content of 6×10^{22} hydrogen atoms/cc of $(U-Zr)H_x$. The uranium is 93.15% enriched. The elements are clad in 10-mil-thick Hastelloy N tubing coated internally with an enamel hydrogen-barrier coating. The active length of the fuel element is 15.5 in. The dimensions and temperatures are given in Table II.

Preliminary data from the fuel element irradiation experiments necessitated a change in the reference design during this report period. Radiation damage caused swelling of the fuel elements. Therefore, the unclad fuel rod diameter must be made smaller to provide space inside the cladding to accommodate the fuel expansion. The decrease in rod diameter decreases system reactivity unless the rods are made longer. Hence the length of the active core has been increased. The increased radial gap between the unclad fuel rod and the cladding tubes results in higher fuel rod and cladding temperatures.

C. CORE VESSEL

The core vessel is a cylinder of 316 stainless steel having a reverse dished head at the inlet coolant end and a torispherical head at the outlet coolant end. The core vessel contains an upper grid plate of 316 stainless steel, a lower grid plate of Hastelloy C, internal BeO, Hastelloy-N-clad reflectors, a coolant-flow baffle plate, and, of course, the 211 fuel elements previously described.

The reference design of the core vessel has changed during this report period to account for the increased fuel element length. In addition, the means of assembling the core has been changed to assist in re-entry dispersal. The

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TABLE II
SNAP 8 NUCLEAR SYSTEM CHARACTERISTICS

Operating Conditions		Nuclear Parameters	
Reactor Thermal Power, kwt	300-600	Average Thermal Flux, n/cm ² -sec	4x10 ¹²
Net Electrical Power, kwe	30-60	Mean Fission Energy, ev	0.20
Operating Life, hr	10,000	Average Isothermal Temperature Coefficient, $\Delta/\text{°F}$	-0.20
NaK Outlet Temperature, °F	1300	Uranium Loading, kg U ²³⁵	6.6
NaK Inlet Temperature, °F	1100	Mean Prompt Neutron Lifetime, -sec	6.7
Operating Pressure, psia	25	Burnable Poison Material	Sm ₂ O ₃
Maximum Fuel Temperature, °F	1490-1550	Reactivity worth, % $\Delta k/k$	2.6
Maximum Cladding Temperature, °F	1450-1465	Initial Cold Excess Reactivity, % $\Delta k/k$	6.1
Reactor Design		Total Reactivity Losses, % $\Delta k/k$	
Fuel Elements		Xenon	1.3
Number	211	Samarium	0.9
Fuel diameter, in.	0.529	Temperature and power	1.8
Cladding diameter, OD, in.	0.560	Hydrogen loss	2.5
Cladding thickness, in.	0.010	Burnup	0.4
Fuel alloy, wt % in Zr	10	Fission product poisons	0.8
Degree of hydriding, N _H (10 ²² a/cc)	6.0	Contingency	1.0
Cladding material	Hast-N	Total Control Drum Worth, % $\Delta k/k$	14.0
Active fuel length, in.	15.5	Shield Parameters	
Internal Reflectors		Material	
Composition	BeO	Neutron shield	LiH
Cladding	Hast-N	Thermal and/or gamma shield	W
Inlet Plenum Grid Plate		Dimensions, Neutron Shield	
Material	Hast-C	Maximum diameter, in.	40
Thickness, in.	5/16	Minimum diameter, in.	31.4
Outlet Plenum Grid Plate		Height, in.	28.6
Material	316 ss	Dimensions, Thermal and/or Gamma Shield	
Thickness, in.	11/32	Diameter, in.	9.5
Core Vessel		Height, in.	1.0
Internal diameter, in.	9-3/16	Neutron Shield Cladding	
Length, in.	21.6	Material	316 ss
Thickness, in.	1/16	Thickness, in.	0.109
Material	316 ss	Maximum Dose at Payload	
Reflector Control Elements		Neutrons, nvt	10 ¹³
Number	6	Gammas, rad (C)	10 ⁷
Material	Be		
Nominal thickness, in.	3.0		

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fuel elements were previously bolted to the lower grid plate. They are now just positioned by the grid plates so that during re-entry when the tie rods which hold the grid plates together burn away, the fuel elements are free to disperse.

The core vessel dimensions are given in Table II.

D. CONTROL AND REFLECTOR SYSTEM

The control reflector system is made of beryllium in two halves so that when the six control drums are fully inserted it forms a hollow right circular cylinder completely surrounding the active core length. The nominal beryllium thickness is 3 in.

The reference design of the control system has not changed significantly in the physical sense except that its overall length has increased. The operation of the control system during long-term control has been altered to provide for a step in reactivity once every 2-1/2 min rather than once every 1 min. This improves the system stability.

E. SHIELD

The reference shield design during this report period has been altered considerably. The thermal shield has been eliminated as the shielding was designed only for the 300-kwt system.

The present shield design now contains a gamma shield, a 1-in. plate of tungsten, located directly below the core inlet plenum, and a LiH neutron shield. The shield now is designed to shield the payload to 5×10^{12} nvt direct and scattered from the nuclear system and 9×10^6 rad (C) gamma direct and scattered from the nuclear system. The LiH neutron shield is a two-section shield containing 99.99% Li^7H at the hot end (end nearest reactor) and natural LiH at the cold end. The shield is designed to shield a payload 8 ft in diameter having a 15-ft separation between the payload and the bottom of the reactor vessel.

F. ANALYSIS

1. Nuclear Analysis

An extensive series of ULCER cases was run to determine whether interface samarium poisoning could be used to reduce the reactivity of the SNAP 8 reactor to below criticality if it dropped into water as a result of a

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launch abort. The experimental correlation of the ULCER program used in this study is uncertain as of this writing and hence it is too early to draw any firm conclusions.

Calculations were performed to determine the reactivity effect of additional amounts of burnable poison over the lifetime of an S8DS or SNAP 8 flight system. The KINDLE burnout program was used with the fast group cross sections obtained from the FORM program and the thermal group cross sections obtained from spectrum-averaged ULCER cross sections. Reactivity as a function of lifetime was obtained for the nonprepoisoned case, the design prepoison case, and double design prepoison case. Doubling the design poison was found to decrease the total reactivity swing from about $2.8\% \Delta k/k$ to about $1\% \Delta k/k$. However, the greater amount of prepoisoning would reduce the possible number of rods which could fail without causing subcriticality. For this reason, the design prepoison amount (same as used in S8ER) is unchanged.

Calculations were made to determine the number of drum void inserts needed to ensure that the SNAP 8 reactor would be subcritical if dropped into water during handling at the launch site prior to actual launch. With the drums at 105° (0° is the fully inserted position), it was calculated that four and five inserts of 2% natural boron in polyethylene would result in multiplication factors of 0.97 and 0.96 respectively when dropped into water. The core vessel was assumed to remain watertight. If the drums were at 180° (maximum out position), only three inserts were needed for a k of 0.97. However, current waterdunk experiments indicate that ULCER, the code used in these calculations, may not be able to accurately calculate the effects of a surrounding water blanket. At present, it appears that the multigroup diffusion programs may be calculating too low a multiplication factor for such systems. For this reason, a decision was made to use inserts in all six control drum voids. The polyethylene inserts were also ruled out due to possible fire hazards. New calculations are underway on 3% boron steel inserts.

2. Thermal Analysis

The major emphasis in the thermal analysis program during this report period was to provide temperatures at which various control system components would operate in the S8ER, S8DS, and in outer space. Also of particular concern was the beryllium temperature in these facilities due to the beryllium corrosion problem.

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Based upon preliminary fission gas release data from the irradiation samples, the maximum cladding creep due to the pressure of the fission gas and the hydrogen is less than 0.5% creep in 10,000 hr. The maximum tensile stress in the blend due to these gases is 150 psi.

The maximum ceramic temperatures for the S8ER rods as a function of hydrogen gap were calculated. For 600 kw and ideal operating conditions, the maximum ceramic temperature increases from 1325 to 1360°F for 0 to 10 mils of radial hydrogen gap, and from 1397 to 1500°F for equivalent worst (all hot channel factors) operating conditions.

3. Shielding Analysis

Major effort during this report period has been directed towards establishing the new design of the shield based upon the new criteria, i.e., the 8-ft-diameter payload located 15 ft from the reactor. In addition, NASA further stated a requirement that the direct dose at the payload be 5×10^{12} nvt from fast neutrons and 9×10^6 rads from gammas. During this report period, a split shield using Li^7H and natural LiH was adopted. These new ground rules added about 4 in. to the thickness of the shield and increased the maximum shield diameter to approximately 40 in.

Calculation were made to determine the effect of capture gammas from the beryllium reflectors at the payload. The capture gammas contribute less than 3% of the design dose of 10^7 rads and hence the gamma shield is only required over the diameter of the core and does not have to shield the reflectors.

A study was made to determine whether the two NaK pipes which pass by the shield should be located close together or on opposite sides of the shield. It appears that although the local dose is somewhat higher for the two pipes side by side, less of the payload is exposed to an overdose (greater than 5×10^{12} nvt) than in the case of the two pipes on opposite sides of the shield.

In addition, various shielding studies were conducted in support of the S8ER Summary Hazards Report* and in reviewing the design of the S8DS facility.

4. Control Analysis

During this report period a stability analysis of the S8ER was initiated. The open loop stability analysis has indicated that the S8ER is generally stable. The closed loop stability analysis is in progress.

*NAA-SR-6958, "SNAP 8 Final Safeguards Summary Report."

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SNAPTRAN 8A, a digital code to predict the power and temperature transients for the SNAP 8 reactors to large step inputs of reactivity (greater than one dollar), was written.

An analog model to simulate the on-off dead-band step controller without using relays was developed. Analog studies were run to determine the effect of addition and removal of the parasitic load, of slow drift out of the dead band, and of power conversion system startup. In general, these studies indicated that the long-term control drum stepping rate should be increased from 60 to 120 sec or greater. These studies are still continuing.

NAA-SR-MEMO-7233, SNAP 8 Reactor Control System, was issued. The report describes the SNAP 8 control system and the reasons behind the selection of the present concept.

G. NUCLEAR SYSTEM WEIGHT BREAKDOWN

The current design weight of the nuclear system is 1440 lb. A breakdown of the weight is given in Table III.

TABLE III

NUCLEAR SYSTEM WEIGHT SUMMARY (lb)

Reactor Assembly		408
Fuel and cladding	187	
Core vessel	16	
Internal reflectors	8	
NaK in core	} NaK	13
NaK in plenums		
Grid plates and flow baffles	8	
Piping, tie rods, etc.	12	
External Be reflector	122	
Support structure	27	
Bottom ring	15	
Shield Assembly		910
Lithium hydride	674	
Thermal shield	48	
Can	125	
Mating rings, misc. structure, cabling, etc.	63	
Electromechanical Hardware		122
Control drives, safety devices, etc.	72	
Misc. instruments	50	
TOTAL		1440

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IV. COMPONENT DEVELOPMENT

A. FUEL ELEMENT DEVELOPMENT

1. Objectives

The objective of this project is to develop fuel materials and fuel fabrication processes, and to fabricate fuel elements which will fulfill the performance requirements of the ultimate SNAP 8 flight reactor system. The immediate task is to develop and fabricate satisfactory fuel elements for demonstration of the SNAP 8 concept in the SNAP 8 Experimental Reactor (S8ER).

2. Materials Development

a. Fuel Alloy

Considerable effort during the reporting period has been devoted to a study of the effect of carbon additions on the properties of SNAP 8 fuels. Addition of a nominal 0.5 wt % carbon to fueled zirconium hydride has been found to be extremely beneficial to grain refinement of the hydrided alloy. The reduction in grain size makes the material less susceptible to cracking and thereby the production yield is increased. Current yield of uncracked hydrided rods is in excess of 90%.

Initial experiments on hydrogen dissociation pressures of SNAP 8 fuel compositions have indicated that the carbon addition produces about a 60% increase in pressure. It is believed that this increase in pressure is due to zirconium atoms being partially tied up by carbon atoms to form zirconium carbide. Consequently, there are fewer Zr atoms to accept hydrogen and the H/Zr atomic ratio for an equivalent N_H is effectively increased. Recent experiments on high H/Zr compositions have not been consistent with previous studies, and, in individual cases, have indicated the dissociation pressure to be higher than that reported previously. The experimental procedures are now under review in an effort to resolve the differences which have been observed. Possible factors contributing to this variation in results are: (1) contamination of the samples with the subsequent formation of gaseous impurity products, (2) reaction of hydrogen with carbon to form gaseous hydrocarbons, and (3) analytical error during the hot extraction process due to only partial release of hydrogen from the sample. At these high hydrogen compositions, small analytical errors in hydrogen analysis will lead to gross errors in the p.c.t. measurements because

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pressure changes are very sensitive to hydrogen composition changes. An accelerated effort is being placed on this problem in order to resolve these differences and to determine the true effect of the carbon additions on the modified fuel.

b. Hydrogen Barrier

(1) AI 8763D Coating

Most of the effort on the AI 8763D coating during the report period has been directed toward studying and evaluating the structural, chemical, and hydrogen permeability changes produced as a function of the time the coating was exposed to hydrogen at elevated temperature. In much of this study, parallel experiments were conducted on Solaramic S14-35A coatings. (The AI 8763D coating is a modification of the Solaramic coating differing in that certain less thermally stable oxides have been eliminated.)

Results of aging AI 8763D and Solaramic S14-35A coatings in hydrogen at 1400°F and 1500°F can be summarized as follows:

- 1) Devitrification of the glass coatings was rapid; only minor changes in appearance and x-ray diffraction patterns were noted at aging times beyond the initial 100 hr.
- 2) Porosity bubbles increased in size with time. After 500 hr, bubbles in the coating began to coalesce and after 1300 hr the coalesced bubbles were of a diameter one-half to two-thirds the thickness of the coating.
- 3) The coating adherence to chromized Hastelloy N seems to be enhanced by the H₂ aging.
- 4) Thermal expansion of AI 8763D coating is slightly higher than that of Solaramic S14-35A. On aging in H₂ at 1500°F, the expansion of both coatings increased for about 50 to 100 hr and then decreased with time at decreasing rate (see Figure 1).

The bubbles formed in the coating on aging are expected to have a major influence on the ability of the coating to retain hydrogen. Consequently, to gain further insight into bubble formation, experiments were conducted to determine the influence of adsorbed gases and possible hydrogen reduction of some oxides in the coating on bubble formation.

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Figure 1. Thermal Expansion of S14-35A and AI 8763D Coatings

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Application to Hastelloy N tubes of a coating made from vacuum outgassed (VOG) 8763 frit revealed that the vacuum outgassing increased the firing temperature by 100°F (probably because of devitrification of the glass during vacuum outgassing at 1500°F). It was also found that the VOG coating could be fired in air on nonchromized Hastelloy N tubes. Regular 8763D coatings cannot be fired in air on nonchromized Hastelloy N tubes because the molybdenum in the metal substrate oxidizes and causes eruptions in the coating. These studies indicate that a large part of the oxygen which contacts the metal substrate comes from adsorbed gases in the coating and not from the furnace atmosphere. Since oxygen also promotes the bond between ceramic and metal, the adherence of the VOG coating was poor. Environmental tests, however, showed that adherence was improved after aging at 1500°F in hydrogen. Metallographic examination of the VOG coating indicated that the bubble structure is greatly reduced by the vacuum outgassing operation.

Another approach was used to obtain a gas-free coating. This involved the formulation of 8763D coating from chemically pure oxides. The largest source of dissolved and adsorbed gases in the frit is from the technical grade carbonates, nitrates, and hydrates used as raw materials. To eliminate this source of gas in the coating, a batch of AI 8763D was made from chemically pure oxides. These materials were mixed well and calcined before they melted and could trap gases. A fusion-flow button test showed that this glass softened and flowed exactly the same as normal AI 8763D. It was applied to stainless steel, Hastelloy N, and chromized Hastelloy N tubes which are now undergoing environmental testing in hydrogen at 1500°F.

To investigate the possibility of partial hydrogen reduction of the oxide coating contributing to bubble formation, samples of powdered 8763D coating and starting frit, both VOG and standard, were sealed in quartz ampoules in hydrogen and heated for 1 hr at 1500°F. The gas was then recovered from the ampoules and analyzed in a mass spectrometer. The results, given in Table IV, support the speculation that the enamel may be partially reduced in hydrogen. Similar experiments are in progress in an argon rather than hydrogen environment to determine the extent to which hydrogen reduction plays a role in producing gaseous products.

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TABLE IV

RESULTS OF HEATING AI 8763D COATING MATERIAL IN HYDROGEN

Material Code Name*	Treatment Material Received Before Test	Gases Detected (mol or vol %)					
		H ₂	CH ₄	H ₂ O	H ₂ S	CO ₂	CO
XD Slip	Sintered at 2300°F	Detected	0.1	32.3	1.3	64.6	1.6
VOG-2 Slip	Applied to SS tube and removed	0.8	Detected	10.9	0.3	86.5	1.4
XD Slip	Removed from production tube	1.8	0.2	53.9	-	43.0	1.2
XD Frit		3.5	0.6	3.3	1.8	88.6	2.2
VOG-2 Frit		9.4	1.0	2.7	1.4	65.6	20.0
VOG-2 Slip	Remelted and Pt added	16.0	19.0	16.0	-	48.0	-
XD Slip	Aged in H ₂ at 1500°F, 1500 hr	33.9	41.9	0.4	-	0.8	22.9
Quartz	Blank ampoule	99.4	0.1	0.1	-	-	0.4

Code names: XD = Standard 8763D

VOG-2 = Second batch of vacuum outgassed 8763D frit

Typical results from measurements of permeation rate of AI 8763D coating as a function of time at temperature are shown in Figure 2. With additional time, the permeation rate decreases to values on the order of 10^{-4} to 10^{-3} cc/hr-cm² after several hundred hours at 1300°F.

Hydrogen permeation rate has also been studied as a function of hydrogen pressure at several temperatures. The results, summarized in Figure 3, show that the permeation rate, ϕ , is approximately directly proportional to the pressure. This means that the limiting step in the permeation process occurs prior to dissociation of the hydrogen molecule (i.e. in the enamel or at the enamel interface). Since the pressure-permeation relationship becomes less linear as the temperature increases, it appears that a temperature effect is also present.

(2) New Coatings

Based on an extensive literature survey, two quaternary oxide systems were selected for investigation. The compositions of the two initial coatings are listed as follows:

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Figure 2. Hydrogen Permeation Rate of Al 8763D Coating

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Figure 3. Pressure-Temperature-Permeation Profiles

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<u>MAST-2</u>	<u>STAB-1</u>
47.7% SiO ₂	43.8% SiO ₂
23.4% Al ₂ O ₃	13.2% Al ₂ O ₃
15.4% MnO	30.2% BaO
13.5% TiO ₂	12.3% TiO ₂

In each system, the TiO₂ was added as a nucleating agent. The glassy coating is applied to the metal substrate and then the coating is subjected to a controlled nucleating process which completely crystallizes the coating. Crystallization produces a more compact structure (better hydrogen barrier) and increases the thermal stability of the material.

A nucleation procedure has been worked out for MAST-2 coating as follows:

- 1) Heat at 600°F/hr to 1400°F.
- 2) Hold at 1400°F for 1 hr to nucleate.
- 3) Heat at 600°F/hr to 2000°F.
- 4) Hold at 2000°F for 1 hr and cool.

Using the above cycle, a thermal expansion specimen was prepared and measured using a quartz dilatometer. Figure 4 is a comparison of MAST-2, AI 8763D, and Hastelloy N thermal expansion.

An x-ray analysis was made on MAST-2 before and after crystallization. The sample was amorphous prior to heat treatment and contained the following crystals after heat treatment: Spessarite Mn₃Al₂(SiO₄)₃, Rutile (TiO₂), and possibly MnO₂, MnO, and SiO₂ (α-cristobalite).

MAST-2 was then smelted up in a larger batch, fritted, and ball-milled with the following mill addition to form an enamel slip:

MAST-2 frit	100 parts
Green Label Clay	6 parts
Bentonite	1/4 part
KNO ₂	1/8 part
Gum Tragacanth	1/16 part
Water	58 parts

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Figure 4. Thermal Expansion of Hastelloy N Cladding and of AI 8763D and MAST-2 Coatings

The optimum firing time and temperature was determined to be 30 min at 2300°F. The coating fired out fairly glossy, but turned to a matte surface after crystallization. No thermal expansion difficulties have been encountered and no typical enameling defects were noted.

Samples of the new coating on 347 stainless steel, Hastelloy N, and chromized Hastelloy N tubing are presently being soaked in a hydrogen furnace at 1500°F for 500 hr. Results will be reported next quarter.

As the firing temperature of MAST-2 is so high (2300°F), a new composition is presently being smelted that should fire considerably lower. The new composition is:

MAST-3	
SiO ₂	43.4%
MnO	25.5%
Al ₂ O ₃	16.1%
TiO ₂	15.0%

The STAB-1 formulation has been found to readily coat stainless steel, Hastelloy N, and chromized Hastelloy N. Optimization of the crystallization

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parameters and environmental testing of this coating are planned for the next quarter.

Subcontracts have been established with both Battelle Memorial Institute and Stanford Research Institute for the development of a stable high-temperature hydrogen-barrier coating for SNAP reactors. BMI will work on highly crystallized coatings of the enamel type, while SRI will attempt the vapor deposition of thin, dense ceramic oxide films.

c. Cladding

Development work has been started on Hastelloy R-235 alloy as a possible substitute for Hastelloy N as a cladding for SNAP fuel rods because of its higher strength and lower molybdenum content. Two programs of mechanical testing of Hastelloy R-235 have been initiated. The first is to determine long-time creep-rupture properties. Samples have been prepared and actual testing will be initiated in the early part of next quarter.

The second program involves the effect of aging R-235 in a hydrogen atmosphere at 1500°F. In this test, tensile samples of Hastelloy R-235 were placed in a closed mullite retort which was subsequently evacuated, heated to 1500°F, and back-filled with one atmosphere of hydrogen. These samples will be maintained in this environment for about 500 hr, then removed to determine mechanical strength and stress-strain properties.

Tubing of Hastelloy R-235 was successfully coated with AI 8763D fired in air. The adherence was not as good as expected, but no chromizing was required. It was noted, however, that after samples were aged in hydrogen at 1500°F for 500 hr, the metal became quite brittle.

d. Compatibility Studies

Compatibility studies were made under SNAP 8 conditions on several materials to be used as core components. Samples of carbon-modified SNAP 8 fuel were tested in contact with Hastelloy N and AI 8763D hydrogen barrier coating. In a test at 1450°F for 320 hr, the 8763D coating was found to be completely compatible with the fuel. However, the fuel showed a reaction with bare Hastelloy N. A diffusion zone of 2.5 mil was observed. A second compatibility test of the same materials of 1000-hr duration at 1450°F has just been completed and the results are being evaluated.

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e. NaK-H Studies

Studies on the solubility of hydrogen in potassium and NaK have continued. Dissociation pressure isotherms at 100°F, shown in Figure 5, indicate that the solubility limit of hydrogen in K is about 0.02 wt % hydrogen while the limit of solubility in NaK is 0.08 wt %. The graph also shows that the dissociation pressures experienced in both the potassium and NaK systems are the same within experimental error. The conclusion from this observation is that the dissociation pressures in the NaK-H system (at hydrogen compositions up to 0.20 wt %) can be attributed to the interaction between hydrogen and potassium rather than between hydrogen and sodium. Further evidence of the reaction of hydrogen with potassium in the NaK system is shown by the equations developed for the hydrogen plateau pressures. Consider the general equation:

$$\text{Log } P_{\text{atm}} = A - B/T(^{\circ}\text{K}) ,$$

then,

System	A	B	Reference
NaK-H	4.10	4864	AI
K-H	4.03	4623	AI
Na-H	10.47	5958	Banus, et al, JACS, 77, pp 2007-2010

Figure 5. Dissociation Pressure Isotherms at 1000°F

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3. Fuel Fabrication Process

The major activities in process development during the quarter were: (a) development of assembly process to yield elements with satisfactory dimensions and hydrogen permeation rates, and (b) improvement of welding techniques for the blind-end "H" caps.

The question of radial clearance between the fuel rod and cladding is critical from the standpoint of reactor design and fuel element assembly. To achieve the maximum heat transfer capabilities, it is preferable to have as small clearance as possible. From the standpoint of ease of fuel element assembly, it is desirable to have as wide a gap as possible. The problem of assembly of fuel elements which have small radial clearances was undertaken early in the quarter. Fifty-one test elements were assembled to determine the effect of radial clearance upon permeation rate of hydrogen due to possible damage to the ceramic coating on the interior of the cladding tube. The first few such elements showed unacceptably high permeation because of the necessity of honing the ceramic cladding in order to correct a taper in the coating thickness. In subsequent development, honing was eliminated by changes in techniques which provided a ceramic coating with minimal taper in thickness. Elimination of the honing operation and improvements in the rod insertion technique resulted in production of assembled fuel elements with leak rates below the maximum tolerable level of 0.7 cc/hr at 1300°F. All these developmental elements were fabricated with blended end-cups, but the welded end-cap was omitted in order to preclude any contributing effect of the welding operation upon permeation rate.

More recent tests show that the welding operation very probably disrupts the ceramic cup blend and results in a high permeation rate of hydrogen from the fuel element. Because of this, a series of test elements have been assembled recently in which the end-cap is welded in place prior to blending of the cup rather than after the cup-blending operation. At a permeation test temperature of 1400°F, approximately half of the elements assembled by blending after welding exhibited permeation rates in excess of 1.6 cc/hr. This, however, is a considerable improvement over earlier test specimens in which blending was performed before welding. In these tests, the permeation rates were generally in excess of 5 cc/hr.

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One of the most severe problems in SNAP 8 fuel-element assembly is due to the ovality of the tube after coating and the distortion of the tube due to nonuniformity of heating during chromizing or ceramic firing. Potential methods of straightening and removing ovality from tubing prior to insertion of the fuel rod are being evaluated.

Difficulties have been encountered with H-cap welding due to "drop through" of weld metal which resulted in a thicker ceramic coating in the weld region. This necessitated machining of the fuel rod on the end to permit it to bottom in the can. The welding of H-caps was improved with the use of a crown chill block, but even so, the technique requires a great deal of care to prevent "drop through."

Electron beam welding is known to be highly adaptable in operations where a minimum movement of material can be tolerated. Therefore, some evaluation tests were performed by Siaky, Hamilton Standard, and others to determine the capability of these machines and the desirability of obtaining one for use in this operation. While only a few samples have been welded as yet, it is apparent that electron-beam welding will probably provide a satisfactory solution to the problem. Procurement of electron-beam welding equipment as soon as possible is planned.

4. Testing and Evaluation

a. Environmental Testing

Additional 1300°F isothermal permeation testing of SNAP 8 developmental fuel elements was performed during the reporting period to investigate the effect of the assembly radial clearance between fuel and cladding. Because problems were being encountered in the welding of A-type end-caps, these elements were tested without such caps, and contained only the reference blend-cup as a seal at the top end. Table V summarizes the measured hydrogen permeation rates obtained for these elements. For the range of radial clearance investigated, the results indicated no statistically significant effect of clearance on permeability.

A number of developmental elements were maintained at 1300°F to determine the effect of time at temperature. Table VI summarizes the results of this investigation. Some elements had significantly lower permeation rates after only 100 to 200 hr at temperature.

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TABLE V

EFFECT OF RADIAL CLEARANCE ON HYDROGEN PERMEATION RATE
OF FUEL ELEMENTS TESTED WITHOUT END CAPS

Element No.	Radial Clearance (mil)	1300°F Isothermal Permeation Rate (cm ³ /hr)
8P-14*	2.0	1.60
8P-15*	2.3	0.72
8P-17*	2.6	2.15
8P-18*	2.7	0.31
8P-19	0.6	8.2
8P-20	0.5	0.46
8P-21	1.3	0.62
8P-22	0.4	0.55
8P-24	1.3	0.85
8P-25	0.6	2.32
8P-26	1.5	1.74
8P-27	0.8	0.47
8P-28	0.4	0.22
8P-29	2.3	0.17
8P-30	2.4	0.18
8P-31	2.4	0.27
8P-32	7.8	0.13
8P-33	2.3	0.13
8P-34	2.4	0.10
8P-36	0.5	0.87
8P-37	2.0	0.47
8P-38	1.8	0.22
8P-40	1.5	0.91
8P-41	0.4	0.44
8P-42	0.4	0.18
8P-43	4.8	0.11
8P-44	0.4	0.25
8P-45	4.7	0.10
8P-46	1.7	0.37
8P-47	2.3	0.11
8P-48	1.3	0.24
8P-49	1.3	0.69
8P-50	2.2	0.40
8P-51	0.8	0.27

*Ceramic coating honed prior to element assembly

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TABLE VI

EFFECT OF TIME ON HYDROGEN PERMEATION RATE OF
FUEL ELEMENTS

Element No.	Total Time at Temperature (hr)	1300°F Isothermal Permeation Rate (cm ³ /hr)
8P-10	10	0.70
	180	0.21
	250	0.22
	450	0.18
	500	0.38
	750	0.32
8P-27	15	0.44
	145	0.32
	270	0.49
8P-28	16	0.50
	90	0.23
	240	0.25
8P-51	28	0.58
	45	0.50
	117	0.49
8P-47	41	0.16
	60	0.12
	110	0.10
8P-36	16	0.83
	18	0.61
	139	0.20
	160	0.18

To investigate the process for welding A-type end-caps to SNAP 8 fuel elements, caps were welded to selected developmental fuel elements and permeation tests were then performed. Table VII compares the 1300°F isothermal permeation rates before and after welding. In addition, data are included for subsequent 1400°F and 1300°F isothermal testing.

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TABLE VII

CHANGE IN HYDROGEN PERMEATION RATES DUE TO WELDING
OF A-TYPE END-CAPS

Element No.	1300°F Permeation Prior to Welding Cap (cm ³ /hr)	After Welding		
		Initial 1300°F Permeation (cm ³ /hr)	Initial 1400°F Permeation (cm ³ /hr)	Subsequent 1300°F Permeation (cm ³ /hr)
8P-10	0.32	0.73	1.65	0.70
8P-27	0.47	0.08	2.07	1.16
8P-28	0.22	0.16	6.78	3.29
8P-31	0.27	1.21	3.81	1.50
8P-32	0.13	0.09	0.31	0.12
8P-36	0.87	0.24	6.68	-
8P-37	0.47	0.11	5.47	2.48
8P-40	0.91	1.42	-	-
8P-41	0.44	2.36	-	-
8P-43	0.11	0.85	2.69	1.11
8P-46	0.37	>>6.0	-	-
8P-48	0.24	0.38	>>7.0	3.62
8P-49	0.69	>>6.0	-	-
8P-50	0.40	1.03	3.99	1.67
8P-51	0.27	0.47	>>7.0	6.15

In the previous quarterly report, data were presented for 1300°F and 2 atm hydrogen internal pressure permeation testing of five S8ER-type* cladding membranes. The results indicated that aging for 500 hr at 1400°F effected a decrease (approximate factor of 10) in the 1300°F permeation rates obtained for 2 atm hydrogen internal pressurization of the membranes. During the current reporting period, the five membranes (Nos. 72, 74, 34, R, and 59) were manifolded to permit permeation testing of the cluster during a continued aging period at 1400°F. The cluster was pressurized with hydrogen to 2 atm, and mean cluster permeation rates were obtained at various intervals until the total time accumulated at 1400°F was 1500 hr. The test temperature was then increased to

*Three coating layers of AI 8763D on chromized Hastelloy N

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1500°F with the internal hydrogen pressure maintained at 2 atm. By the end of the report period, 50 hr at 1500°F had been accumulated and the test was being continued. Table VIII presents the mean permeation rates obtained for the cluster during the current report period.

TABLE VIII

MEAN HYDROGEN PERMEATION RATE OF FIVE S8ER
PROTOTYPE CLADDING-TUBE CLUSTER AT
2 ATM INTERNAL HYDROGEN PRESSURE

Test Temperature	Accumulated Time at Test Temperature (hr)	Mean Cluster Permeation Rate at Test Temperature (cm ³ /hr-cm ²)
1400°F	500	-
1400°F	546	0.68 x 10 ⁻³
1400°F	566	0.51 x 10 ⁻³
1400°F	638	0.51 x 10 ⁻³
1400°F	663	0.44 x 10 ⁻³
1400°F	689	0.50 x 10 ⁻³
1400°F	737	0.60 x 10 ⁻³
1400°F	903	0.64 x 10 ⁻³
1400°F	1021	0.44 x 10 ⁻³
1400°F	1338	0.46 x 10 ⁻³
1400°F	1500	0.50 x 10 ⁻³
1500°F	50	1.0 x 10 ⁻³

Note: During the 1500 hr period of testing at 1400°F, a total of 13 thermal cycles (1400 to 600 to 1400°F) were endured by the cluster.

b. Irradiation Experiments

The postirradiation examination of Capsule NAA-53-1 continued throughout the quarter at the Component Development Hot Cell (CDHC). Capsule NAA-53-2 arrived at the CDHC early in the quarter and examination of this capsule was begun. Both of these irradiation experiments contained two fuel compartments designed to operate at fuel central temperatures of 1250 and 1450°F. The design of the irradiation experiment using modified SNAP 8 fuel (Capsule NAA-53-3) was completed.

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The dimensional and density measurements made to date on the declad fuel specimens in NAA-53-1 and -2 are summarized in Table IX. The irradiation temperatures given in the table are estimated average fuel central temperatures and the burnup given is estimated from temperature data. Detailed reduction of irradiation temperature data and radiochemical burnup analysis are in progress.

TABLE IX
DIAMETER AND DENSITY CHANGES IN IRRADIATED FUEL

Capsule	Specimen	Estimated Average Central Temperature (°F)	Estimated Burnup (metal at. %)	Increase in Diameter (%)	Decrease in Density (%)
53-1	Upper	1285	0.4 to 0.5	1.3	0.8
53-1	Lower	1390	0.4 to 0.5	1.9	3.2
53-2	Lower	1355	0.9	4.2	4.6

All specimens were broken into at least two pieces after decladding so that accurate length measurements were not possible. Estimates were made of length changes from (1) approximate measurements of the broken pieces, (2) density and diameter measurements, and (3) diameter measurements and the assumption that the dimensional changes are isotropic. There is no agreement between the length estimates obtained by the three approaches.

Postirradiation examination of NAA-53-1 specimens is complete except for metallographic examination, burnup analysis, and hydrogen analysis. Equipment for the latter analysis is currently being assembled and checked out. Fission gas release (between fuel and cladding) was measured and compared with calculated values for a recoil release mechanism. Excellent agreement between experimental and calculated recoil release was obtained. On the basis of this agreement, the expected fission gas pressure inside the cladding of a SNAP 8 flight reactor fuel element after 10,000 hr of operation at 600 kw was calculated to be less than 20 psia for the probable minimum void volume.

Diameter measurements made on the fuel specimens from NAA-53-2 prior to decladding showed a 1.5% increase in cladding OD contrasted to no measurable increase in NAA-53-1 specimens. It had been planned to measure the hydrogen permeation from both NAA-53-2 specimens prior to decladding.

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However, a NaK leak into the fuel, apparently due to a crack in the weld on the lead-in for the central fuel thermocouple, was observed in the lower specimen. Therefore, this specimen has been de-clad and the dimensional and density measurements listed in Table IX were made. The clad upper specimen has been set aside until hydrogen permeation equipment can be set up in-cell.

The capsule design modifications for NAA-53-3 have been completed and parts are being fabricated. The fuel will be inserted as five 1/2-in. pieces instead of one 2-1/2-in. piece in the hope of preventing fracture during irradiation or disassembly and thereby obtaining more conclusive swelling measurements. The fuel has been machined to size for insertion into the capsule. It is expected that NAA-53-3 will be inserted in the MTR approximately July 1, 1962 and will remain until its burnup is 0.4 to 0.5 metal at. %.

c. Nondestructive Testing

An autoradiometric uranium analyzer has been designed and assembled for nondestructively measuring the uranium content and distribution in SNAP fuel rods, prior to cladding. The gage is a combination recording scintillation counter and recording rate meter which automatically scans fuel rods at the rate of about 3 rods per hour. The uranium content per inch of rod length as well as total uranium for the rod is indicated on two recording scalars, and circumferential distribution is graphically displayed by a recording rate meter. To date, all production fuel rods have been examined by this gage.

Two methods have been developed for rapidly identifying normal and enriched fuel elements, fuel rods, and raw materials. One method measures the intensity of the 0.18-Mev gamma radiation from U^{235} which is approximately 20 times greater for enriched alloy than for alloy containing normal uranium. This method has been employed for sorting finished fuel elements with the autoradiometric uranium analyzer. The second method measures the ratio of beta-to-gamma radiation from a given sample. In practice, two measurements of the sample are required with a thin window G-M counter, one reading with an open window and a second reading with a 1/16-in. aluminum filter (beta radiation absorber) placed between the G-M tube and the sample. The ratio of the first to the second readings for normal uranium samples is approximately 10 to 12:1 and for enriched samples is approximately 1.5 to 3:1. The proper use of these two methods will greatly reduce the possibility of erroneously intermixing items containing normal or enriched uranium.

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A special radiographic technique has been developed for examining H-cap welds for internal flaws and wall thinning. The technique employs a new fine-focal-spot x-ray machine operated at low kilovoltage, a new single-emulsion, fine-grain, high-contrast x-ray film, and precise angulation of the radiation beam. The resulting radiographic films have disclosed porosity as small as 0.002 in. diameter, and nonfusion and thinning as shallow as 0.0005 in. To aid in the interpretation of the radiographs, a special penetrometer gage was developed with slots and holes of varying depths. The depths of these "artificial defects" are now being routinely compared with natural defects in the H-cap welds during regular quality testing operations. To expedite the radiographic process, a new x-ray booth has been installed and successfully put into operation in the SNAP quality testing area.

B. FUEL ELEMENT FABRICATION

1. Summary

The scaleup from R & D to repeat fabrication of SNAP fuel elements proceeded during the third quarter according to the master schedule. The following table shows the general breakdown:

TABLE X

FUEL ELEMENT FABRICATION PRODUCTION PROGRESS

		Scheduled	Produced
Alloyed and formed rods	Normal	96	96
	Enriched	250	250
Hydrided rods	Normal	24	24
	Enriched	250	250
Welded tubes		370	360
Coated tubes and cups		360	330
Assembled elements	Enriched	250	75
Qualified elements*		145	0

*Qualified elements are elements which comply fully with S8ER fuel specification

Process capabilities, using existing equipment, have been demonstrated in the areas of melting and forming, cladding and assembly. Quality test results on completely assembled elements, which were obtained in the last month of the

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report period, revealed two major problem areas: (1) lack of uniformity of dimensions within the dimensional specifications, and (2) excessive hydrogen permeation rates. This situation is discussed in more detail below.

2. Fabrication Process

a. Hydriding

During the hydriding of the quantity of fuel elements indicated above, no cracked rods were produced. The addition of carbon as an alloying constituent has solved the problem of cracking fuel rods during hydriding.

Obtaining the required composition for SNAP 8 fuel rods is a minor problem. Continuous and improved adjustment of the hydriding cycle has resulted in an approximately 80% yield on composition after the first hydride cycle. The remainder of the fuel rods have been rehydrided and this step has produced the required composition.

b. Coating

During this period, a problem of the flaking off of the chromized layer on the SNAP 8 cladding tubing was encountered, and subsequently solved. The flaking action was first thought to be due to a cleaning compound in which the chromized tubes were polished prior to shipment from the supplier. However, after this step was dropped from the process, the tubes continued to react in the same manner. A cooperative investigation between AI and the vendor isolated the use of a contaminated chromizing compound as the cause. The vendor located another supplier for the compound and is presently producing chromized tubes which meet specification requirements.

Continued development has substantially eliminated the excessive running and pitting of the enamel slip in the SNAP 8 cladding tube during the coating and firing steps. This was accomplished by improvement in the drying rack used and an improvement in the drying procedure. Running was caused by jarring and rough handling on placing the tube into the drying rack, and was eliminated by slight modification of the rack to facilitate loading. Excessive pitting in the radius at the blind end of the tube has been greatly decreased by providing a heat shield between the heating element and the end-cap of the cladding tube to decrease the rate of drying in this portion of the tube. During a 3-week period following solution of the chromizing problem, approximately 175 SNAP 8 cladding tubes have been ceramic-coated and submitted to final inspection.

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Thus, the rate of coating is approximately 12 tubes per day. At present more equipment is being received and put in operation which should increase coating capability to 20 to 25 tubes per day.

No problem of appreciable consequence has been noted in ceramic coating of the SNAP 8 end-cups. A refinement in the handling procedures for these cups has largely eliminated the chipping previously noted on the outside diameters.

c. Cladding Materials

The SNAP 8 fuel element is being clad in seamless INOR 8 tubing. Approximately 1700 ft of this material is available and negotiations are in process for backup commitments on this long lead-time item.

d. Welding

Welding of the blind-end-cap continued at a lower than anticipated yield through the quality-control acceptance cycle. The primary causes for rejection were thinning in the weld zone and incomplete fusion.

Experimental work in an attempt to alleviate the thinning has resulted in a redesign of this end-cap to incorporate the addition of an approximately 0.015-in. extension on the edge of the end-cap, providing a "lip." This lip is intended to serve as filler metal for the weld and is expected to prevent base metal from being drawn into the interface during fusion. Development work with this concept is being continued.

Incomplete fusion is being corrected by a continuing tightening of welding parameters; i.e., electrode positioning, gap between electrode and work piece, and turntable synchronization.

Welding capability has been improved by installation of a new 15-amp welding unit, an automated turntable, and an inert gas glovebox.

e. Blending

Mechanical fit-up problems have been experienced during the blending operation of the SNAP 8 elements, partly due to the chipping of the coating of cups mentioned above and, in part, due to the assembly techniques being developed. As a remedy, an improved collar guide for the inserting of the cup has been designed and is in use. This design improves the rigidity of the press mechanism used for the insertion of the blend cup into the coated tube.

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f. Quality Assurance

A planned quality assurance program is in operation which makes it mandatory that all parts going into the final fuel element must meet quality requirements at specific points within the process prior to being up-graded to the next step. The history of each fuel element is permanently recorded from the basic alloying constituents through to the complete final dimensions and leak-check.

The general quality control procedures have been outlined in a Quality Control Manual. Further procedures for specific types of contracts and a procedure for the control of vendor-supplied material are in the draft stage.

Finished fuel elements and components are inspected radiographically with a 110-kv Picker x-ray unit. Uranium concentration data is being obtained with a newly acquired autoradiometric uranium analyzer. A program has been developed to destructively sample fuel rods and coated tubes, and is in use. Outside sources are being explored to obtain increased capacity for fuel alloy production. Ingots made by Nuclear Metals, Inc., with induction melting and hot extrusion techniques, have been evaluated and found to be generally satisfactory. With further testing of this material, important data can be determined concerning the limits of the carbon content and alloy composition control as a function of type of melting and extrusion techniques.

The quality testing program has revealed dimensional control and hydrogen permeation deficiencies in the assembled elements. These problems are currently causing a delay in the schedule for producing S8ER fuel elements. The most serious dimensional deficiency is in the total indicated reading (TIR) of the assembled element. Improvement in the outside diameter and ovality of the element is also required. A program is in progress to determine the distortion produced in the cladding at the various process stages and to select the best procedure for reducing the distortion or providing a straightening step or steps in the process prior to final assembly.

Also in progress, is a program to determine the source and nature of the high hydrogen permeation rates measured in the elements assembled to date and to establish process modifications which will reduce the permeation rates. A preliminary evaluation indicates that blending the end-cups after welding the A-cap may result in significant improvement. As techniques are developed to prevent distortion of the cladding, the mechanical abuse to the coating during rod insertion will be reduced.

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C. REFLECTOR AND DRIVE DEVELOPMENT

1. Self-Welding Tests

Phase II of the high-temperature, high-vacuum study on material self-welding continues. The first ten combinations have been evaluated. The testing to date has considered the adhesion as a function of time, temperature, and unit (apparent) area loading. Results indicated that time and temperature are the dependent variables. Adhesion has occurred in some cases only after many hundreds of hours of interface contact. This same condition can be achieved in a shorter time by increasing the temperature. The area loading is the indeterminate variable as it depends on surface finish, flatness, modulus of elasticity, emissivity, etc.

Other significant results substantiate the "diffusion" theory for self-welding. Beryllium did not stick to beryllium after 686 hr of surface contact at 100 psi load. On the other hand, beryllium has exhibited adhesion to 316 stainless steel, 304 stainless steel, Al_2O_3 , and BeO. The diffusion theory states that the activation energy required to cause interface diffusion is greater for "self-diffusion" than for diffusion of "foreign" atoms.

The facility for the self-weld program is being expanded to include 20 independent test chambers, each capable of vacuums of 10^{-9} Torr at temperatures of 1000°F . This environment can be maintained as a closed system for indefinite periods. Some facilities have been holding these conditions for six months.

2. Springs

A spring relaxation test for operation at 1000°F for 1000 to 10,000 hr has been planned. Test hardware design, manufacture, setup and checkout has been completed and is awaiting delivery of the test springs. Special springs holding devices are being designed for nuclear irradiation damage study tests.

3. Reflector Drum Bearings

A series of tests were run on developmental drum bearings. These tests included five thermal cycles from 200 to 1000°F , 20 g shocks, vibrations to 15 g at 3000 cps, and temperature-vacuum-torque tests in excess of 300 hr. The resultant torque of this testing was less than design maximum. A plot of vacuum, temperature, and torque vs time is shown in Figure 6.

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Figure 6. Control Drum Bearing Set Temperature-Vacuum Torque Test No. 1

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The shock and vibration fixture with simulated drum load and mounting is shown in Figure 7. The bearings are shown in the unused condition and post thermal cycling condition in Figures 8 and 9. The temperature-vacuum torque test setup is shown in Figures 10 and 11. The condition of the bearings after complete testing appeared quite satisfactory, and is shown in Figures 12 and 13.

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Figure 7. Control Drum Bearing
Vibration Test Fixture

4. Reflector-Control Configuration

Following a design concept study on possible reflector-control configurations, a conceptual design was prepared (Figure 14) and detail drawings are being prepared for the bearings, gears, and reflector parts for acceptance testing on a complete developmental reactor system (non-nuclear). All of the components used in this design have been tested for proof of principle.

5. Reactor Safety Components

A design concept layout is being prepared for a safety system which will assure safety for ground personnel who will handle the reactor and prepare the control system for launch. This system will include boron filler blocks which are bolted into the control drum cavities to prevent possible accidental driving-in of the drums and resultant nuclear hazards, and a key lock for each drum, which will be used in the final readying operation before launching the reactor.

Development work was also continued on the following devices:

a. Differential Thermal Expansion Device

A differential thermal expansion device, which consists of a low coefficient of expansion tube and a high coefficient of expansion rod is under development. Upon descending temperature, differential negative expansion will break a tension bolt made from a highly brittle material. This bolt constitutes the connection of fittings of the reflector band, and upon breakage, the two halves of the reflector band will permit dispersal of the reflectors, thus rendering the reactor subcritical.

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Figure 8. Lower Control Drum Bearing Before and After Thermal Cycling Test

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Figure 9. Upper Control Drum Bearing Before and After Thermal Cycling Test

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Figure 10. Control Drum Bearing Torque Test Apparatus

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Figure 11. Control Drum Bearing Torque Test Fixture

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Figure 12. Upper Control Drum Bearing After Complete Testing

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Figure 13. Lower Control Drum Bearing After Complete Testing

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Figure 14. Nuclear System Design Layout

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A number of tests were conducted with various materials of the tension bolt. H-11 tool steel was selected as the material most brittle in the temperature range, with A286 steel as a second choice. The shape of the stress concentration was selected as a 60° VEE notch with an A/R of 30. Test results are given in Table XI.

A new design layout was made incorporating a spring engagement device which will be activated only after the design temperature of the reactor has been reached. The band fittings were redesigned to eliminate transfer of bending and shear loads to the tension bolt.

b. Reflector Band-Fitting Disconnect Squib

An alternate approach for disconnecting the reflector band fittings by means of a squib was examined. Both explosive-bridge-wire (EBW) and pyrofuse squibs were considered and bids of experienced subcontractors were received. The comparison is presently leaning towards adoption of the pyrofuse system with a completely self-contained pyrotechnic-mechanically initiated piezoelectric system. This is much lighter than the EBW system, which, moreover, required extraneous electrical sources. Further evaluation of these systems is contemplated with special emphasis on irradiation tests of component samples when received.

Initiation of squib firing may be effected by several methods which are of parallel nature:

- 1) Clock Mechanism: After one year of operation a clock mechanism, driven by either mechanical or electrical power, will drive a firing pin into the primer cap, thus starting the voltage which will generate the impulse for the squib.
- 2) Telemetry: In case of any malfunction of the vehicle or of the reactor at any time, a telemetered impulse will drive a firing pin into the primer cap, as above.
- 3) Low Temperature Sensitive Device: If the reactor should fail to become critical for any reason and the temperature of the APU drops below 40°F, a temperature sensitive bimetal strip will deflect to unlock a spring mechanism which drives a firing pin into the primer cap, as above.

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TABLE XI

TENSION BOLT MATERIALS TEMPERATURE TESTS

Material	Notch Shape					Bolt Diameter (in.)	Temperature to Break, Cool from 900°F (°F)
	Neck (0.027-in. wide)	Rectangle (0.036-in. flat)	Square (0.025 x 0.095 in.)	Hole (0.062 in.)	VEE (60°)		
Brass					x	0.055	470
Brass					x	0.060	540
Brass		x				0.190	760
Brass			x			0.190	405
Brass				x	x	0.080	465
Brass					x	0.060	500
Brass					x	0.060	335
Brass					x	0.060	485
Molybdenum	x					0.060	705
Tungsten	x					0.040	605
A-286 Steel					0.010R x	0.060	450
A-286 Steel					0.010R x	0.060	520
A-286 Steel					0.010R x	0.057	560
H-11 Tool Steel					0.010R x	0.054	650
H-11 Tool Steel					0.0015R x	0.057	530
H-11 Tool Steel					0.0015R x	0.057	600

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- 4) Voltage Sensitive Relay: If for any reason the APU should lose criticality before the completion of a year, the voltage drop caused by the descending temperature connected with the loss of power will unlock a firing pin and permit it to fire the primer cap, as above.

All these mechanisms have been under study and show promise of parallel usage, permitting end-of-life actuation in any possible event.

c. Reentry Shutdown Actuation

The reentry shutdown actuator is a stainless-steel band stretched around the periphery of the reflector assembly. The band is composed of a series of silver braze joints of an alloy which will melt when heated by reentry aerodynamic heating. Several alloys have been selected which have melting points in the range of the calculated reentry band temperatures. Tensile test specimens have been fabricated and tested to determine the 0.2% yield and ultimate tensile strength of these alloys at various temperatures from room temperature to 1200°F. "Incusil 15" (Western Gold and Platinum Alloy) and "630 VTG" (Handy and Harman Alloy) appear to have adequate strength at the calculated environmental temperatures and appropriate melting points. Both of these alloys have been tested in high vacuum and at temperatures up to 900°F to determine the rates of vaporization. In both cases the rate is less than 18×10^{-8} gm/in²/hr which will not result in any appreciable loss of strength in the joint, nor any material change in the melting point. These materials are now undergoing long-term creep-rate and creep-rupture testing to determine safe design stress levels. Preliminary data has been plotted and new test stress levels determined by extrapolation of this data. It is anticipated that this creep test program will require approximately ten more months to complete on the two alloys under investigation. Test specimens have been fabricated for radiation damage testing and are scheduled for irradiation in May 1962.

D. SENSORS AND ACTUATORS DEVELOPMENT

1. Startup Programmer

Basic requirements for the SNAP 8 startup programmer have been determined. It will require a time base of approximately ten minutes with discrete outputs on six different channels to accomplish the functions required during

startup. The programmer will be a solid-state device using much the same basic designs as used in the SNAP 2 startup programmer. Experience gained from the SNAP 10A timers is also directly applicable to this design.

2. Temperature Sensor-Switch

The temperature sensor-switch presently being considered for SNAP 8 is the thermal differential expansion switch currently being qualified for SNAP 10A. The qualification of these switches has proceeded in the following areas:

- a) Six switches have received brief functional and calibration tests.
- b) One switch has been placed on life test at temperature and has completed approximately 4,000 switching operations.
- c) Two switches are being placed on life tests at the time of this report.
- d) Three switches have completed shock and vibration testing.

One switch of an early design which was reported in the last quarter has now accumulated over 150,000 switching operations. Time response measurements have been made. It has been determined that the time response of the switch is dependent upon the method of installation of the switch into the NaK pipe; therefore, an estimate of time response for this switch under SNAP 8 conditions will not be forthcoming until the installation method is known.

A number of other temperature sensor-switch combinations have been investigated during the quarter. These combinations are intended as a backup for the thermal differential expansion switch. Stability tests are being conducted on two combinations (a) a platinum sensor - a-c transistorized switch, and (b) a platinum sensor - d-c transistorized switch. Stability tests were completed on a platinum sensor - meg-amp switch combination with satisfactory results. (The 500-hr stability test was within the required $\pm 2^{\circ}\text{F.}$)

3. Controller

The logic for the SNAP 8 controller was worked out and was breadboarded. The control scheme requires that the three drums be stepped alternately with a provision for keeping them "even" despite reversals. This necessitates a somewhat more complicated logic design than the SNAP 10A controller. In addition, provision has also been made for changing rates during startup.

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Various electromechanical schemes are also being considered, as well as the transistorized version described above.

4. Actuators

Two basic types of actuators are required for SNAP 8. One type of actuator is required to drive the three startup drums and a second type to drive the three control drums for the entire one-year operation. The startup drum actuator will be a continuous-drive d-c motor which will drive the startup drum through 180° in ten minutes. This actuator will be rewound and reassembled for high-temperature operation using the techniques developed under the SNAP 10A actuator program. The control drum actuator will be an advanced version of the SNAP 10A control drum actuator. Actuators of this type have gone through the following tests:

- a) Temperature-vacuum tests (750 hr at 800°F and a vacuum of 1×10^{-5} Torr).
- b) Shock and vibration tests (no detectable damage to any portion of the actuator after testing in accordance with SNAP 10A qualification test procedure).

Figure 15 shows this actuator in its assembled form and also shows all the components of this assembly.

A second type of actuator being developed for the control drum application is the linear actuator. A working model of this actuator is shown in Figure 16. This actuator has been operated at room temperature and will undergo high-temperature and vacuum testing shortly. Several other conceptual designs using this principle have also been made.

5. Cable Harness

SNAP 8 will utilize the cable harness technology being developed for SNAP 10A. Recent tests in this area showed that quartz, glass, and Phosroc insulation material is satisfactory. Quantities of twisted-pair shielded cables using each of these insulating materials were ordered for further testing. Miniature connectors of the type to be used have been thermal shock tested to 1200°F with no indication of physical or electrical damage.

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Figure 15. Continuous Drive D-C Motor Control Drum Actuator

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Figure 16. Linear Control Drum Actuator

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6. Safety Circuits

The various safety systems of the SNAP 8 reactor require electrical power sources and firing units. These include squib firing units, destruct firing units, and shutdown-actuator firing units. The firing units being developed and qualified under SNAP 10A are directly applicable to the SNAP 8 requirements.

7. Diagnostic Instrumentation

A preliminary listing of the diagnostic instrumentation requirements has been made. Most of the transducers required will be obtained as off-the-shelf items. Drum position-sensors, however, must be developed because of the high-temperature environment which they encounter. The sensor being developed and qualified by a vendor for SNAP 10A will meet the SNAP 8 requirements. However, an effort is also being made at Atomics International to develop a high-temperature position-sensor. High-temperature insulation concepts and materials developed under the insulation development program are being applied to the AI position-sensor development program. A synchro, rewound as a simple variable reluctance resolver, has performed well functionally, and two improved versions are being fabricated for high-temperature testing. All demodulators for both of the position-sensors mentioned above will consist of transistorized electronics.

8. High-Temperature Insulation

Tests of encapsulating cements have continued. Approximately 100 spools have been tested. As a result of these tests, a new encapsulating cement, SAP II, has been developed and a suitable magnet wire material which is most compatible with this cement has been determined. This system exhibits a combination of physical strength and electrical insulation far superior to that of Allen PBX cement at both 800 and 1000°F.

A spool test was completed which gave a preliminary evaluation of the Westinghouse encapsulating system. The results of this test compared well to our SAP II system.

Table XII shows the relative effectiveness of the insulating systems presently under consideration.

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TABLE XII

INSULATION RESISTANCE COIL-TO-COIL

Encapsulant	at 800°F (ohms)	at 1000°F (ohms)
SAP II	18.2 M	1.1 M
PBX	6.5 M	395 K
Westinghouse	14.3 M	1.2 M

9. Radiation Test Program

Specifications were released for the radiation testing program for both high and low flux components. Requests for proposals for this program were sent to seven irradiation facilities, with five facilities submitting proposals. These proposals were evaluated in detail and a site was recommended. This selection, however, has not yet received final approval.

E. SHIELD DEVELOPMENT UNIT

1. Material Compatibility

a. Stainless Steel With Lithium Hydride

Preliminary results from the 1000-hr corrosion tests of various stainless steels with lithium hydride are listed in Table XIII. Depth of reaction at 900°F was quite small with no measurable amount detected in the case of CRES 347 or 19-9-DL. A considerable increase of depth of reaction zone was noted with CRES 304 at 1200°F and at 1400°F. All specimens showed carburization at 1200°F and decarburization at 1400°F.

b. Special Materials With Lithium Hydride

Specimens of zirconium hydride showed complete structural failure when in contact with lithium hydride at 900, 1200, and 1400°F for 1000 hr. The specimens of beryllium and beryllium oxide have not yet been subjected to metallographic analysis. Results will be reported next quarter.

c. Tensile Testing of CRES 316 Exposed to Lithium Hydride

The 1000-hr tensile test specimens are being prepared for testing. Data on the 900, 1200, and 1400°F effects of lithium hydride on 316 tensile strength will be reported next quarter.

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TABLE XIII

LITHIUM HYDRIDE CORROSION TESTS
ON VARIOUS STAINLESS STEELS

Material	Test Temperature (°F)	Depth of Reaction Zone (in.)
304 SS	900	0.001
304 SS	1200	0.006
304 SS	1400	0.015
316 SS	900	0.001
316 SS	1200	0.002
316 SS	1400	0.002
347 SS	900	NIL
347 SS	1200	0.004
347 SS	1400	0.003
19-9-DL	900	NIL
19-9-DL	1200	0.002
19-9-DL	1400	0.001

d. Min-K-2000 Insulation Material

A series of tests have been performed on Min-K-2000 insulation to evaluate its use as a thermal insulating barrier between the reactor and radiation shield. Compressive strength was measured at 10% deformation for both the as-received and outgassed material. The as-received material strength was 190.5 psi while the evacuated specimen was 239.5 psi.

Tests were run on the outgassing potential up to 1200°F. An encapsulated specimen generated a gas pressure of 260 psi at 1200°F due mostly to the breakdown of the resin binder in the material. After outgassing at 1200°F, no further gas evolution was noted. An analysis was performed on the gases evolved. Results are reported in Table XIV.

2. Lithium Hydride Poisoning Studies

It has been established that the heat generation from the $\text{Li}^6 (n, \alpha) \text{H}^3$ reaction is excessive in the volume of the shield nearest the reactor. Removal of the Li^6 isotope from the natural LiH by the use of Li^7H (99.99% Li^7) would

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TABLE XIV

GASES EVOLVED FROM MIN-K-2000 INSULATION

Gas	Mole %	Gas	Mole %
H ₂	6.1	O ₂	4.4
Methane	21.6	A	0.5
Co ₂	23.6	C ₂ H ₂ (Ethylene)	3.8
N ₂	38.5	C ₂ H ₆ (Ethane)	1.4
Note: Water content is less than 1%			

lower the heat generation, however, it becomes necessary to add a nuclear poison to the Li⁷H in order to maintain equivalent thermal neutron capture. Gadolinium at 0.3 wt % was selected for the additive because of its high capture cross section and its predominately (n,γ) reaction.

Samples of lithium hydride were reacted with Gd metal and Gd₂O₃ for 1 hr at 1325 ±25°F. Examination of the solidified mass revealed crystals of various colors indicating nonhomogeneity. Gd metal was then reacted with lithium hydride at 1325 ±25°F for 63 hr. The crystals were much more uniform in appearance after the long heating, although some evidence of nonhomogeneity was still present. Crystalline samples are currently being analyzed to determine the concentration of Gd and its distribution throughout the solidified hydride mass. A larger melt is being prepared to determine the effect of size on homogeneity.

3. Cast and Powder Shield Development

a. Fabrication Facility

The cast and powder shield fabrication facility was planned and drawings and specifications prepared. The contract for the installation of the laboratory was placed with a construction company. All necessary equipment has been ordered. Scheduled completion of the facility is May 1962.

b. Two-Region Cast Shield

Because of the high cost of Li⁷H (Li⁷ 99.99% pure), a design and development effort was begun to determine the feasibility of fabrication a two-region shield. The region nearest the reactor would contain 260 lb of high purity Li⁷H with 0.3% gadolinium poison. The balance of the shield would contain 470 lb of

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natural lithium hydride. A basic approach has been developed which would utilize a gravity feed hopper technique for the casting, and would separate the two materials by use of two concentric hoppers. The liquid LiH would flow through concentric pipes into the shield container. The Li⁷H region would be separated from the balance of the shield by a stainless-steel diaphragm. Thus the shield container is divided into two regions, each fed by its own hopper during the casting operation, with both regions being cast simultaneously.

4. Lithium Hydride Radiation Damage Studies

Examination of the various inpile radiation testing facilities has resulted in the selection of Hanford K West reactor at Richland, Washington, for the performance of the inpile phase of the LiH radiation damage test.

Design of the test package is proceeding. A six-specimen package will be used containing five specimens of lithium-7 hydride and one specimen of natural lithium hydride. Specimens of Min-K-2000 insulation will be included. All LiH specimens will be cast and separately encapsulated. Radiation time will be selected to provide a thermal neutron dose of 5.33×10^{19} nvt. Inpile measurement of gas evolution, gas pressure and temperature will be performed.

Capsule fabrication will begin in the fourth quarter of FY 1962 with a mockup package schedule for preliminary nuclear tests in June. Inpile testing is scheduled for the first quarter of FY 1963.

5. Shield Container Emissive Coatings

A low emissivity coating will be required on the outside surfaces of the shield container to facilitate removal of the internally generated heat. Analysis indicates that 0.85 to 0.90 emissivity will be required. Present plans are to use a plasma-sprayed chromium-oxide coating. Tests will be performed to determine the rate of evaporation of this coating in a vacuum at 900°F.

Since the bottom of the shield will face high temperature zones inside the vehicle, a low emissivity coating is required. Several approaches are being investigated. A Dalic process nickel coating with a mirror polish and a second coating of plated rhodium of 0.0002 to 0.0003 in. shows some promise, subject to its durability in a vacuum. The usual gold coating will also be investigated as well as a simple, polished, stainless-steel surface.

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V. SYSTEMS TESTS

A. SNAP 8 EXPERIMENTAL REACTOR (S8ER)

1. Introduction

The first nuclear test in the SNAP 8 development program is the SNAP 8 Experimental Reactor (S8ER). This will be a test of the core only, without power conversion equipment. Test objectives are (a) to demonstrate reactor performance at both 300 and 600 kw_t, and (b) to determine reactor performance characteristics. The purpose of the test is to (a) verify the SNAP 8 core design, and (b) to provide experimental data on the dynamic behavior of the reactor required to design the automatic startup and control system.

The S8ER core is essentially the flight reactor core and weighs 200 lb. Since this is a test of the reactor, components outside the core vessel have been designed to meet the test objectives and do not duplicate the flight hardware.

The first phase of the reactor test will be the initial fuel loading and zero power physics tests. These tests will be conducted in a critical facility (Building 373) at the Santa Susana Field Test Laboratory. Then the reactor will be installed in the S8ER Power Test Facility (Building 010). This facility was previously used for the SNAP 2 Experimental Reactor and has been modified to accommodate the higher power S8ER.

2. Nuclear Analysis

A wide variety of calculations was performed to obtain a more detailed understanding of the nuclear characteristics of the S8ER.

a. Critical Loading

Calculations indicate that 160 fuel rods will be required for criticality of the presently planned S8ER critical assembly with the lucite rods. The last fuel rod at the core edge has been calculated to be worth 0.1% $\Delta k/k$ more than the lucite rod it replaces.

b. Axial Hydrogen Redistribution

Multigroup calculations indicate that the reactivity loss due to axial hydrogen redistribution could be as high as 0.5% $\Delta k/k$.

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c. Possible Reactivity Loss Due to Control Drum Bearing Tolerances

The maximum reactivity loss (and also maximum control loss), due to radial tolerances in control drum bearing locations, has been found to be 0.2¢/mil of tolerance for each control drum.

d. Composition of the Lower Grid Plate

Calculations indicate that a reactivity loss of less than 0.1% $\Delta k/k$ at power will result from constructing the lower grid plate of 316 SS rather than Hastelloy C.

e. Range of Possible Multiplication Factors for the As-Built S8ER

The range of possible multiplication factor (k) was estimated, taking into consideration possible manufacturing tolerances and possible errors in calculation. The maximum possible error was estimated to be about $\pm 3.2\% \Delta k/k$, with a probable error of about $\pm 2\% \Delta k/k$. This corresponds to a probable error in the critical loading of about 20 fuel rods.

f. Validity of Calculated Control Drum Worth

A preliminary analysis was made of the CURE code diffusion theory calculations of SNAP 8 control drum worth on the basis of available experiments.

It was found that if CURE studies of the SNAP Experimental Reactor (SER) are adjusted to account for the different void diffusion coefficients used, the CURE results would seem to overstate control drum worth by about 20%. If the experimental SER drum worths are scaled to the S8ER configuration using the changed core radius, reflector length interface structure, and drum diameter, the CURE results understate control drum worth by about 10%. This discrepancy points up the importance of the experimental S8ER control drum calibrations.

g. Temperature Coefficients and Power Defect

A recalculation of the partial temperature coefficients reported in the last quarterly progress report was completed for a power level of 600 kw. In general, close agreement with the previous work was obtained. The results are tabulated in Table XV. The spectrum coefficient has not been re-evaluated.

The difference in power defect between the radial and axial models is attributed to the uncertainty in how to treat the transverse buckling terms. The power defect calculated from the sum of the individual terms represents the sum

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TABLE XV
RESULTS OF PARTIAL TEMPERATURE
COEFFICIENT CALCULATIONS

	$\% \frac{\Delta k}{k}$	$\frac{1}{k} \frac{\Delta k}{\Delta t} \text{ } ^\circ\text{F}$
Axial fuel expansion (1330° F)	-0.52	-0.052
Upper grid expansion (1300° F)	-0.29	-0.029
Expansion of both grids (upper 1300° F, lower 1100° F)	-0.38	-0.042
Reflector heating (no upper bearing)	-1.03	-0.125
Power defect (cold to 600 kw)		
Radial model	-1.74	
Axial model	-1.89	
Sum of individual terms	-1.93	
$\beta_{\text{eff}} = 0.008$		

due to axial fuel expansion, expansion of both grids, and reflector heating. This value should not necessarily agree with the values calculated directly because of interaction effects.

h. Status

The nuclear analysis of the S8ER has proceeded to the point where experimental verification of calculated values is the most important remaining item. Calculations have been initiated to predict all physics experiments which will be performed on the S8ER. The comparison of theory and experiment will provide guidance for future calculations. The two areas of greatest uncertainty at this time are control drum worth and fuel temperature coefficient of reactivity.

3. Core Engineering

The vendor's drawings for fabrication of the "shutdown shield" were approved and fabrication was started. The shutdown shield will support the core vessel, will be the neutron shield during operation, and will provide gamma shielding at the control drives when the reactor is shut down.

Fabrication was started on integral-type core vessels, in which the lower head section and cylindrical tank are formed from a billet by coldwork. The vendor who will perform the coldworking operation completed the

fabrication of a mandril which will be used in the spinning operation during formation of the bessels. Minor modifications in the core vessel external design were changed as a result of changes in the design of the reflector element supports.

Hastelloy N tubing which will be used to clad the beryllium oxide internal reflector elements has been received. End caps for these elements were procured. Checkout of techniques for welding the end caps to the tubing was initiated.

4. Reflector Drive Engineering

a. Redesign of Reflector Control Drum Support and Bushing System

(1) Reason for Change

The control drum support and bushing system for the S8ER has been modified from the original design to increase the performance reliability of the control drum drives. The control drum upper bushing on the original design was subjected to high thrust loads due to the control drum mass, and to high temperatures resulting from heat conducted from the high temperature section of the core vessel. Developmental test data has shown that these bushings were satisfactory for radial loads, but were marginal for thrust loads, at 1300°F. Therefore the bushing at the top of the reactor, which took the thrust load due to the drum weight, was eliminated. The thrust load is now taken entirely by a bearing in the shutdown shield. The bushing on the bottom of the control drum remains a self-aligning, radial bushing.

(2) Description of Modification

The revised design eliminates the control drum upper bushing while retaining the control drum lower bushing, which is subjected only to lighter radial loads at ~1050°F.

The revised design is shown schematically on Figure 17. The details of the drive system located above the shutdown shield remain unchanged. The beryllium control drum is attached to a long support shaft, the upper end of which is rigidly attached to the main drive shaft. The lower end of the support shaft is guided in a self-aligning radial bushing located in the frame assembly which is fastened to the lower end cap of the reactor core vessel. These frame and bushing assemblies are very similar to the original design. With this modification, the mass of the control drum assembly is supported by the duplex ball

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Figure 17. Typical Control Drum Support and Drive

bearing located near the top of the main drive shaft. The loads imposed upon this bearing, which is located in a relatively cool atmosphere ($\sim 200^{\circ}\text{F}$), are well within its rated capacity. The control drum lower bushing serves only to provide radial positioning of the control drum support shaft.

At the initial installation, the control drum assembly and the drive shaft assembly will be positioned to obtain optimum alignment relative to the reactor vessel and the top of the shutdown shield. During reactor operation, some misalignment between the main drive shaft and the control drum lower bushing will occur, owing to differential thermal expansion of the various components. This misalignment will be accommodated by deflection of the support shaft and the self-aligning action of the bushing. In addition, the reaction forces generated during scram and snubbing will deflect the support shaft radially. The beryllium control drum is fixed to the support shaft only at the lower end. The upper end of the control drum is connected to the shaft in a manner which (a) allows relative axial movement between the shaft and the control drum, and (b) maintains radial positioning. Thus, stresses due to differential thermal expansion between the shaft and the control drum are eliminated.

The control drive shaft sleeves through the shutdown shield have been increased in size to a diameter of 1-3/4 in. This modification provides a minimum radial clearance of 3/32 in. between the control drum drive shaft and the sleeves at all times.

(3) Reflector Assembly Test

Plans were initiated for a test of an S8ER reflector assembly. The test will consist of two phases:

- a) A test of a full length assembly, with a stainless-steel element simulating the actual reflector element. This high-temperature test can be conducted outside of a beryllium oxide containment vessel, permitting much easier inspection and clearance-checking during the test.
- b) A test of a beryllium element at temperature with calculated operating stresses applied.

b. Critical Experiment Reflector and Drives Test

The external beryllium reflector elements were fitted to both the critical experiment core vessel and the mock core vessel. The reflectors with

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Figure 18. Critical Experiment Test Configuration

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their drive assemblies, were installed on a test table in the configuration that will be used during the critical experiment (see Figure 18). The mock core vessel has the same exterior configuration as the critical experiment vessel but contains electrical resistance heaters located inside the tank.

Functional checkout of the complete reflector and drive assembly was completed. During the test, a temperature of approximately 700°F was maintained all around the reflector elements. Room temperature was maintained around the drive assemblies. The only problem uncovered during this test was an occasional failure of the scram clutches to disengage. The scram clutch armatures have been chromeplated and the drive system reassembled for further testing.

c. Beryllium Corrosion Protection

(1) Introduction

A program is underway to develop a suitable coating to protect the beryllium reflectors from destructive oxidation in the S8ER environment. The beryllium reflectors, machined from Brush Beryllium Co. Grade S-200B QMV Block, are expected to reach a maximum temperature of 1300°F at full power in a helium environment which will probably contain less than 100 ppm oxygen. As maximum beryllium temperature is dependent upon emittance, high surface emittance of coated beryllium is desired. Reactor operation equivalent to 3000 hr at full power is the minimum goal. It is expected that the protective coating will permit an additional 2000 to 3000 hr of reactor operation at lower power.

(2) Basic Research on Beryllium Oxidation

In an effort to solve the problem of high-temperature beryllium oxidation, a basic study has been underway to establish the effect of vapor pressure of oxygen, metal purity, and microstructure on the oxidation rate. It has been found that the oxidation rate of beryllium initially follows the parabolic law observed by other investigators. The oxidation rate varies approximately as the $1/4.6$ power of the partial pressure of the oxygen. This is in close agreement with the $1/6$ power suggested by the beryllium ion vacancy defect theory. For every 50 to 70°F increase in beryllium temperature, the oxidation rate approximately doubles. After the initial decreasing rate of oxidation, the rate of oxidation starts to increase rapidly reaching catastrophic rates. Data

indicates that these increasing rates seem to occur after a critical thickness of beryllium oxide film is reached. The rapid oxidation is attributed to the metal fragmentation at the surface which increases the exposed surface.

It was concluded that the effects of oxygen pressure and temperature could also be used as a basis for evaluation of test data on the oxidation of coated beryllium.

(3) Evaluation of Oxidation-Resistant Beryllium Coating Studies

Research and development work on a high-temperature oxidation-resistant coating for beryllium has been conducted by Armour Research Foundation under contract to Atomics International. Three basic types of protective coatings are being developed and adapted to well-known coating techniques used for refractory metals. An evaluation has been made of these and other commercial and proprietary commercial coatings.

(a) Ceramic Coating

Oxide coating applied by flame-spraying (oxyacetylene torch), direct enameling, and enamel over other base coats was attempted. While most flame-sprayed coatings were considered too porous to provide adequate oxidation resistance, the Armour barium-titanate coating was considered as a potential. Several samples with this coating were produced. Although the coating, as formed, was uniform and adherent, it spalled off from the base beryllium after it oxidized for 90 hr in a 1410°F laboratory atmosphere.

Enameling with various glass-type enamels under various firing temperatures was attempted on bare beryllium, on a base coat of flame-sprayed on a base coat of flame-sprayed oxides or metals, and on anodized beryllium. All attempts to enamel bare and anodized beryllium failed due to an inability to produce a uniformly adherent coating. Enamels adhered to the flame-sprayed base coat, but the bond between the base coat and beryllium failed and the entire duplex coating spalled off when the samples were cooled from the firing furnace.

(b) Gas-Plated Diffusion Coating

Diffusion coatings (produced by the reaction between beryllium and chromium, titanium, and silicon alone or in combination) have been produced by the pack cementation process. The various process parameters such as atmosphere, temperature, and pack design were evaluated. Two coatings

which looked promising, one chromized and one titanized, were produced from this study. Both samples lasted ~ 110 hr in the 1410°F laboratory atmosphere. The coatings failed by forming small local blisters which had oxidized beryllium under the blisters. As both coatings appeared to be equal in performance, and as Armour was more familiar with chromizing, additional development was concentrated on chromizing. The latest chromized samples have shown considerable improvement. Because these coatings are very thin (few micron range), it has been difficult to conduct analytical studies on the coating before, during, and after oxidation to predict the oxidation mechanism.

(c) Metal Surface Coating

From previous experience in producing oxidation-resistant coatings for refractory metals, it was considered that a metallic surface coating would provide the most reliable protection. Various nickel-chrome, titanium-copper, titanium-nickel, and titanium-beryllium base brazing alloys have been applied by slurry-coating and furnace-firing techniques. In all cases the coatings failed upon cooling by breaking away from the base beryllium. These failures were attributed to nonwetting between the coating alloy and the beryllium, probably due to the presence of a minute amount of beryllium oxide. Further development resulted in a titanium-silver base coating which adhered to the beryllium successfully. However, it was thought to be doubtful that this coating would have sufficient oxidation resistance. The success with silver as a wetting agent led to a duplex silver-nickel coating. This coating is produced by first silver-plating the beryllium and then nickel-plating an outer coating. The silver is fused to the beryllium either before or after nickel-plating. Samples with this coating were exposed successfully for ~ 400 hr in a 1400°F laboratory atmosphere.

(d) Evaluation of Other Coatings

Various commercial and proprietary coatings were furnished by Atomics International to Armour for evaluation. These were tested by being oxidized in a 1410°F laboratory atmosphere. The plasma-arc-applied alumina, zirconia, beryllia, and calcium zirconate coatings showed very little protective ability. All samples failed by excessive oxidation or by cracking of the coating. An anodized coating, produced by Nuclear Metals, Inc., provided limited protection; however, after 60 hr the samples had many areas of local oxidation.

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Two coatings which showed promise were an Atomics International coating and an anodized coating produced by Brush Beryllium Co. Further testing indicated that the AI coating was marginal. The Brush anodized coating still shows potential and is being further evaluated.

(e) Oxidation Testing (see Table XVI)

Coatings were initially screened by being subjected to oxidation in a 1410°F laboratory atmosphere. Further screening was conducted in a 1330°F laboratory atmosphere. Coatings which appeared to be promising after the screening tests were then exposed to 1% air in argon at 1330°F. The specimens were inspected periodically and their condition noted. The coatings were considered to have failed when an indication of excessive oxidation appeared on the surface, that is, local oxidation defects or on the overall surface of a specimen. The samples were usually weighed before and after exposure.

(f) Discussion of Potential Coatings

Of all the coatings developed, only three were considered capable of providing adequate oxidation protection to the beryllium reflectors: (1) silver and nickel, (2) Brush anodized, and (3) chromized. The original silver and nickel coating was produced with ~1 mil of silver-plating plus ~3 mil of nickel-plating. The silver apparently acts as a stress-absorbing layer as well as a binding agent.

During oxidation testing, some samples developed one or more fairly large bumps on the surface. A microscopic cross section of these bumps showed that a surface layer still remains adhered to the beryllium with a void between it and the intact surface layer. The beryllium under the bump is still protected. When an AI nuclear analysis indicated that no more than a 0.10-mil layer of silver could be tolerated, Armour produced a revised coating using a 0.1-mil layer of flash silver-plating. A photomicrograph of this coating, as-formed, showed the almost complete absence of the silver layer, most of the silver being absorbed by the beryllium surface. Examination of this coating after exposure revealed considerable separation between the outer nickel layer and the beryllium similar to the bumps. From these results, it was concluded that the thin silver would not provide the same performance as the thick silver. Further effort on this coating is being continued on a limited basis.

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TABLE XVI

OXIDATION TEST DATA FOR POTENTIAL COATINGS

Coating	1410 $\pm$ 7°F Lab Air		1330 $\pm$ 5°F Lab Air		1330 $\pm$ 5°F 1% Air	
	Hours	Failure	Hours	Failure	Hours	Failure
A.I (J. Page)	110	Edge defect Failure one end	320	Local blisters General failure	405	General failure
Anodized (Brush)	110	Local blisters BeO at blisters	380	Oxidation at ends	1188	Surface bumpy, Glazed, & cracked To continue test
Titanized	110	Oxidation at local blisters	400	Oxidation at local defects	745 [†]	OK Removed from test
Chromized	110	Oxidation at local blisters	400	Oxidation at local defects	745 [†]	OK Removed from test
Chromized (Improved)	360*	Failed when furnace temp lost control	-	-	883	OK Test to continue
Ag + Nickel	400*	Same as above	-	-	690 [†]	Several bumps Surface intact Test stopped

*Hours to last inspection at 1390 to 1410°F temperature range. Sample failed over weekend when furnace temperature increased to 1380°F.

[†]Samples removed to release equipment for other tests. Data as of 4-3-62.

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The Brush anodized coating has displayed the longest dilute air oxidation resistance to date. Because this is a proprietary process, very little is known. However, it is apparent that the corners and the electrode-gripping areas are the weak points. Also, the coating is too thin for metallographic examination. To further evaluate this coating, more information on the process is required. In addition, emittance of the anodized surface after exposure should be determined.

The chromized coating produced by using an anhydrous chrome chloride actuator appears to be superior to those produced using potassium fluoride. The difference is probably due to better uniformity and purity of the coating with the former process, since it is assumed that a chrome-rich beryllide surface is formed in each case. However, since the coating is so thin (guessed at a few microns), it cannot be readily analyzed by metallographical or x-ray diffraction analysis. Because of its thinness, further evaluation is required to determine the compatibility of the coating with the alumina and stainless-steel surfaces to which the beryllium will be attached. Emittance on the chromized surfaces produced with KF activation were checked after 400 hr exposure in 1% air at 1330°F. The values range from 0.65 at 800°F to 0.72 at 1200°F, which will help to reduce beryllium temperature. No particular problem is anticipated in coating the "through" and "blind" holes on the reflectors.

(g) Possible Suitable Coatings and Oxidation Failure

Two types of coatings were selected from the Armour work as candidates for use on the S8ER reflector, an anodized coating prepared by Brush Beryllium Co., and a chromized coating prepared at Armour. Each of these coatings is extremely thin, probably one micron or less. During the chromizing treatment, a chromized layer several mils thick formed, but this failed to adhere to the beryllium.

Because the protective coatings under study are so thin, the oxidation behavior of the coated metal involves oxidation of beryllium metal, but differs because of the surface modification. For the coated samples, the induction period prior to accelerating oxidation and failure was about five or six times as long as for uncoated beryllium. Variations with temperature and with oxygen pressure were similar for coated and uncoated material. Accelerating oxidation is caused by fragmentation of the metal in the first several mils

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of surface. In the case of uncoated beryllium this occurs all over the surface but with a preference for starting at sharp corners. Failure of chromized or anodized samples occurred at isolated spots as though at flaws in the protective film. At these spots, metal fragmentation penetrated deeply (several mils) into the metal within a few hours after failure began.

(h) Microscopic Observations

The mechanism of accelerating oxidation of uncoated beryllium is not yet well understood, but distinctive microstructural changes in the surface have been observed prior to and during acceleration. Some very limited observations were made on coated samples and are reported in Table XVII.

(4) Further Developmental Program

To provide additional information for making the final choice between chromizing and anodizing, a series of oxidation tests at 1375 and 1450°F and criteria for examination of the samples have been outlined for the next quarter.

5. Instrumentation and Control Engineering

Certain modifications to the instrumentation system were planned and some completed during the reporting period.

The original secondary NaK system heater controller would not control accurately enough for some of the projected physics experiments. A new controlling system, using a deviation amplifier coupled to a three-mode controller, was selected as an alternate. This unit will control the secondary system heater within a fraction of a degree.

More thermocouples are required on the piping system to improve operating safety and to gain more information on the heat transfer system operating characteristics. All of the thermocouples have been installed and work is continuing on the thermocouple extension leads to the control room.

A new control turret was designed and installed in the critical facility building for the critical experiments.

6. Emergency Power Supply

The existing emergency power supply for the S8ER Test Facility (Building 010) is gasoline-engine-driven, rated at 2 kw, 115 v ac, 60 cycle,

TABLE XVII

RESULTS OF BRIEF MICROSCOPIC OBSERVATIONS OF SELECTED COATINGS

Type of Coating	Test Conditions	Observations
Brush anodized	Prior to oxidation test	The surface was blackened, but not uniformly. At high magnification, it could be seen that there were areas a few microns wide of apparently bare metal.
Brush anodized	After 1200 hr at 1330° F in 1% air	The surface was rough and bubbled but had a glassy shine, except for a few places where the bubbles were fractured and gray metal could be seen inside the bubbles. The bubbles suggest that accelerating oxidation may have already begun.
Brush anodized	After 100 hr at 1410° F	The surface was white to greenish and somewhat bumpy. There were also spots of powdery growth corresponding to the points where the sample was mounted during anodization.
Armour chromized	880 hr at 1330° F in 1% air	No failure or imminent failure was indicated. The surface was dark green to black. (Note: This coating is improved chromizing.)
Polished section - Brush anodized	400 hr at 1330° F in air	No surface film could be seen. The edge was jagged but an unoxidized surface was not available for comparison. There were no pores below the surface similar to those observed for uncoated beryllium after oxidation.
Polished section - Armour chromized	380 hr at 1330° F in air	The failure spots were mounds of metal fragments mixed with oxide, each mound penetrating several mils below the metal surface as well as projecting above the surface. The remainder of the surface did not show porosity or fracture.
Polished section - Al coating	320 hr at 1330° F in 1% air	Most of the surface showed severe fragmentation of the metal.

single phase. This power supply is connected to the instrumentation bus through an automatic bus transfer switch which makes connection after a power outage.

The minimum load required to be supplied by the emergency power supply after a shutdown due to a power outage is summarized in Table XVIII.

TABLE XVIII
EMERGENCY POWER SUPPLY LOAD
REQUIREMENTS

Instrument	Demand (watts)
Core outlet temperature recorder	55
Stack radiation monitor	1,725
Constant air monitor	575
Area radiation monitors	100
Nitrogen system radiation monitor	80
Keithly (nuclear instrument)	65
Total	2,600

The total demand wattage shown above is for the running equipment. The stack monitor has a 1-hp motor. It is possible that this unit would require starting from a complete shutdown condition by the emergency power source. The starting load is approximately 4.5 times the running load. (Running load for the motor only is 1,660 w.)

To meet the above requirements, a 7.5-kw emergency power unit is being provided to replace the present 2-kw unit. (See Figure 19.) A low-voltage transfer device will switch the vital loads to the emergency power supply when the normal supply voltage has dropped 30%. This will reduce starting requirements for the stack monitor and permit addition of nuclear instruments, associated recorders, and NaK level indicators (total 1,460 w) to the emergency bus. A load distribution switchboard for the vital circuits will permit sufficient starting capacity for the stack monitor in case of complete shutdown by manually tripping the various loads and restarting after stack monitor is on the line. By connecting all the nuclear instruments to the emergency power supply, only one voltage regulator is required. An additional voltage regulator would be required

BREAKER 1 - SPARE
 BREAKER 2 - SOLA TRANSFORMER
 BREAKER 3 - STACK MONITOR
 BREAKER 4 - CONSTANT AIR MONITORS
 BREAKER 5 - NITROGEN AND AREA MONITORS
 BREAKER 6 - TRC-10 RECORDER
 NR-3 COUNTRATE RECORDER
 NR-2 KEITHLY RECORDER

BREAKER 1 - LINEAR AMPLIFIER (1)
 SIGMA AMPLIFIER (3)
 COMPENSATED ION CHAMBER POWER SUPPLY (2)
 BREAKER 2 - LINEAR COUNTRATE METER
 MICROMICROAMMETER
 "J" TUBE TRANSFORMERS
 PRE-AMPLIFIERS AND SAFETY AMPLIFIERS
 BREAKER 3 - NR-1 LOG N RECORDER
 2 KEITHLY LOG N AMPLIFIERS
 NR-4 FLUX RECORDER
 NC-1 FLUX CONTROLLER
 LOG COUNTRATE METER
 SCALER
 BREAKER 4 - SPARE

Figure 19. Emergency Power Distribution Unit

if the nuclear instruments were split between the emergency and normal power supplies. The voltage regulator is presently installed and of sufficient capacity to provide all the nuclear instrumentation.

7. Heat Transfer and Service Systems

a. Redesign of Containment Vessel Gas Purification System

(1) Reason for Change

The presence of 1 to 2% of oxygen at operating temperatures will cause serious corrosion of the beryllium control drums. Attempts have been made to solve the corrosion problem by development of corrosion-resistant, high-emissivity coatings for beryllium. The progress made to date in this development effort has not been sufficient to ensure that a completely satisfactory solution will be reached in time to be of use for the S8ER. Control of containment vessel atmosphere oxygen concentrations to less than 100 ppm has been adopted as a means for minimizing the corrosion problem. The recirculating gas purification system to be used will provide a gas flowrate of 30 l/min and should satisfactorily control oxygen concentration.

(2) Equipment Description

Figure 20 shows the arrangement of the recirculating gas purification system and the vacuum pump. The system is designed for 200 psig at 400°F. The recirculating gas purification system will be composed of the following major components: NaK absorption columns, two helium compressors, and an oxygen analyzer. The NaK system will include two vessels containing approximate one gallon of NaK each. A fill line and a drain line will be provided for each vessel. On both the inlet and the outlet of the absorption train will be traps of sufficient capacity to contain all the NaK that might be blown in either direction by a sudden change in pressure. The absorption columns and the traps contain knitted stainless steel mesh to disentrain any NaK vapors. The absorption columns are expected to give satisfactory performance for six months, after which they will be replaced by a spare set. The absorption columns will absorb oxygen and water vapor from the helium by chemical reaction as the helium passes through NaK.

The two helium compressors will be installed in parallel. One compressor will be operated at a time to achieve a flow of 30 l/min. Ten to

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Figure 20. Recirculating Gas Purification System

twenty cc/min of this flow will be periodically routed to the oxygen analyzer from the containment vessel inlet or from the compressor discharge. After sampling, this gas will be exhausted to the stack. Periodic checks with the oxygen analyzer will ensure that vessel atmosphere purity is adequate for the prolonged life of the beryllium control drums. These periodic checks will also serve to detect possible small leaks in the shutdown-shield cooling-air line before such leaks increase containment vessel pressure to the alarm trip point.

The recirculating gas purification system will include a helium makeup system, two sets of double solenoid valves, a pressure indicator, block valves, and a radiation monitor. The helium makeup will replenish any helium

losses from the system and maintain a pressure of about 2 psig in the containment vessel. The function of the solenoid valves will be to isolate the containment vessel gas in case radioactive gas is released inside the containment vessel. The signal to close the normally closed solenoids will be received from the containment vessel high-pressure alarm and the scram bus. If the primary-system rupture disc breaks, NaK from the NaK rupture line in the vessel will not contaminate the gas purification system, because the scram bus will close the solenoids before the rupture disc bursts. The high pressure scram for the primary NaK system will be set at 40 psig and the rupture disc will burst at 50 psig. However, even if the solenoid valves do not close, system containment will be maintained, because this entire system is designed for 200 psig at 400°F, which is well above the pressure which would result from any accident. Each set of double solenoid valves can be isolated from the containment vessel for maintenance by a manual block valve. A pressure indicator will be installed at the outlet of the two compressors to check the dynamics of the system. A radiation monitor will be installed on the outlet line of the containment vessel to aid in detecting NaK leaks within the containment vessel.

The vacuum pump will permit evacuation of most of the free air in the vessel. This will aid in desorbing gases from the containment vessel's interior surface. The vacuum pump will be attached to an existing 2-in. pipe that extends from the trench located at the vessel's flange to the upper compartment above the shutdown shield. A vacuum sensor will be located in this compartment as well as in the inlet to the vacuum pump.

After a satisfactory vacuum has been obtained, the vessel will be pressurized with helium to 2 psig, and the gas will then be recirculated continuously through the gas purification system.

b. Redesign of Containment Vessel Gas Transfer System

(1) Reason for Change

A gas holdup system will be added to the existing containment vessel design. This modification will permit the removal of gas from the containment vessel in the event of a fuel-rupture accident. Transfer of gas from the containment vessel will decrease gas pressure in this vessel and hence reduce any leakage at the vessel-head joint. This will minimize the release of radioactive gases to the reactor room and surrounding atmosphere.

(2) Equipment Description

Figure 20 shows the arrangement of the gas holdup system. This system will be capable of reducing the containment vessel pressure to atmospheric pressure and compressing all the gas removed into the underground tank. The gas holdup system will be composed of a helium compressor, a shielded, gas-holdup tank, and necessary piping and valves. Tank design is in accordance with the ASME Boiler and Pressure Vessel Code for unfired vessels containing lethal gases. Pressure indicators will be installed to determine the containment vessel and gas-holdup tank pressure. Valves will be of the bellows seal type. The compressor, its block valves, and its pressure indicators will be mounted outside the reactor room. The piping from the compressor outlet valve to the gas holdup tank will be installed underground.

This system will be used only to relieve a large overpressure in the containment vessel. It is anticipated that the transfer would be made ~24 hr after an accident occurred. During normal operation, the containment vessel isolation valve inside of Building 010, the compressor inlet and bypass valves, and the holdup tank isolation valve will be closed. The holdup tank sample valve will be closed, capped, tagged, and sealed to prevent accidental opening.

c. Second Electromagnetic Pump Test

A second electromagnetic pump which will be installed in the primary heat transfer equipment vault of the S8ER facility (Building 010) was tested in January 1962. Design specifications are as follows:

- 1) Fluid — NaK 78
- 2) Flowrate — 140 gpm
- 3) Pressure — 40 psia
- 4) Temperature — 1100°F
- 5) Power — 480 v, 3 phase, 4 wire, 60 cycle

The pump developed 196 gpm flow at a head of 40 psi with a NaK temperature of 1300°F. Pump terminal currents were 200, 177, and 175 amp at the three terminals of the pump with a pump voltage of 400 v. Nine KVAR capacitors were required for power factor correction; line currents to the capacitor bank were 103, 83, and 108 amp.

d. Primary System Drain Line Modification

(1) Reason for Modification

An additional primary heat transfer system drain line is required to drain the NaK coolant from the equipment and piping located in the primary equipment vault, in the event a NaK leak develops while the reactor is operating at some significant power level. NaK should remain in the core following reactor shutdown to aid in the removal of "after-glow" heat generated in the core. The "after-glow" heat has been calculated to be 5% of operating power at 1 sec after reactor shutdown. To drain the existing system, it is necessary to discharge NaK through the core. This system modification will make it possible to maintain the NaK coolant in the core, while the bulk of primary NaK coolant in the primary equipment vault is drained from the system.

(2) Design Criteria and Description

The drain line should be located at the lowest elevation possible in the primary heat transfer system. Installation of principal piping in the primary equipment vault has been completed; therefore, location of an additional drain line is fixed by existing hardware and the availability of vault space and pipe sleeves to the secondary equipment vault.

The proposed drain line will be fabricated of 1-in., schedule 10, 316 stainless-steel pipe. The drain line will connect an existing drain valve, located at the electromagnetic pump inlet, to the existing drain line at its exit to the secondary equipment vault. The additional drain line will not cause undue stresses on the existing primary heat transfer system.

8. Experimental Planning and Operations

a. Experimental Program Summary

Prior to startup of the S8ER in Building 010, the S8ER Critical Experiment will be conducted in the SNAP Critical Facility. The objectives of this experiment will be to provide data for checking calculational techniques, to make desired physics measurements, and to calibrate the system prior to installation in the power test facility. The following low-power measurements will be made during the critical experiment.

- 1) Critical loading with poisoned BeO internal reflectors,
- 2) Critical loading with unpoisoned BeO internal reflectors,
- 3) Control element calibration,
- 4) Isothermal temperature coefficient to 700°F,
- 5) Reactivity worth of external beryllium reflector shims,
- 6) Reactivity worth of S8ER fuel rods,
- 7) Flux distribution measurements,
- 8) Pile "noise" measurements, and
- 9) Reactivity effect of various materials in the core.

The following outline is a brief description of the S8ER experimental program to be conducted in Building 010.

(1) Phase I -- Dry Critical Tests

(a) General Preparation

Upon completion of the S8ER Critical Experiment, the S8ER fuel and reflector drives will be removed from that assembly, and transferred to the power test facility. The reflector drives will then be mated to the S8ER reactor assembly and given a final functional checkout. The assembly will then be lowered into the containment vessel. A determination of component scram times will be made with the fully installed assembly. When all pertinent preoperational testing has been completed successfully, the system will be considered ready to accept fuel.

(b) Dry Critical

Initially the core will be filled with lucite rods. Loading will be accomplished by replacing the lucite rods with fuel rods. It is anticipated that no adjustments of the reflector shims will be required, because the correct reflector shim thickness will have been determined during the critical experiment.

(c) Control Drum Calibration

Immediately upon completion of the fuel loading, both the integral and differential control drum worths will be measured for each drum. The conventional "bump period" method will be used to make these measurements. The

curves developed from these tests will be utilized in determining the excess reactivity.

(2) Phase II — Heat Transfer System Checkout

Because it is not possible to seal the primary coolant system before the fuel loading, the final heat transfer system checkout must be performed after the dry critical experiment. These operations will all be performed with a subcritical reactor.

- a) The coolant inlet and outlet lines, the shutdown shield center plug, and the shield cooling lines will all be welded in place, and each weld will be inspected by x-ray.
- b) The system will then be purged with helium and leak tested, using a helium leak detector. A helium atmosphere will be retained in the system upon completion of the leak check.
- c) NaK will be loaded into the system.
- d) A system cleanup will be conducted by circulating hot NaK. Electrical heat will be utilized. This NaK will be drained and discarded.
- e) The system will be recharged with fresh NaK.
- f) All heat transfer system components (i. e., EM pumps, air blast heat exchanger, intermediate heat exchanger, flowmeters, and plugging meter) will be given functional tests to ensure proper operation.
- g) The control drums and drives will be checked functionally at temperature.

(3) Phase III — Wet Critical Tests

Several important tests and measurements will be performed during this phase of the program. All measurements made here will be used in establishing an S8ER "base point" upon which all future operations will be evaluated. Electrical heat will be used in all tests requiring high temperature.

(a) Wet Critical

A low power, ambient temperature, wet critical measurement will be made. This will be used to establish the initial system reactivity. At various times throughout the program, similar measurements will be made for determining the effect of operation on reactivity.

(b) Control Drum Calibration

A determination of both the integral and differential reactivity worth curves of all six control drums will be made at the temperature established as the "base point" for the wet critical. The conventional "bump period" method will be used.

Step 1 will be repeated at 1200°F core average. If any unexplained changes in the calibration curves are noted, a recalibration will be conducted at intermediate temperatures.

(c) Isothermal Temperature Coefficient

The reactor temperature will be raised in 100°F steps, using electrical heat. The system reactivity will be measured at each step after temperature equilibrium has been attained.

(d) Hydrogen Loss Rate Test

The reactor will be monitored over a two-week period of low-power high-temperature operation, to determine the initial hydrogen loss rate. System heat will be provided by the electrical heater.

(4) Phase IV – Power Operation

This portion of the experimental program is most important in achieving S8ER objectives, and thus it will occupy a major portion of the operating time.

(a) Approach to Power(i) Power Calibration

At some sensible power level such that neutron signal in the ionization chambers is well above the noise signal, the core inlet and outlet coolant temperatures and coolant flowrate will be measured to compute the thermal power. Channels VI, VII, and VIII will then be calibrated in terms of thermal power by calorimetric measurements.

(ii) Approach to Power

At a constant core average temperature ($\sim 700^{\circ}\text{F}$), stepped increases in the power level (for example, 10, 50, 100, 200, and 300 kwt) will be made. Measurements of power coefficient, reactor stability (pile "noise" spectrum measurements), and xenon poisoning will be made at each power level.

(iii) Approach to Operating Temperature

After the reactor has attained the 300 kw_t power level and it is assured that systems are functioning satisfactorily, the coolant outlet temperature will be increased in 200°F steps up to 1000°F.

(b) Power Operation - 300 kwt, 100 hr

The reactor power level will be maintained at 300 kw, with a coolant outlet of 1000°F, continuously for ~100 hr. This period will be used to demonstrate the satisfactory operation of the reactor system at power and near design temperature.

(c) Physics Tests

A portion of the data required for these tests will be obtained during the initial approach to power.

(i) Power Coefficient

Upon completion of the power run, the reactor power level will be maintained at 300 kw, and the coolant temperature will be reduced again to $\sim 700^{\circ}\text{F}$. From this starting condition, the reactor power will be reduced in 50 kw steps and the system reactivity determined after each change. Three conditions will be investigated:

- Constant coolant inlet and outlet temperature
- Constant coolant inlet temperature and flow rate
- Constant core average temperature and coolant flow rate

After completion of one test it will be necessary to increase the power level to 300 kw and reestablish xenon equilibrium before the next test is initiated.

(ii) Pile "Noise" Measurement

The pile "noise" analysis will be utilized during the S8ER Critical Experiment to obtain the ratio of the effective fraction of delayed

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neutrons to the prompt neutron lifetime and the zero-power transfer function. This technique of pile "noise" analysis will be repeated during the initial power operation to obtain the power transfer function.

(iii) Effects of Coolant Flow Rate on Reactivity

The reactivity effects of coolant flowrate will be investigated by monitoring the system reactivity at various flowrates while the reactor is maintained at a constant power level and coolant inlet temperature.

(iv) Xenon Transient to 300 kwt

The reactor power level will be increased from zero power to 300 kwt with a constant core average temperature ($\sim 700^{\circ}\text{F}$). The system reactivity will then be closely monitored to determine the xenon poisoning effect.

(d) Sustained Power Operation - 300 kwt, 1500 hr

Upon completion of the xenon transient test, the coolant outlet temperature will be increased to 1300°F . The reactor will be maintained at these conditions for ~ 1500 hr.

(e) Additional Physics Tests

(i) Hydrogen Redistribution Test

The coolant inlet temperature will be reduced as much as possible ($\sim 600^{\circ}\text{F}$) while the reactor is still operating at 300 kwt. When the inlet coolant temperature has reached its minimum, the power level will be reduced to ~ 200 watts and maintained at this power level until xenon equilibrium is established. The core temperature will then be raised to operating temperature ($\sim 1000^{\circ}\text{F}$) with electrical heat. Then the reactivity will be monitored to observe the effect of hydrogen redistribution.

(ii) Control Drum Calibration

The reflector control drums will be recalibrated to determine what effect, if any, power operation has on the reactivity worth of beryllium.

(iii) Pile Oscillator

A reactivity oscillation experiment will be conducted at various operating power levels and temperatures. This data will be used to supplement and amplify the data obtained from the reactor "noise" analysis conducted earlier in the program.

(f) Sustained Power Operation - 600 kwt, 2000 hr

(i) Power Calibration

The nuclear instrumentation will be recalibrated in preparation for 600 kwt operation. Calorimetric methods will be used.

(ii) Xenon Transient

Upon completion of the power calibration, the power level will be increased in steps (25, 50, 100, 200, . . . , 600 kwt) to the operating level, while the core average temperature is maintained at $\sim 1000^{\circ}\text{F}$. After reaching the 600-kwt power level, the system reactivity will be monitored several days to determine the amount of xenon poisoning.

(iii) Approach to Operating Temperature

When the xenon poisoning has attained its equilibrium value, the core outlet temperature will be increased to 1300°F , with the reactor at a power level of 600 kwt. This operating condition will be maintained for ~ 2000 hr.

(5) Phase V - Post Power Tests

(a) Hydrogen Redistribution Test

Upon completion of the 600 kwt operation, the hydrogen redistribution experiment will be repeated.

(b) Control Drum Calibration

The reflector control elements will be recalibrated immediately following power operation at various temperatures.

(c) Hydrogen Loss Rate Test

This test will be a repeat of one performed earlier in the experimental program.

(d) Kinetic Response Test

The kinetic response of the S8ER to small ramp reactivity inputs will be examined. The information gained in these tests will be valuable in evaluating prompt reactivity coefficients.

b. Pile Oscillator Experiment Design

Nuclear design for the S8ER oscillator has been established as a result of analyses performed this quarter. A design based on a reflecting principle is

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necessary because the oscillator must be located external to the reactor (5 in. from the reflector). The S8ER oscillator employs beryllium as the reflecting material and samarium metal as the poison "cover." The samarium causes the area of reflecting material unobstructedly exposed to the reactor to diminish and increase sinusoidally with time.

The mechanism is comprised of five identical segments which, when joined together, provide maximum reactivity strength. The detachment of individual segments allows modification of the oscillator's overall reactivity amplitude. Each segment is essentially a beryllium cylindrical slug with a 1/4-in.-thick strip of samarium metal imbedded in the lateral surface. The height of the samarium varies sinusoidally, and upon wrapping the strip about the beryllium slug, one complete cycle is described (Figure 21).

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Figure 21. Oscillator Segment

As the slug and samarium revolve together about their common axis, the area of beryllium which is "uncovered" and exposed toward the reactor varies sinusoidally with time. The reactivity then can also be shown to vary sinusoidally with time, if the assumption is made that a differential area of beryllium has constant importance (reactivity) when exposed at any height along the segment.

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The peak-to-minimum difference in reactivity for all five segments is estimated to be $1.6 \pm 1.0\%$. This value was obtained through extrapolation of previous experiment reflector worth data, and theoretical S_N transport calculations. The drive motors allow the frequency of the reactivity to be varied continuously from 0.01 to 100 cpm.

c. Operations Special Tasks

The "SNAP 8 Experimental Reactor (S8ER) Final Safeguards Summary Report," NAA-SR-6958, was issued. An addendum, NAA-SR-6958 Add. 1, was also written and issued. The addendum included all information which might require AEC approval before reactor startup, and included reactor operating philosophies, administrative controls, the experimental program, postconstruction testing, and recent design changes. Certain operating procedures were included as examples of the procedures that will be used at the S8ER.

Preparation of the Operations Reference Manual continued. A description of the S8ER was prepared for use in training S8ER operating personnel.

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B. SNAP 8 GROUND PROTOTYPE TEST (S8DS)

1. Introduction

The SNAP 8 Ground Prototype Test is the second phase of the SNAP 8 development program. The SNAP 8 Ground Prototype Test (S8DS) consists of the SNAP 8 Nuclear System (NS) being developed by Atomics International under contract to the AEC and the SNAP 8 Power Conversion System (PCS) being developed by the Aerojet General Corporation under contract to NASA. The NS and the PCS will be coupled together to form the SNAP 8 Electrical Generating System (EGS) a power plant intended to produce 30 kw of electrical power on board satellites or other space vehicles. The SNAP 8 Ground Prototype Test, however, is not primarily concerned with the final flight configuration of the NS and PCS, but rather it is the first test in which the NS and PCS are coupled together to operate as an EGS. While flight type components and configurations will be used where possible the components will actually be arranged in a breadboard configuration. The test will be performed in the Ground Prototype Test Facility (Building 059) which is being constructed by the AEC at the Santa Susana Field Test Laboratory.

The objectives of the SNAP 8 Ground Prototype Test are as follows:

- a. To provide early compatibility data for the nuclear and power conversion systems;
- b. To develop and demonstrate startup and control methods;
- c. To demonstrate a 90-day endurance capability prior to the first feasibility flight in 1965;
- d. To demonstrate a 10,000-hr operational capability; and
- e. To demonstrate the performance of the flight reactor and the nuclear environmental capability of the lithium hydride neutron shield.

2. Experimental Plans

Prior to connecting the NS to the PCS for the EGS tests, the NS and PCS will be subjected to individual shakedown tests to obtain information relative to their operating and control characteristics and to ensure that they will function in a manner satisfactory for operation in the EGS.

The first shakedown test will have the NS connected to a heat transfer system. This heat transfer system will consist of a primary and a secondary

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heat transfer loop utilizing strictly nonflight-type equipment arranged to dissipate all heat generated by the reactor to the atmosphere without any power generation.

The second shakedown test will involve connecting the PCS to a NaK heater which substitutes for the eventual NS. In this way, the PCS can be fully tested to determine its performance characteristics.

After successful completion of the above two shakedown tests, the NS and PCS will be coupled together for testing as an EGS.

Following is a sequential description of the test program:

- a) All NS test support equipment and part of the PCS test support equipment will be installed in the test facility.
- b) The NS dry critical experiment will be performed in a separate facility.
- c) The core vessel head will be welded on at the critical facility and the vessel will be transported to the test facility.
- d) The lithium hydride neutron shield and the core vessel will be installed in the reactor pit. The NS test loop and fill-and-drain system will be connected, and the reactor control drums and reflector will be installed remotely.
- e) The NS shakedown tests will be performed.
- f) The PCS and other PCS test support equipment will be installed in the facility. The NaK fill-and-drain and the helium cover gas and vacuum systems will be disconnected from the NS and connected to the PCS NaK loop.
- g) The PCS shakedown tests will be performed.
- h) The 375-kw PCS shakedown heater will be removed from the facility. The NaK fill-and-drain system will be reconnected to the NS, and the PCS will be connected to the NS.
- i) The EGS shakedown tests will be performed.
- j) The 90-day EGS endurance test will be performed and continued through 10,000 hr of operation.
- k) The PCS will be removed from the facility.
- l) The NS will be removed remotely and transported to a hot cell for examination.

Figure 22 shows the flow diagram of the NS shakedown test system. The PCS shakedown test system is shown in Figure 23, and Figure 24 shows the flow diagram for the EGS shakedown test system.

3. Design Criteria

In general, the test systems will be designed for:

- a) The maximum degree of flight system performance simulation;
- b) The maximum yield of analytical information;
- c) The minimum amount of on-site work required for installation and checkout;
- d) The minimum time required to achieve test objectives;
- e) Remote maintenance and removal of radioactive equipment;
- f) Personnel access to test equipment located in the vault in a minimum of time after reactor shutdown;
- g) The prevention of primary system contamination during piping changeovers; and
- h) Safety of operating personnel and test equipment.

4. Nuclear System Description

The reactor core, core vessel, reflector, reflector control elements, control drives, automatic controller, and lithium hydride neutron shield will be essentially flight system equipment. Modifications as required, however, will be made to permit installation in the reactor pit, remote maintenance and disassembly, repeated startups and shutdowns, and installation of test instrumentation. The reactor will be designed for operation at a nominal 600-kwt level. Operation, however, will be at a nominal 300-kwt level when the NS is coupled to the PCS.

The reactor outlet temperature will be $1300 \pm 20^{\circ}\text{F}$. This temperature is the principal reactor control signal during automatic operation. The control drums will be actuated by this signal to maintain the outlet temperature within the 40°F temperature band of 1280 to 1320°F.

The reactor inlet temperature will be a nominal 1100°F. The actual value will be determined by boiler conditions and the power dissipated to the

Figure 22. Ground Prototype NS Shakedown Test P & I Diagram

Figure 23. Ground Prototype PCS Shakedown Test P & I Diagram

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Figure 24. Ground Prototype EGS Test P & I Diagram

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parasitic load on the reactor coolant inlet pipe. The reactor inlet pressure will be ~ 35 psia.

The reactor will be designed for a 12,000-hr life at a 300-kwt power level. The core vessel design pressure is 45 psia (in vacuum) with a core inlet plenum temperature of 1200°F and a core exit plenum temperature of 1400°F . The flight system core will be modified to include approximately eight thermocouples in the upper plenum for analyzing the coolant temperature distribution, to provide mounting attachments for installation in the reactor pit, and to provide attachments as required for remote removal and disassembly. The reactor will be capable of withstanding 100 reactor startup and scram cycles. The temperature rate of change in the NaK coolant at the reactor exit plenum is not expected to exceed the following values during these cycles:

300 $^{\circ}\text{F}/\text{min}$ from 100 to 700°F during startup

300 $^{\circ}\text{F}/\text{min}$ from 1100 to 1450°F during startup

50 $^{\circ}\text{F}/\text{min}$ from 1300°F to shutdown temperature during a scram

The maximum temperature of 1450°F may be reached for periods as long as 5 min during startup.

The flight system control elements will be modified to permit the numerous shutdowns and restarts required by the test program, and to permit remote installation and removal. The short-lived startup actuators on the flight reactor will be replaced by long-lived actuators.

The reactor inlet and outlet pipes will be 1-1/2-in. diameter, schedule 5, 316 stainless steel. They will extend from the reactor in the reactor pit up to about 1 ft above the vault floor where convenient connection may be made to the test support loop or the PCS. The core inlet and exit temperature, inlet pressure, and differential pressure sensors will be attached to the reactor inlet and outlet nozzles as close to the reactor as practical. Bolted flange joints enclosed in seal-welded cans will be used for convenient connection of the nuclear shake-down heat transfer system or PCS. This type of joint will also serve to prevent possible ingress of metal chips into the NaK system when piping tie-ins are made. The piping will be designed to minimize the NaK inventory in the EGS.

The neutron shield will be the flight shield, modified to permit attachment to a support structure in the reactor pit. It will not be physically attached to the reactor vessel; but it will be placed as close as possible, with respect to the

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reactor vessel, in its flight system position. The shield will be instrumented with pressure sensors to measure the gas pressure buildup inside the shield and with thermocouples to determine temperature profiles.

An automatic controller for the reactor will be provided. It will be of a breadboard configuration located in the control room where it will be readily accessible for modifications as required to vary startup and control parameters.

Flight system instrumentation will be used to the extent practical. Included will be flight-type coolant temperature sensors to be used in conjunction with the automatic reactor controller. Instrumentation will be required to monitor the performance of all reactor system components.

Nuclear instrumentation will be provided to give complete coverage of reactor flux measurements from startup source level to beyond design power level in three power ranges; startup, intermediate, and power. The two startup channels will use fission chamber detectors connected through suitable amplifiers to logarithmic countrate meters. The countrate meters will give a visual indication of count rate and will also have an audio indication (poppy) of countrate changes. The logarithmic countrate meters will have associated relay contacts used for minimum flux startup interlocks and a period circuit connected to the logic amplifier. Wet criticality instruments consisting of a master scaler, slave scaler, and a digital printout scaler will be connected into startup channels 1 and 2. The two intermediate channels will use compensated ionization chambers connected to Log N and period amplifiers. An extended-range channel using a compensated ion chamber, a linear micro-micro amplifier, and a recorder will be included to provide wide range monitoring of nuclear flux. The power level will be indicated on pen recorders and remote indicators on the console. The period output will be connected to a remote indicator and a logic amplifier. The three power channels will use uncompensated ion chambers feeding power level amplifiers. Malfunction and prescram alarms will be provided. On low-power channels, there will be individual period scrams. On the three high-power channels there will be a choice of two out of three coincidence or individual scram circuits. Bistable scram trips will be connected to the logic amplifier.

The output of the logic amplifier will control the drum scram clutches.

5. Nuclear System Shakedown System Description

a. Heat Transfer System

The heat transfer system for the nuclear shakedown test will utilize a two-loop system. This system is shown in Figure 22. Since the NaK 78 in the primary loop will circulate through the core, it will become radioactive. For this reason, the primary heat transfer system will be installed in the equipment vault which will provide the necessary shielding. The primary heat transfer system will include a surge tank, the tube side of the intermediate heat exchanger, a linear induction electromagnetic pump, an electromagnetic flowmeter, a plugging meter, a 30-kw electrical simulated parasitic load heater, and the reactor. The nominal flowrate in the system for 600 kw operation is ~ 140 gpm with a reactor outlet temperature of 1300°F and reactor inlet temperature of 1100°F .

The secondary heat transfer system will consist of the shell side of the intermediate heat exchanger, an airblast heat exchanger, a surge tank, a 150-kw electric heater, a linear induction electromagnetic pump, and an electromagnetic flowmeter. Since no radioactivity will be induced in the NaK coolant circulated in this system, it will be installed in the high-bay area of the reactor building and outdoors without shielding. The nominal flowrate in the system for 600-kw operation will be about 140 gpm with a hot-leg temperature of 1250°F and a cold-leg temperature of 1050°F . The electric heater will be installed in parallel with the airblast heat exchanger, and it will be used only for preheat or for certain experiments, e. g., high-temperature, zero-power experiments.

All valves in the equipment vault will be nitrogen-actuated, cylinder-operator, bellows-seal valves, since remote operation is required for shielding considerations. All piping, valves, and equipment will be designed to permit maximum drainage of NaK. The primary drain piping and valves will be arranged so that the system can be drained down to a level just above the reactor core when reactor afterheat removal by natural convection is required or the system, including the reactor, is completely drained.

Two helium gas supply systems will be provided. One system will supply helium to the primary surge tank, the primary fill tank, and the drain tank. The second helium supply system will supply helium to the secondary surge tank, the secondary fill tank, and the secondary drain tank. A third inert gas system will be provided. This system will supply nitrogen for the operation of the cylinder operated valves in the test vault.

Two vent and vacuum systems will be provided. One of these will provide gas holdup volume for radiolytic decay, and it will exhaust through the building radioactive vent system. The primary surge tank, fill tank, and drain tank will be on this system. The second vent and vacuum system will vent to the building stack, and it will serve the secondary surge tank, fill tank, and drain tank.

The primary and secondary fill tanks will be provided with diffusion cold traps for removal of impurities from the NaK coolant. The cold traps will be cooled by a mechanical refrigeration system.

b. Instrumentation

Coolant pressures, flows, levels, and temperatures, and cover gas pressure will be the process variables monitored in the primary and secondary heat transfer systems. The instrumentation will be selected to minimize the probability of a reactor shutdown due to instrument failure. All scram parameters will be monitored by two or more sensors. Thermocouples will be duplicated.

The test loop will be instrumented to permit manual and automatic control of the temperature during shakedown tests. NaK pressures and tank levels will be recorded in the control room on multipoint strip chart recorders. The primary and secondary surge and drain tanks will be provided with "J" tubes for level determination. NaK pressures will be measured using electric pressure transmitters provided with NaK-type diaphragm seals.

The 150-kw electric heater and 30-kw electric simulated parasitic load heater will be provided with indicating temperature controllers located in the control room. Remote control of the upper airblast heat exchanger louvers and the fan speed will be from the control room.

The manual and automatic controls, recorders, and indicators for the nuclear shakedown system will be located in the control room with the PCS instrumentation. The instruments concerned only with nuclear shakedown tests will be grouped together, to the extent practicable, so that they will not confuse the EGS tests. The PCS and NS instrumentation will be laid out in graphic panel form. The principal controls for the PCS and NS will be mounted on a console which can be operated and monitored by a single operator. The console and instrument panels will be arranged so that all instruments on them are visible

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to an operator seated at the console. The annunciator panels will be mounted above the annunciator parameter.

Both manual and automatic scram instrumentation will be provided. The automatic scram will shut down the reactor when the following unsafe nuclear system and test loop operating conditions occur:

- 1) High reactor pressure,
- 2) Low reactor pressure,
- 3) High reactor differential pressure,
- 4) High reactor exit temperature,
- 5) High reactor power,
- 6) Short reactor period,
- 7) Low primary NaK flow,
- 8) Low primary NaK surge tank level, or
- 9) Earthquake.

During the EGS tests, some of the PCS parameters will also be connected to the reactor scram circuitry. These might include:

- 1) Mercury dump,
- 2) Loss of radiator cold wall coolant,
- 3) Turbine speed,
- 4) Loss of NaK pump power, and
- 5) Alternator short.

A d-c power supply for the control element actuator will be provided to furnish power for the control elements during the NS shakedown tests and EGS tests in the event the PCS does not supply this d-c power.

6. Facilities Description

The SNAP 8 Ground Prototype Test Facility is presently under construction in the AEC triangle at the Santa Susana Field Test Laboratory. The test building and miscellaneous outdoor equipment pads associated with it will provide facilities for the ground prototype test program. The facility building is divided functionally into four major areas. These are as follows:

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- a) General support and operating area,
- b) High bay area,
- c) Vault area, and
- d) Equipment room area.

Figures 25, 26, 27, and 28 show the building arrangement, and Figure 29 indicates the location of outdoor equipment.

The general support and operating area provides space for the control room, data reduction and instrument repair room, offices, change area, and facility equipment room. Within the control room will be located the instrumentation racks and control console for the test systems. Located here, also, is a small panel for control of the facility environmental system. The office area provides space for data reduction activities, two offices, and a health physics area. The change room and toilet area provide facilities for three shifts of 10 men each. The toilet area also provides for an additional 5 visiting personnel. The facilities equipment room houses the air conditioning and heating equipment and the hot water heater.

The high-bay area provides unobstructed access to the test vault through hatches, plugs, and similar accessories. Space is provided at the west end of the area for nonradioactive NaK test equipment. Controls for the crane in the vault, television system, and television monitoring panel are located in the same area. Small plugs and a trench system are provided around the periphery of the floor above the vault for instrument cables, electrical control wiring, and the like. The north wall of the high-bay area will be used for routing of the test systems process piping between the vault and the equipment in the high bay and outdoor equipment. A 10-ton bridge crane is provided in the high bay for handling equipment and concrete plugs.

The test vault area is designed for installation and operation of the basic components of the test systems. The portions of the test systems installed in the vault include the reactor, all components of the primary NaK loop, the boiler and rotating machinery package, and the prototype radiator condenser. The reactor will be installed in one of the two reactor pits located in the west end of the vault. A shielded pipe chase in the floor for primary NaK connects the two reactor pits. A recess is provided within the vault for the cooling coil and fans for vault environmental control.

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Figure 25. Ground Prototype Test Facility, First Floor Plan

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Figure 26. Ground Prototype Test Facility, Sectional View

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Figure 27. Ground Prototype Test Facility, Elevations

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Figure 28. Ground Prototype Test Facility, Test Vault Sectional View

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Figure 29. Ground Prototype Test Facility, Plot and Utility Plan

The entire vault is enclosed by concrete shielding for protection of personnel against radiation, and lined with steel plate to prevent leakage to the exterior in excess of 1% of the vault volume per day at a differential pressure of 3-in. water gauge. Penetrations sealed to the vault liner are provided through the shield walls for electrical conduits, piping, plugs, and ductwork. In general, welding is used for penetration sealing. A staircase is provided for personnel access to the vault area.

Remote handling facilities will be installed by AI in the vault to service the equipment in the vault.

The equipment room area is divided into the process equipment room and the electrical equipment room. The process equipment room contains only equipment required for control of the vault environment and/or facility equipment which may become contaminated. All piping which may become contaminated is located in a large trench area. The holdup tank compressors and the prefilter in the infiltration air exhaust system are located within a permanent concrete shield wall enclosure. The electrical equipment room contains all facility distribution panels, motor control centers, the emergency generator, and auxiliary equipment. Space is provided for later installation of additional electrical equipment.

Concrete pads located adjacent to the building and in an area in the vicinity of the fence at the north end of the site accommodate process equipment, test equipment, and other facilities. The vault cooling kerosene heat exchanger equipment occupies the concrete pad adjacent to the north wall of the building immediately outside the equipment room. Radiator cooling equipment, airblast heat exchanger, and other test equipment will be installed on the pad on the east side of the building. The electrical substation, liquid nitrogen storage, contaminated gas holdup tank, and the facility cooling tower is installed in the area at the north end of the site.

Facility process systems are provided for vault environment control, radioactive gas discharge, and shield cooling. An air conditioning system supplies air for distribution to the control room, offices and data reduction room. Heating is provided for all building areas except the mechanical and electrical support areas.

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There are auxiliary systems other than those mentioned above to serve the facility. There is a compressed air system to supply instrument systems and for the operation of tools. A cooling tower system is provided to supply cooling water for the air compressor after-cooler, shield cooling heat exchanger, and air conditioning refrigerant condenser. A drain system is provided to collect contaminated drainage from the shower, lavatory, and floor drain in the hot change room, from floor sinks in the process equipment room, and a subsurface drainage system. An intercommunication system is installed to enable two-way conversation between the control room and the high-bay area, process equipment room, or vault. For personnel safety, a mercury leak detection system, an area radiation monitoring system, a stack radiation monitoring system, and an airborne particulate monitoring system is provided. There is a fire fighting system consisting of portable dispensing receptacles containing CO₂, "Met-L-X," and water.

A three-channel television system is provided for remote viewing and manipulation within the vault. The control console for the system is located in the secondary equipment area adjacent to the location of the invault crane controls.

An electrical substation steps the primary distribution system electrical power from 4160 to 480/277 v. It feeds a 3-phase, 4-wire facility distribution system. One bus in the facility distribution system will supply those loads requiring emergency power provisions. On failure of the normal power supply, an emergency power supply is switched in to assume the load on the bus. The emergency power is obtained from a natural-gas, engine-driven alternator.

7. Status

a. Coordination

Coordination efforts have been centered around establishing the following:

- 1) The division of responsibility between AI and AGC with respect to the Ground Prototype System Tests which are to be conducted in the Ground Prototype Test Facility (Building 059) at Santa Susana, and
- 2) Test equipment layouts in the test facility.

A basic agreement with respect to the Ground Prototype System Tests was reached by AI and AGC and submitted to the AEC and NASA for approval.

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However, the details of the operating organization and administration procedures for performance of the tests were not finalized.

A vault layout of the test systems equipment which appears to be acceptable to both AI and AGC is shown in Figure 30. The design of the AI test support equipment is currently based on this layout.

b. Heat Transfer System

The heat transfer system requirements were confirmed, and a preliminary process and instrumentation diagram was prepared. Equipment arrangement and piping layout studies were initiated. The criteria of minimum piping lengths in the primary heat transfer system were adhered to in order to minimize pressure drop and NaK volume. This has led to the consideration of possible use of bellows in the primary piping systems in order to reduce the bending moments resulting from the close coupled piping system.

The design effort has progressed to the point that permits initiation of procurement of some long-lead delivery items. Specifications for the intermediate heat exchanger and the airblast heat exchanger have been prepared and released for procurement action. A specification for the high-temperature NaK bellows seal valves has also been prepared and released for procurement.

c. Instrumentation

A preliminary nuclear and scram bus block diagram has been prepared (Figure 31), and specifications for the nuclear instrumentation have been completed. However, revisions concerning the operating power supply requirements during power line outages are being considered.

Studies were made of the total power requirements of the EGS electrical system to permit continued EGS operations during a power outage. Procurement action for long-delivery-time instruments and material for instrumentation installation was initiated.

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Figure 30. Preliminary Equipment Arrangement, S8DS Primary NS and PCS Shakedown Test

Figure 31. S8DS Preliminary Nuclear Block Diagram and Scram Bus