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PROGRESS REPORT

SNAP 2/10 REACTOR

JULY-SEPTEMBER 1961

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NAA-SR-6794
SNAP REACTOR,
SNAP PROGRAM
M-3679 (25th Ed.)

Classification cancelled (or changed to) **UNCLASSIFIED**
letter 4/17/73

by authority of Bram Feldman

by GG

DTIE, date 6/18/73

PROGRESS REPORT

SNAP 2/10 REACTOR

JULY-SEPTEMBER 1961

(Title Unclassified)

Editor

W. M. HAUSSLER

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THE PREVIOUS PROGRESS REPORT IS:

NAA-SR-6694

April-June 1961

Issued: December 1, 1961

NAA-SR-6794

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I. PROJECT OBJECTIVES

Objectives of the SNAP 2/10 reactor program are to:

- 1) Provide nuclear and mechanical designs for SNAP 2/10 reactor components and associated structure.
- 2) Provide fuel elements for SNAP 2/10 programs.
- 3) Undertake research and development to increase fuel element reliability and improve fuel material properties as well as processing and fabrication techniques.
- 4) Conduct nuclear, heat transfer, mechanical, and safety and hazards analyses as required to insure that reactor systems designs comply with all requirements and specifications for SNAP systems.
- 5) Undertake appropriate component and subcomponent development and testing to insure the availability of adequate hardware for reactor systems components.
- 6) Demonstrate the feasibility of the SNAP reactor concept and obtain design information through operation of SNAP reactor systems.

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II. PROGRAM SUMMARY

The one-piece fuel element (without end reflectors) was selected as the reference element for both the SNAP 2 and SNAP 10 reactors on the basis of core optimization studies and nuclear analysis. Specifications for the fuel element were prepared during the quarter.

Analysis of SNAP 2/10 reactors under various accident conditions involving water immersion or water reflection of the core was carried out for the preliminary hazards report.

Detailed design and operational planning, procurement of long lead-time hardware, and preparation of the hazards report for the SCA-4 (SNAP Critical Assembly No. 4) program were initiated.

Radiation dosages through the reactor shields and radiation scattering from the control drums and other auxiliaries surrounding the reactor vessel were determined. Various modifications were made in the design of these assemblies in accordance with the results of the analysis.

Zero-power tests on the S2-DS-1 reactor at Santa Susana were completed. They demonstrated that the design calculations for the core closely predicted its actual performance.

The S2-DS-1 was operated for a period of 1150 hr at SNAP 10 conditions to accumulate experimental information on xenon buildup effects and long-term reactivity losses.

The reactor structure for the S10-PSM-1 system was delivered to the test site and incorporated into the system assembly.

Mechanical design of the S10-FSM-1 reactor structure and shield was initiated.

Re-entry effects were investigated and design modifications were made to the reactor structure and core design to facilitate ejection of fuel elements from the reactor vessel during re-entry.

Material property determinations and evaluation of various fabrication techniques for the lithium hydride shield material were begun in the component development laboratory.

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Design layouts for the SNAP 2 shield assembly were started, together with thermal analysis of the shield under orbital operating conditions.

In component development, a breadboard circuit for the reactor control system was assembled and operated. Test work was also initiated on various mechanical and thermal reactor shutdown actuators. Welding investigations are under way for the reactor core vessels together with hydraulic testing of coolant flow distribution through the core.

Screening tests on bearing compatibility and spring relaxation are continuing and work was initiated on evaluation of gear materials and gear tooth coatings for operation under SNAP environmental conditions.

Environmental testing is also under way on temperature sensors, micro-switches, and various electronic parts and insulators.

Modification of the high-temperature vacuum chamber has been completed to permit operation up to 1000°F and 10^{-9} mm Hg vacuum.

In fuel development, evaluation of a new fuel alloy material containing 0.5 wt % carbon was initiated. The carbon modification promises to give a stronger, finer-grained alloy which should resist cracking during the hydriding step more successfully than the alloy currently in use.

Fabrication of fuel for the SCA-4 facility was started during the quarter in the existing "brickpile" hydriding furnaces.

In view of the poor hydriding yield from the new, large furnaces, the decision was made to construct eight additional brickpile units to meet the forthcoming production requirements for hydrided fuel rods.

Further development of the cup assembly for making an improved seal at the blended end of fuel elements resulted in specifying this cup-blend as standard for the SNAP 2/10A elements. The average leakrate from assembled fuel elements is now about $0.4 \text{ cm}^3/\text{hr}/\text{element}$ at 1200°F isothermal—or, roughly half that of the elements used in the S2-DR core. Fuel elements assembled with the cup-blend showed good resistance to shock and vibration damage which had previously increased the leakrate from the elements.

Results of a 1000-hr 1300°F compatibility test between beryllium and Hastelloy N, both plain and chromized, show that the two form a diffusion zone up to 20 mils thick.

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Hot cell examination of the six specimen SER elements began this quarter with external measurements and gamma scans. A full-size S2-DR-type fuel element has been assembled into an irradiation test capsule to be run in the WTR (Westinghouse Test Reactor). Insertion is scheduled for October.

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III. REACTOR SYSTEM ANALYSIS

A. FUEL ELEMENT

Optimization of the core design based on the previously conducted nuclear analysis resulted in selection of the one-piece fuel element as the reference design for both SNAP 2 and 10 reactors. Detailed specifications for the fuel element were prepared during the quarter. The fuel element is shown in Figure 1 and pertinent design criteria are listed below:

- 1) Maximum fuel temperature - 1250°F
- 2) Thermal cycling requirements - the elements shall withstand 20 temperature cycles from 70 to 1250°F at a rate of 150°F/min
- 3) Fuel material - 10 ± 0.24 wt % of uranium in a uranium-zirconium hydride alloy with an isotopic enrichment greater than 93% U^{235} .
In addition, each fuel element shall contain 128.4 ± 3.1 gm of U^{235} .
- 4) Hydriding requirements - all fuel material shall have an N_H of $6.5 \times 10^{22} \pm 1 \times 10^{22}$ atoms of hydrogen per cc of fuel. An assembly of 37 fuel elements shall have an average N_H of $6.5 \times 10^{22} \pm 0.03$ atoms/cc.
- 5) Fuel burnup - the peak fuel burnup in the central element is approximately 1% of fissionable atoms.
- 6) Cladding material - the cladding material shall be Hastelloy N alloy with the gap between fuel alloy and cladding filled with helium.
- 7) Hydrogen barrier coating - the entire fuel element shall be internally coated with Solaramic S-14-35A-SM-2 containing 6.6 ± 0.66 milligrams of Sm_2O_3 per inch of fuel rod length. This hydrogen barrier coating shall limit the rate of hydrogen leakage from the fuel element to values of 0.8 cc/hr at 1200°F and 0.04 cc/hr at 900°F.
- 8) Fuel element weight - each fuel element shall weigh 3.32 lb.
- 9) Dimensions (inches):
 - a) Active fuel diameter - 1.212 ± 0.002
 - b) Active fuel length - 12.250 ± 0.002

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Figure 1. SNAP 2/10 Fuel Element

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- c) Cladding thickness - 0.0150 ± 0.0005
- d) Element OD - 1.250 ± 0.001
- e) Element length - 12.450 ± 0.003

B. ACCIDENT ANALYSIS

Analysis of the SNAP reactors under accident conditions was started during the quarter with preparation of the preliminary hazards report. Reactivities associated with various environments and various accident conditions were analyzed for the SNAP 10 reactor using the AIM-6 diffusion code. Results are shown in Table I.

TABLE I
SNAP 10 REACTIVITIES IN VARIOUS ENVIRONMENTS
(Dollars)

Condition	Reactivity with NaK in Core	Reactivity with Water in Core
Complete Be reflector	2.4	9.6
Be reflector with drums out	- 5.6	2.0
Aluminum reflector	-14.5	- 5.4
Be reflector plus poison blocks	- 5.6	2.1
Water-reflected core	1.3	10.4
Complete Be reflector, plus water	7.8	16.0
Be reflector, drums out, plus water	6.8	15.0
Aluminum reflector, plus water	-13.6	- 4.5
Be reflector with poison blocks, plus water	- 3.5	5.5

As seen in Table I, the more serious accidents involve either water immersion or water reflection of the core. Experimental studies on water reflection were conducted in the SNAP critical assemblies using wax as a water simulator. These experiments indicated that water reflection of the reactor would produce an excess reactivity of \$1.30. Calculations for the same situation using the AIM code resulted in a subcritical reactivity of \$1.47, a difference of \$2.77 from the experimental value. As data from the SCA-4 critical assembly (currently in final design) is generated, it is expected that this uncertainty between experimental and calculated reactivities will be resolved.

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C. POWER EXCURSIONS AND REACTIVITY ANALYSIS

The available transient codes for calculating power excursions in SNAP reactors were surveyed for use in the hazards study and none of them adequately described the transient behavior of fuel rods in SNAP 2/10 reactors. A new code, SNAPTRAN, was initiated to permit beryllium transient calculations. The thermal dynamic model in SNAPTRAN will simulate six representative fuel rods and their associated coolant channels, and will couple them to the reactor kinetics equations. Each fuel rod will be divided into radial and axial nodes and into four separate regions of fuel alloy, hydrogen gap, hydrogen barrier, and Hastelloy N tube.

Detailed planning for the SCA-4 critical assemblies was initiated during the quarter. The following three experiments have been planned for the SCA-4 assembly:

- 1) SCA-4A - a flexible critical assembly using beryllium of various heights. Both the rotating control drum and flopout concepts will be investigated with this machine.
- 2) SCA-4B - water-immersion and flooding experiment.
- 3) SCA-4C - an exact mockup of flight system reactors using FS-1 fuel and FS-1 reflector.

Detailed design of the SCA-4A experiment was completed. The finished critical assembly is shown in Figure 2. Procurement of the beryllium for this assembly has started. Detailed planning and design for SCA-4B has been initiated, together with preparation of the hazards report for the entire SCA-4 program.

A nuclear survey of SNAP 2 and 10 reactivity requirements was undertaken to establish the required samarium concentration in the fuel elements and to investigate the operating characteristics of the reactor. Table II shows the reactivity requirements as determined by this effort.

During this work it became evident that the reactivity associated with axial redistribution of hydrogen in the SNAP 10 fuel elements was an important effect contributing to the successful long-term operation of the reactor. The equilibrium redistribution of hydrogen for SNAP 10A was calculated and is shown in

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Figure 2. SNAP Critical Assembly (SCA-4) Machine

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Figure 3. The reactivity associated with this change in hydrogen concentration is 18¢. Detailed study of the transient behavior of this phenomenon is extremely complicated, and preparation of a new digital code has been initiated to investigate these effects.

TABLE II
SNAP 2/10 REACTIVITY REQUIREMENTS

	SNAP 2	SNAP 10
Temperature and power defect	\$2.41	\$1.95
Equilibrium xenon	0.23	0.12
Burnup, fission products, and prepoison	0.24	0.21
Equilibrium samarium	1.15	1.15
Hydrogen leakage	0.82	0.11
Hydrogen redistribution	0.23	0.18
Total	\$5.28	\$3.72
Prepoisoned	1.36	1.36
Control	\$3.92	\$2.36

Analytical efforts were also initiated to investigate the orbital dynamics and equilibrium space temperatures of the reactor. This effort will also consider re-entry phenomena heating effects, starting at the free-molecular regime and extending into the continuous heating regime.

A detailed evaluation of the gamma ray sources for the SNAP 2 and 10 reactors was performed. The respective neutron shields were then analyzed for gamma effectiveness. In SNAP 10, with the one-piece fuel element, gamma intensity was reduced to 80% of the design limit of 10^7 r/yr at the dose plane. In SNAP 2, with one-piece fuel, the gamma intensity just meets the design limit at the dose plane.

Design of the control drums and drum bearing assemblies for SNAP 2 and 10 was brought to a stage where a radiation scattering analysis could be made. It was determined from this study that neutrons were scattered from these assemblies to the dose plane and beyond, since the bearings and partially-rotated

Figure 3. SNAP 10 Average Hydrogen Content vs Axial Position

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drums extended beyond the 14-in. diameter design point for the shield. Radiation in the outermost 6 in. annulus of the dose plane was calculated to be 6-40 times greater than the one year tolerance limit of 10^{12} nvt and 10^7 r. The decision was made to modify the control drums and thereby reduce the amount of material projecting beyond the 14-in. diameter design point. In addition, the drum bearings will be reduced in size, and together with other supports in this region of the reactor, will be made principally of titanium. The choice of titanium is based on an evaluation of radiation scattering from several high-temperature materials as shown in Table III.

TABLE III
RADIATION SCATTERING OF MATERIALS CONSIDERED

Material	Relative Scatter Dose Compared to Carbon (for Equal Volume of Material)
Natural carbon	1
Titanium CRES 446 CRES 316	1.25
Inconel X Haynes 90 Deva 3-F-12	1.5
Al ₂ O ₃ MoS ₂ Titanium carbide Deva 3-N-12	1.9
Chromium carbide	2.75

Preliminary designs for the SNAP 2 and 10 shields were completed and are shown in Figures 4 and 5. In SNAP 2, the shield weight, based on a 14-in. diameter core, is 255 ± 5 lb without shield casing; for SNAP 10, the weight is 174 ± 4 lb without casing. In both cases, a weight saving of 71 lb in the shield was achieved through modification of the control drums and bearings.

A literature search was made to determine the best available data on radiation resistance of various payload instruments. The latest 1960 data indicate

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Figure 4. SNAP 2 Shield

Figure 5. SNAP 10 Shield

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that our present limitation of 10^{12} nvt and 10^7 r over the whole dose plane is too restrictive. The outermost 3-6 in. of the reference dose plane will be occupied by instrumentation for the reactor supplied by AI. Since this region will contain very few semiconductors, most of it will successfully absorb a dose of about 10^{15} nvt. Further evaluation of this problem will proceed when a list of tentative components is available.

Preliminary calculations of nuclear heating in the SNAP 10 shield indicated that a total of 200-300 watts will be generated. In the SNAP 2, shield heating is expected to be almost twice this value, or about 500 watts.

D. S2-DS-1 OPERATION

The S2-DS-1 reactor at Santa Susana was run through a series of zero-power tests to measure the physics parameters of the core and then operated at SNAP 10A conditions for a period of 1150 hr. Three long-term operational runs were accomplished during the zero-power tests. During the first of these, which ran for 370 hr at an average core temperature of 882°F , the core experienced an increase in reactivity of 3.9¢. Expressed on a linear basis, this represented a gain of 0.19¢/day. Several experimental runs were conducted at the end of the test, by varying the reactor power level over several decades, to establish the effective strength of the source and to determine if the increase in reactivity was caused by a buildup of fission product gammas in the fuel, which would have the effect of increasing the source strength through the γ -n reaction in beryllium. These experiments indicated, however, that the critical position for the power level at which long-term testing was being conducted, was independent of the source. Consequently, when the second and third tests, which ran for 181 hr at an average temperature of 600°F and 177 hr at an average temperature of 882°F (duplicating the first test), showed no perceptible change in reactivity, the initially observed change was assumed to be due to thermal relaxation of the structural components of the core.

Steady-state operation for 1150 hr was accomplished during the quarter at SNAP 10 conditions of 882°F average core temperature and power level of 30.5 kw. The equilibrium xenon effect on reactivity was determined to be approximately 14¢ and the long-term loss in reactivity for the 1150-hr operating period was 10.7¢. Expressed on a daily basis, the long-term loss varied from 0.32¢/day at the start of the run to 0.20¢/day at its termination.

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IV. REACTOR SYSTEM DEVELOPMENT

A. REACTOR AND STRUCTURAL SUPPORT

The reactor structure for the S10-PSM-1 system was delivered to the test site and incorporated into the system assembly.

The design of the reactor vessel for the SNAP 2/10A systems was modified to include spring-loaded mountings for each fuel element. This will permit individual insertion of the elements during the core assembly sequence and will also insure ejection of individual elements from the reactor vessel during re-entry. In addition, the closure between the reactor vessel and the vessel head was redesigned to include a stand-off lip seal, which will facilitate burnup of the joint and consequent separation of the head and vessel during re-entry.

A nickel-iron alloy is under investigation for use as the grid plate material inside the pressure vessel. Results for the alloy so far indicate satisfactory tensile strength, ductility, and compatibility with the NaK coolant. Studies of the alloy's physical and thermal properties are continuing.

The support structure for the SNAP 10 reactor vessel was redesigned to minimize radiation scattering. These modifications also resulted in a weight reduction for the overall reactor structure. The design for the support structure for the SNAP 2 reactor vessel was initiated during this quarter. The SNAP 2 concept differs from that of SNAP 10A in that the reactor vessel is attached directly to the shield casing.

B. SHIELD

The shield assembly for the S10A-PSM-1 system was completed at ORNL (Oak Ridge National Laboratory), and, at the end of the quarter, was in transient to AI. The shield casing for this assembly was fabricated in the AI shops and the LiH was installed at ORNL. During the fabrication step, both brazing and welding techniques for assembling the casing were evaluated. The welded concept was discarded because of excessive distortion and the brazing technique adopted for all future shield assemblies.

The meteoroid protection for the shield was reviewed and as a result of the lack of information on the validity of theory associated with the bumper concept

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of meteoroid penetration, the shield casing design was revised to include more positive protection.

A preliminary design of the shield for the FSM-1 system was completed. This design includes a gas expansion space to allow for pressure buildup in the helium cover gas during operation of the system. Internal gas pressure during operation has been calculated to be approximately 44 psia with hydrogen diffusion through the casing expected to total 0.6 lb during the first year.

Preliminary concepts for the SNAP 2 shield support structure and the interconnections between shield and reactor were evaluated. Analysis is continuing.

Preliminary results of the shield assembly thermal analysis indicate that peak temperatures in the shield during operation will be less than 800°F. Analysis is continuing with final completion expected during the coming quarter. Work was initiated during the quarter on evaluation of the physical and chemical properties of lithium hydride, but no pertinent results are available as yet.

C. REFLECTOR AND CONTROL DRIVES

Preliminary layouts of the reflector-drive assembly for the SNAP 10 reactor were completed. Detailed design of the brackets for the assembly is under way, as well as design and layout of the primary beryllium structure.

Two types of drum drive mechanisms will be required for the SNAP 2/10 reactor systems. The first, for high temperature ground tests of nuclear systems, will incorporate a scram spring for the drums together with a scram release clutch and drive motors with position indicators on the drums. After ground testing, this mechanism will be removed and the flight control equipment attached. The flight control mechanism includes stepping motor actuators with gearing on the two active control drums and drive-in springs on the two safety drums.

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V. COMPONENT DEVELOPMENT

A. CONTROL SYSTEM

1. Breadboard Circuit

A breadboard control system, based on digital techniques and using vacuum tubes, was assembled and operated. This system, outlined in Figure 6, includes input and output logic, and timer and timer reset circuits. Parts have been ordered for a second breadboard using transistorized plug-in modules. This technique will allow quick assembly, checkout, and modification of the system. When development is complete, the breadboard circuit together with environmental specifications will be used to procure a flight-qualified package from a vendor.

2. Startup Programmer

Generalized specifications were received for the S2-PSM-1 breadboard programmer. This programmer will provide complete flexibility for system checkout through the use of punched tape timer units and patch panel interconnections. Mounting of parts and wiring of this unit is 65% complete.

3. Shutdown Actuator

The primary mechanism for initiating reactor shutdown is release and ejection of the neutron reflectors surrounding the reactor core. This action causes sufficient reduction in core reactivity, through increased neutron leakage, that the chain reaction is no longer self-sustaining and the reactor is shut down. The trigger for this shutdown mechanism is a thin metal retaining band which holds the two halves of the reflector assembly against the reactor core vessel. When the band is severed by the shutdown actuators, the reflectors are shucked off by the spring-loaded ejector mechanism.

Three types of shutdown actuators have been considered to assure safe launch, operation, and re-entry of the reactor. The ground command actuator will be operable during launching and in orbit when radio contact with the vehicle is practicable. This device is basically an explosive pin-puller which employs an electrically detonated explosive charge to remove a shear pin linking the shutdown trigger band. The pin-puller was selected over other devices such as

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Figure 6. Breadboard Control System - Block Diagram

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solenoid latches and resistance heat switches because of simplicity, low weight, high ratio of energy to weight, and high reliability.

A test facility is currently being fabricated to verify the reliability of the pin-puller and to evaluate the effects of high temperature and vacuum. The end-of-life actuator is a bimetallic thermal expansion device which senses NaK outlet temperature from the reactor, and severs a link in the trigger band when the NaK temperature drops below a preset value. Detailed design of this device is complete and testing will be initiated during the next quarter. The re-entry shutdown actuator is the trigger band itself which is designed and located on the reactor system so that it will be severed during re-entry by aerodynamic heating. The trigger band thus serves as a backup to the end-of-life shutdown actuator, and also assures separation of the reflector assembly during re-entry so that the core vessel and reactor fuel are exposed to re-entry heating. Two designs for trigger mechanism to sever the band are under study. One uses the thermal expansion of the band to sense the rise in temperature and trip an overcenter mechanism to release the band. The other uses silver alloy shear joints which sever the band by melting at a preselected temperature.

A fixture for testing operation of the overcenter device is shown in Figure 7. Good repeatability of the triggering action has been obtained with the device operating at the estimated re-entry temperature of 1200°F for the band.

The test fixture for the shear joint is shown in Figure 8. Triggering repeatability at temperature is satisfactory for this concept but further data on the creep rate of suitable silver brazing alloys is required before it can be incorporated into flight systems designs. Samples of the shear joints using various silver alloys are currently undergoing creep tests at stress levels and temperatures anticipated for these components during orbital operation of the reactor. Other tests to check the effects of high vacuum and radiation on the mechanical properties of the brazing alloys are also under way.

B. CORE VESSEL

1. Mechanical Development

Tests are under way to determine weld joint design and welding parameters for the SNAP 10 core vessel. Both heliarc and electron-beam welding

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Figure 7. Equipment for Testing Overcenter Device Operation

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Figure 8. Test Fixture for Shear Joint

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techniques are being investigated. Efforts are also under way to determine if a forming technique can be used for the thin-walled vessel (1/32-in. thick) to eliminate the need for a circumferential weld as specified by the original design.

2. Hydraulic Tests

Hydraulic testing is under way to insure proper flow distribution of coolant through the reactor core. To facilitate testing, NaK flow is being simulated by water at equivalent hydraulic conditions in accordance with established model theory. A photo of the hydraulic test loop for the SNAP 10 core is shown in Figure 9. The core model with two inlet pipes and one outlet is in the center of the photo. Water is circulated by the pump (not shown), through two flow-meters, through the core model, into a reservoir and back to the pump. Flow and pressure drops are measured by manometers and velocities are determined by a salt solution concentration technique using electrodes installed in dummy fuel elements in the core. Further details of this method are described in NAA-SR-6439.

Testing of the first model of the SNAP 10 core is complete and a summary report is in preparation. Figure 10 shows the NaK velocity profile for this core configuration at a NaK flowrate of 4420 lb/hr and temperature of 900°F. Figure 11 shows the relationship between NaK flowrate and pressure drop over the flow range from 26 to 140% of design flow rate. No attempt was made to influence the shape of the flow profile in this core configuration. Tests are now under way with a new configuration utilizing a baffle plate below the core to produce a flow profile closer to the ideal profile shown in Figure 12.

C. SHIELD

1. Materials Development

Materials development has centered on laboratory investigations of the basic parameters of the lithium hydride compact being used for the SNAP 10A/2 reactor shields. Outgassing and dissociation rates of the hydride are being checked since this affects the gas pressure buildup within the shield container. Tests are being run on cold-pressed specimens over the temperature range from 500 to 1400°F. Preliminary information indicates some unexpected gas release at about 800°F. Test work on this problem is continuing. Test programs

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Figure 9. Hydraulic Test Loop for SNAP 10 Core

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Figure 10. SNAP 10 Hydraulic Core Model - NaK Velocity Profile
(97.0% of Design Flow)

Figure 11. Pressure Drop Across SNAP 10 Reactor with 900°F NaK Flow

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have also been established to measure Young's Modulus and ultimate tensile and compressive strengths of the cold-pressed lithium hydride. Tests will cover the temperature range from ambient to the melting point. Specimens are currently being fabricated at ORNL and are expected at AI shortly.

2. Shield Studies

Studies have been made of the SNAP 2/10 shield configuration to determine the effects and extent of radiation damage and hydrogen loss by diffusion.

A report on the radiation damage effects is in preparation. The work on hydrogen diffusion is continuing but analytical efforts so far indicate that about 0.04% of the total hydrogen in the shield will be lost by diffusion through the SNAP 10A container during its operational lifetime. This exceeds the amount of hydrogen generated by the $\text{Li}^6(n, \alpha)\text{H}^3$ reaction and should therefore prevent any buildup of hydrogen. Helium from the reaction, however, does not diffuse through the container at a measurable rate and may therefore cause some pressure buildup.

D. COMPONENT MATERIALS TESTS

1. Bearing Materials

Several additional combinations of bearing materials were run in the friction-measuring apparatus. Tests were conducted at 1000°F temperature and 10^{-5} mm Hg vacuum. The results of the latent tests are summarized in Table IV. Details of the bearing compatibility tests run to date and the test apparatus for measuring friction coefficients are described in NAA-SR-6476. Table V summarizes the properties of material combinations showing low friction coefficients.

2. Spring Relaxation Tests

Spring testing so far has indicated that Inconel 718 and Rene 41 have such physical properties and creep characteristics that low relaxation rates can be expected at the stress levels and temperature conditions prevalent in the SNAP reactor systems. Early tests on torsion springs of Rene 41 stressed to 48,000 psi in tension at 1200°F have shown only 9% relaxation after 4300 hr. This testing will be extended to include helical springs at various shear stress levels. A photo of the spring test apparatus is shown in Figure 13.

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TABLE IV
BEARING MATERIAL COMPATIBILITY TESTS

Test No.	Rider Sample	Base Sample	Friction Coefficients			
			Room Temperature and Pressure	24 Hours at 10 ⁻⁵ mm Hg and Room Temperature	7 Days at 10 ⁻⁵ mm Hg and 1000°F	Final at Room Temperature and Pressure
XXVI	Al ₂ O ₃ on 316 SS	Haynes 90	0.84	0.46	0.50	0.60
XXVII	Al ₂ O ₃ on 316 SS	C-Graphite	0.20	0.14	0.18	0.14
XXVIII	Deva-3-N-12	Haynes 90	0.46	0.68	0.25	0.33

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TABLE V
LOW FRICTION COMBINATIONS

Test No.		Friction Coefficient Maximum at Vacuum and 1000°F	Coefficient of Expansion (in./in./°F x 10 ⁻⁶)	Hardness Room Temperature	Room Temperature Strength (psi)	
					Compressive	Tensile
XX	C Graphite CDJ-83	.22	1.2	110 R _e	33,000	7,300
	Haynes No. 90		8.6 (32-1472°F)	60 R _c	-	60,000 (annealed)
XIV	Al ₂ O ₃ (Hot Pressed) (or LA-2)	.23	(See test XXV & IX)			
	C Graphite (CDJ-83)		(See test XX)			
XXV	Deva 3-F-12	.23	6.6 (70°F)	36 R _b	30,000	13,000
	Al ₂ O ₃ (Hot Pressed)		4.5 (77-1652°F)	9-9.5 Mohs	450,000	-
IX	Al ₂ O ₃ (LA-2)	.39	3.9 (70-1832°F)	1,000-1,200 (VPN) 300		
	TiC (K162B)		5.3	89 R _a	420,000	93,000
XXIV	Deva 3-N-12	.40	7.4	-	36,000	14,000
	Al ₂ O ₃ (Hot Pressed)		(Estimated for Pure Ni)			
IV	Cr ₃ C ₂ (LC-1A)	.49	6.4 (70-1800°F)	850 (VPN) 300	-	-
	Al ₂ O ₃ (LA-2)		(See test IX)			
X	C Graphite (CDJ-83)	.47	(See test XX)			
	Haynes Stellite No. 3		7.3 (0-1100°F)	55 R _c	310,000	55,000

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Figure 13. Spring Test Apparatus

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3. Gear Drive Tests

Work has begun on evaluation of gear materials and surface coatings for operation under SNAP environmental conditions. Gears of Haynes 90, Deva 3-N-12, and Inconel X materials spray-coated with Al_2O_3 and TiC are being manufactured for test.

4. Self-Weld Testing

During early tests of bearing materials at temperature and high-vacuum conditions, some combinations of materials showed high starting friction after relatively long periods of contact with no movement. Consequently test work was initiated to evaluate the self-welding effect of all materials on the SNAP reactor system that would be required to separate or slide after long periods of static contact. Several combinations of materials with low friction coefficients are currently undergoing static contact tests in a vacuum chamber at 1000°F and 10^{-6} mm Hg vacuum. Photos of this apparatus are shown in Figures 14 and 15. An additional test chamber capable of operating at 1000°F and 10^{-9} mm Hg vacuum is currently under construction. Testing at the higher vacuum condition will be initiated during the next quarter.

E. HIGH-TEMPERATURE SUBSYSTEMS AND COMPONENTS

1. Special Fasteners

Basic tests are in the planning stage to develop application techniques and to determine mechanical properties of beryllium and titanium bolts in beryllium tapped holes for use in SNAP reactor environments. Information on galling properties, shear strength, shock sensitivity, torque efficiency, and optimum torquing techniques will be determined.

2. Temperature Sensors and Microswitches

Environmental testing continues on various components required for the reactor control system. A thermal switch of the differential expansion, rod-and-tube type, which has ultimate application as a thermal sensor, has so far operated successfully on test for 450 hr and 20,000 switching cycles while controlling within 20°F of the set point. Additional units have been ordered for further functional and environmental testing. Testing has also started on a

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Figure 14. Equipment for Static Contact Test, 1000°F and 10^{-6} mm Hg

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Figure 15. Vacuum Chamber for Static Contact Test, 1000°F and 10^{-6} mm Hg

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transistorized bistable amplifier using a TIMM* thermionic diode as the signal source. In addition, two high-temperature microswitches have been running on test at 800°F. Switch No. 1 failed after 3550 hr and 297,000 cycles. Switch No. 2 had operated successfully for 330,000 cycles when the test was terminated to examine the switch.

3. Actuators

The reference design for the control drive actuator has been changed from a rotary solenoid concept to the stepper motor described in the last report. This change was made to reduce the number of mechanical parts in the device which would have to operate in the high-temperature region. Design efforts are under way to optimize the size and weight of the motor for the storage requirements of the SNAP control drive. Two stators for this motor are currently being wound and encapsulated by Westinghouse, whose technique has produced insulation which has functioned successfully for 2500 hr at 1000°F and 10^{-5} mm Hg vacuum.

4. Electronic Parts and Insulation

A free-running multivibrator, to serve as a pulse source for the timer in the control system, was obtained from the General Electric Co., built from their TIMM components. These components utilize miniature heaterless vacuum tubes and other parts capable of operation at 1080°F. The multivibrator has so far successfully operated at temperature for 1900 hr. Testing is continuing.

High-temperature insulation development continues with tests of several commercially available encapsulating compounds which have survived earlier screening tests. A statorette built by GE as part of the 1000°F generator development on the HOTELEC program has been under test for 200 hr at 1000°F and 10^{-5} mm Hg vacuum with no degradation of electrical characteristics.

Inquiries have been made to vendors regarding availability of high-temperature radiation-resistant telemetry transducers to sense angular position in the reflector assembly. Two vendors who claim to have units suitable for 800°F operation are being investigated.

*Thermionic Integrated Micromodules

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F. ENVIRONMENTAL FACILITIES

The high-temperature vacuum chamber has been modified to operate at temperatures up to 1000°F with vacuum of 10^{-9} mm Hg. This facility incorporates a combination of pumping systems consisting of a mechanical forepump, a 16-in. oil diffusion pump with baffles cooled by water and liquid nitrogen, an electronic ion pump, and a cryoplate system using liquid hydrogen shielded by liquid nitrogen. The vacuum chamber itself, shown in Figures 16 and 17, is constructed of 304 stainless steel plate, 1/4-in. thick, rolled into a 4-ft cylinder with an ID of 30 in. The inside surface of the chamber is polished to a mirror finish. Instrumentation is located in a separate tubular section 18 in. in diameter by 18 in. long. This tube, shown in Figure 18, provides all connections for test apparatus inside the vacuum chamber. The tube is mounted on a mobile dolly and is mated to the vacuum chamber by an adapter flange which allows disconnect for instrument maintenance or calibration without destroying the vacuum in the chamber. The internal heating system in the vacuum chamber as shown in Figure 19, consists of twelve 1600-watt tubular quartz lamps. This system provides controlled temperatures in the chamber up to 1000°F. An independent bakeout system mounted on the exterior wall of the chamber provides controlled bakeout temperatures from ambient up to 800°F. The system has been operated on shakedown runs with all vacuum pumps except the liquid hydrogen cryoplate in use. Pressures as low as 8×10^{-8} mm Hg at 1000°F have been maintained during these runs with specimens under test. It is fully expected that the design objective of 10^{-9} mm Hg vacuum will be achieved during the next quarter when the cryoplate pump is placed into operation. Additional information concerning this high-temperature vacuum system is available in NAA-SR-6490.

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Figure 16. Vacuum Chamber, 1000°F and 10^{-9} mm Hg

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Figure 17. Vacuum Chamber, 1000°F and 10^{-9} mm Hg (Back View)

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Figure 18. Tube of High-Temperature Vacuum Chamber

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Figure 19. Heating System of High-Temperature Vacuum Chamber

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VI. FUEL DEVELOPMENT

A. FABRICATION

All SNAP 2/10 fuel rods produced during this quarter were from billets made by the new triple-melting technique. Nine billets were alloyed and extruded during the quarter; two of these were enriched, yielding 14 rods, and seven were normal, yielding 49 rods. As the quarter closed, seven additional billets were in production including one of the new alloy containing 0.5 wt % carbon. To meet the fuel element assembly rate of 20/month with a yield of 50% in hydriding, 40 rods/month must be supplied for hydriding. At the end of the quarter, the melting-extrusion step was operating satisfactorily at this rate.

In the hydriding step, the two brickpile furnaces processed 20 enriched rods during the quarter, of which 11 were uncracked. At the same time, 16 enriched rods and 12 normal rods were hydrided in the new large furnaces, making a total of 28. However, only two rods, both of normal material, were successfully hydrided in these furnaces without cracking. Rods were processed in the furnaces in loads of 6 per furnace, as designed, and in two-rod and four-rod loads, with no increase in production of uncracked rods. By the end of the quarter, it was apparent that hydriding could not be accomplished in the new furnaces at an acceptable yield, and action was taken to achieve the desired production rate of 20 rods/month by installing eight more brickpile furnaces identical in design to the two units currently installed. A target date of December 30 was set for start of production in these new units.

Sufficient fuel rods are currently on hand for 27 elements for the SCA-4 facility. Assembly is scheduled to begin in early October with delivery of a total of 37 elements for SCA-4 scheduled for completion by November 15.

During the quarter, 154 Hastelloy N cladding tubes have been received, 54 seamless from the Le Fiehl Manufacturing Co., and 100 seam-welded from NTH Products, Inc. The present inventory is as follows:

Seamless:	17 coated, ready for use
	27 at vendor for coating
	10 destroyed in shipping accident
Seam-welded:	38 used
	12 coated, ready for use
	50 uncoated, in receiving inspection

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Le Fiell is also fabricating an additional 120 Hastelloy N seamless tubes to be delivered by October 30. At this time, assuming the quality of the delivered tubes is satisfactory, another 120 tubes will be ordered from sheet material now on hand. In addition, 15 Hastelloy N strips are being cold-rolled from 0.020 to 0.015 in. to make welded tubes at NTH. About 45 tubes will be available from this source. Finally, 35 ft of Hastelloy N bar is on order for use in end caps and cups.

B. RESEARCH AND DEVELOPMENT

1. Process Development

Research and development in process development continues toward elimination of fuel rod cracking during the hydriding process. A program has been drawn up aimed at understanding the behavior of the fuel alloy during hydriding with special emphasis on the relationship between degree of cracking and physical properties of the alloy material. Six general areas will be examined as follows:

- a) Effects of Impurities - Effects of oxygen and nitrogen on the kinetics of hydriding will be studied. Stresses due to formation of hydrogen gradients in the fuel alloy caused by inhibiting effects of these gases will be determined and correlated with the degree of cracking.
- b) Dilatometer Measurements - Isothermal expansion of zirconium alloys during hydriding will be measured, and anisotropic growth as a function of specimen geometry will be noted.
- c) Isopleth - A phase diagram of the hydrogen-zirconium - 10 wt % uranium system will be constructed.
- d) Preferred Orientation - The cracking resistance of alloy structures produced by varying the degree of preferred orientation, grain size, and amount of cold work, will be investigated. Effects of these parameters on the kinetics of hydriding will also be noted.
- e) Cracking - Physical measurements, such as dilatometric expansion, and electrical resistivity will be obtained on specimens during the hydriding cycle to pinpoint the start of cracking in the fuel alloy.

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- f) Metallographic Examination - Specimens from various hydriding procedures and pretreatments will be examined metallographically to determine the relationship of particular phases or structures with the degree of cracking.

Concentrated effort on a second approach to the cracking problem was started in late September with development of a fuel alloy containing 0.5 wt % carbon as an additional alloying element. This modification results in an extremely fine-grained structure in the fuel material which is maintained throughout extrusion and hydriding. Work on the GE/ANP program with zirconium has demonstrated that the addition of carbon produces a tougher, stronger alloy which is not so susceptible to cracking during the hydriding process.

Results with this new alloy at AI have so far been very encouraging. Six fuel rods with the 0.5 wt % carbon addition have been hydrided to H/Zr ratios as high as 1.75. The hydriding cycle was patterned after the procedure successfully used on pure zirconium on the ANP program. It included an initial "inoculation" step to an H/Zr ratio of 0.04, prior to the conventional carefully controlled hydriding treatment to the final H/Zr ratio of 1.75.

Two of the 6 fuel rods so treated survived the procedure with no cracks, three had hairline cracks about 1/16 in. deep, and one developed a major crack at a casting flaw. In no case were the carbon-modified as badly cracked as the noncarbon rods which were identically treated as control samples.

Work is continuing with the carbon-modified fuel alloy on investigation of physical properties, dissociation pressure, irradiation behavior, and compatibility with other materials used in the SNAP fuel elements.

Further development of the cup assembly for making an improved seal at the blended end of the fuel element has resulted in specifying the cup blend as standard for the SNAP 2/10 fuel element. The technique consists of shrink-fitting a thin-walled ceramic-coated cup into the open end of the coated cladding tube, over the end of the fuel rod. The cup-tube interface is then blended and the element sealed by welding an uncoated cap in place over the tube. A sketch of the cup-blend assembly is shown in Figure 20.

Comparative shock and vibration tests of fuel elements with and without the cup-blend seal were conducted during the quarter. Results of leakrate measurements following these tests are listed in Table VI. A cluster of 7 fuel elements

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Figure 20. Fuel Element Cup-Blend Assembly

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TABLE VI
LEAKRATES OF FUEL ELEMENTS FOLLOWING
SHOCK AND VIBRATION TESTS

Element No.*	Rod No.	cm ³ /hr Before Vibration and Shock Testing		cm ³ /hr After Vibration and Shock Testing	
		1200°F	900°F	1200°F	900°F
P-2	126-6	2.27	0.030	1.88	-
P-3	135-4	1.15	-	1.56	0.021
P-5	125-1	1.78	-	1.72	0.010
P-6	126-1	1.17	0.025	1.04	0.009
P-7	110-6	0.96	0.029	1.18	0.019
PR-2	110-8	1.10	-	1.21	0.023
PR-4	119-7	0.81	-	0.70	0.024
PR-5	120-7	0.83	-	0.97	0.024
PR-6	130-2	0.87	0.010	3.05	0.041
PR-7	130-3	0.81	0.021	1.04	0.017
PR-8	125-3	0.26	0.004	0.23	-
PR-9	110-9	0.93	0.027	2.22	0.033
PC-8	137-6	0.33	0.027	0.27	0.008
PC-9	107-1	0.59	0.024	1.24	0.018
PC-10	110-5	0.45	0.062	0.34	0.008
PC-11	126-2	1.12	0.018	1.10	-
PC-12	110-3	0.37	0.003	0.43	-
PC-1	120-6	1.32	0.019	0.83	-
PC-2	124-3	1.42	0.023	0.49	-
PC-3	124-4	0.40	-	Rupture†	-
PC-4	124-5	1.34	0.033	1.63	-
PC-5	126-3	0.31	0.005	0.34	-
PC-6	126-5	0.32	-	0.43	-
PC-7	135-6	0.50	0.013	0.66	-

*Designations are as follows:

P = 3-piece S2-DR design.

PR = 3-piece S2-DR design with fuel and reflector having rounded edge.

PC (8-12) = 3-piece with rounded edges and new developmental cup blend.

PC (1-7) = 1-piece with rounded edges and new developmental cup blend.

†Testing furnace ran away when controller malfunctioned.

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was also subject to shock and vibration testing with resulting leakrates as shown in Table VII.

TABLE VII
LEAKRATES OF FUEL ELEMENT CLUSTER
FOLLOWING SHOCK AND VIBRATION TESTS

Element No.	cm ³ /hr Before Cluster Vibration Test	cm ³ /hr After Cluster Vibration Test
PC-1	0.83	1.39*
PC-5	0.34	0.35
PC-6	0.43	0.36
PC-7	0.66	0.70
PC-14	0.57	0.52
PC-16	0.92	1.13
Dummy	--	--

*Note from Table VI that PC-1 returns to original leakrate before any vibration testing - the 0.83 figure therefore is questionable.

Leak tests have shown that the cup-blend seal is performing satisfactorily. Reproducibility of tight blends is excellent and the average leakrate for a fuel element with a cup-blend seal is approaching 0.4 cm²/hr, or about one-half that obtained on the fuel elements used in the S2-DR (SNAP 2 Development Reactor) core loading.

A program controller is being installed in the blending process line to operate the blending cycle on an automatic basis and provide more reproducibility than the manual technique currently in use. An apparatus to speed up measurement of leakrate from finished fuel elements was installed during the quarter. Figure 21 shows the bank of four furnaces used in this equipment and Figure 22 shows the control panel.

A full-size S2-DR-type fuel element has been assembled into an irradiation test capsule, designed to operate at 1200°F fuel-clad temperature to the design SNAP 2 burnup in the WTR (Westinghouse Test Reactor). The element has a central fuel thermocouple and a transducer to monitor pressure buildup

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Figure 21. Permeation Test Furnaces

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Figure 22. Control Panel for Permeation Test Furnaces

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inside the cladding. The cladding is fitted with 9 thermocouples to measure temperature throughout the irradiation cycle. A flux depressor is currently being fabricated for the experiment in order to obtain an even burnup across the fuel diameter. The capsule will be shipped to the WTR upon completion of the flux depressor, with insertion scheduled for October 1961. A second irradiation capsule, duplicating the first, is also being assembled for insertion into the WTR one cycle later than the first.

2. Materials Development

High-temperature X-ray diffraction studies were started to define the phase boundaries of the ZrH system. This work is expected to lead to better understanding of the effects of thermal cycling and the phase changes that occur during hydriding. Two hydrides with H/Zr ratios of 1.59 and 1.73 were studied to 1020°F during the quarter. Work with other hydride components is continuing.

Results of a long-term compatibility test of beryllium and Hastelloy N were available this quarter. Beryllium in contact with Hastelloy N and chromized Hastelloy N was held at 1300°F for 1000 hr in an inert atmosphere. Metallographic examinations showed diffusion zones at the interfaces of both combinations. The zones showed a laminar structure on the beryllium side of the interface and a second phase next to the Hastelloy. The thicknesses of the diffusion zones were measured with the following results:

Beryllium <u>vs</u> Hastelloy N	10.7-21.2 mils
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Beryllium <u>vs</u> chromized Hastelloy N	10.5-10.6 mils
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The results of this study indicate that Hastelloy N and beryllium are not compatible at this temperature.

Hot cell examination of the six spent fuel elements taken from the SER core loading was initiated during the quarter. A check of the overall dimensions of these elements showed no detectable change from the pre-irradiation measurements made before operation in the SER core. Four gamma scans were taken along the length of each element, with the element rotated a quarter turn between scans. All scans were similar in contour. They exhibited no sharp discontinuities which would indicate cracks or separations in the fuel alloy nor did they show distinct increases at the ends, indicative of excessive flux peaking

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from the Be end reflectors in the elements. The center of the flux peak plateau was located approximately 6 in. from the top (hot end) of the 10-in. long fuel rod. Averages of the four scans for each element are given in Table VIII.

TABLE VIII
GAMMA SCANS FOR SER FUEL ELEMENTS

Element No.	Reactor Position	Peak/Average	Average*
M-4	Center	1.20	1.00
T-9	II Ring	1.27	0.62
10-R-8	III Ring	1.24	0.75
10-R-5	IV Ring	1.25	0.72
Q-5	V Ring (outside)	1.21	0.68
U-10	V Ring (outside)	1.22	0.56

*Relative to M-4 as unity

Specimens have been cut from elements 10-R-8 and T-9 for metallographic examination during the forthcoming quarter. The hydrogen leakrate of the elements will also be determined and an apparatus for making these measurements inside the hot cell is currently under construction.

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