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PROGRESS REPORT FOR SNAP
(SYSTEMS FOR NUCLEAR AUXILIARY POWER)
GENERAL SUPPORTING TECHNOLOGY
FEBRUARY - APRIL 1965
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AEC Research and Development Report

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By

L. S. MIMS

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The Progress Report for the previous quarter was NAA-SR-10796.

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I. PROGRAM OBJECTIVE

The objective of the General Supporting Technology Program is to give technical assistance to the SNAP (Systems for Nuclear Auxiliary Power) programs under development. This assistance includes development of improved analytical techniques as well as assembly, development, and correlation of theoretical and experimental data necessary for the successful accomplishment of present SNAP objectives. It

also includes the outline of development problems for the future applications of SNAP to other operational requirements. The General Supporting Technology Program accomplishes its objective through basic research and development in the fields of compact reactor physics, radiation environment definition and advanced shielding studies, heat and mass transfer studies, and applications engineering.

II. SUMMARY OF EFFORT

Nuclear experiments using both clean and regionally poisoned cores were performed using the SCA-5 critical assembly. Comparison of experimental data with analytical predictions showed areas of fair agreement and other areas where significant discrepancies were present.

The PRISM code for calculating multigroup scattering matrices was modified to improve its accuracy and efficiency.

The Tape Conversion Program (TCP) was completed and successfully operated. Data from the Aldermaston Library Tape for sodium and iron were intercompared with similar data from the Atomics International tape.

The application of variational techniques to SNAP nuclear calculations was studied. Proper application of these techniques shows promise of significantly reducing the computation time required for complex problems.

The SNAP Basic Nuclear Technology Advisory Committee was organized. This committee, which met three times during the report period, is serving as a forum for the exchange of information and views on SNAP nuclear analysis problems.

A description of the new 21-group cross section library for use with the DTF code in performing shielding calculations was completed. Certain significant discrepancies are observed in the results obtained using this library and the results produced by older libraries. The source of these discrepancies is being investigated.

The SLAP code, which provides a two-group P_3 treatment for neutron calculations used in the revised slab geometry SHOE shield optimization code, was checked out satisfactorily. A simpler optimization code that treats radiation attenuation as the sum of a series of exponentials was also prepared and checked out.

The ANNEX-3 Monte Carlo shielding code was used to analyze the results of a series of STIR lithium hydride shielding experiments. Very good agreement between calculations and experimental results was obtained.

A preliminary study of the effects of space radiation on selected reactor components indicated that degradation of the high emissivity coating material could lead to a power reduction of 1 to 3%/yr for a SNAP 10A system in a 700-n. mi circular polar orbit.

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The Shield Test and Irradiation Reactor (STIR) achieved full operational status. Shielding experiments utilizing lithium hydride and heavy metal slabs were initiated.

Experiments were initiated to investigate the effects of different gamma and neutron spectra in producing radiation damage in silicon transistors. Preliminary results appear to support the postulate that hard reactor leakage gammas are more damaging than Co^{60} gammas.

Additional digital subroutines were prepared which permit the UPSTART II program to also simulate the startup transient behavior of SNAP 10A systems as well as SNAP 8 systems. Satisfactory operation of the program in simulating a SNAP 10A system startup was demonstrated.

Reduction of noise analysis data obtained from the single channel, crossflow hydraulic model indicated that the factor relating measured crossflow velocity to average crossflow throat velocity is 0.76. Using this relationship, crossflow velocities measured in the SNAP 8 core can be converted to average crossflow throat velocities for use in heat transfer and hydraulic analytical codes.

The investigation of mercury supersaturation behavior was initiated with two concurrent efforts. One part of the investigation is being performed in conjunction with the NAA Science Center and involves the study of the basic condensation phenomenon through use of cloud

chamber techniques. A simple cloud chamber using methyl alcohol at room temperature was assembled and tested. If satisfactory, the results of these tests will provide the basis for the design and construction of a high-temperature, high-pressure cloud chamber for use with mercury. The second part of the investigation involves direct study of the dynamic vapor expansion process. For this purpose, design studies of a continuous flow, nozzle expansion loop were conducted and a preliminary equipment layout was prepared.

The investigation of the effect of mercury vapor at 700 to 900° F and at partial pressures of up to 1 psia on silver wire was unintentionally terminated due to overheating of the test furnace resulting in damage to the test capsules. The duration of exposure at the time of test termination was about 3 months, or three times the original goal of 1 month. There was no significant change in the observed resistivity of the silver wire during the test.

All capsules for the NaK-hydrogen partial pressure experiment were assembled and NaK loading was initiated.

The study of the properties of niobium-lined, Type 316 stainless-steel tubing continued.

"SNAP Technology Handbook, Volume III, Refractory Fuels and Claddings," NAA-SR-8617, Volume III, was issued.

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III. AREAS OF EFFORT

A. NUCLEAR ANALYSIS

The major responsibility of the Basic Nuclear Analysis Project is to ensure a competent analytical approach to nuclear problems within the SNAP reactor projects. This includes not only the fulfilling of current analytical needs, but also the anticipation of future problems arising from the upgrading of SNAP design concepts. This entails the development, evaluation, and application of basic nuclear analysis techniques. Current programs are in the interrelated areas of:

- 1) Critical Experiments Analysis
- 2) Cross Section Development
- 3) Theory and Method Development.

1. Critical Experiments Analysis

The preliminary analysis and specification of the SCA-5 critical experiments program, reported in the previous quarter, has been completed. An outline of the experimental program is given in Table 1. The core diameters and hydrogen densities to be investigated in this program are compared with present SNAP reactors and advanced SNAP concepts in Figure 1. Power flattening (or fission-density flattening) by means of hafnium and europium poisons will be investigated. Some experimental measurements will be compared with calculated values to evaluate analytical techniques; other experiments (e.g., Numbers 4, 5, and 11 in Table 1) will provide data that cannot be readily obtained from one-dimensional calculations.

Preliminary measurements of critical heights, control drum worths, and flux distributions were made using the unpoisoned small core, and using the small core with 1 wt % HfH_2 in the central 6 rings consisting of 91 fuel rods.

Figure 1. Core Diameters and Hydrogen Densities to be Investigated in the SCA-5 Compared with Present SNAP Reactors and Advanced Concepts

In most cases, the core was filled with water to a level corresponding with the top of the fuel. In some cases, however, the water height was changed slightly to adjust the reactivity of the assembly for experimental purposes. In Figure 2, the experimental values of k_{eff} for several core heights are compared with calculated values from one-dimensional, radial S_n calculations using the DTF-II code with 16 energy groups. The axial height in the code is simulated by a transverse buckling. From the figure it may be seen that the calculated values of k_{eff} are 6-1/2% low. Of this difference, 1/2% is attributed to neglecting the 1/4-in. water reflector at the bottom of the core. Other possible sources of error, such as the use of a transverse buckling, and the approximate treatment of the Be (n, 2n) reaction are being investigated.

Measurements of control drum worth using the small [nominal 9-in. diameter (see paragraph B for additional details)] SCA-5 core were performed with results as shown in Table 2.

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TABLE 1
OUTLINE OF SCA-5 MINIMUM EXPERIMENTAL PROGRAM

Core Configurations			1	2	3	4	5	6	7	8	9	10	11
Core	No. of Rings*	Type of Poison	Critical Height	k_{eff} vs Core Height	Control-Drum Calibrations	Control-Drum Interactions	Control-Drum Backings	β_{eff}/k^*	Fission Densities	Danger Coefficient Measurements with 2-6 wt % HfH_2 , 0.3-0.6 wt % Eu_2O_3	Spectrum Measurements	Axial Reflector Thickness	Differential Water Height Worth
Small Core (Total 11 Rings)													
A		Unpoisoned	<u>wd</u>	<u>wd</u>	<u>wd</u>	<u>wd</u>	wd	wd	wd	w	wd	wd	d
B	1-3	1.0 wt % HfH_2	w		w				w				
C	1-6	1.0 wt % HfH_2	<u>w</u>	<u>w</u>	<u>w</u>	<u>w</u>	w	w	w		w	w?	
D	1-3	0.1 wt % Eu_2O_3	w		w				w				
E	1-6	0.1 wt % Eu_2O_3	w		w				w				
F	1-7	0.1 wt % Eu_2O_3	w	w	w	w?		w	w		w	w?	
G	1-6	0.5 wt % $HfH_2^\dagger$	w		w				w				
H	1-7	0.5 wt % $HfH_2^\dagger$	w	w	w	w?		w	w				
Large Core (Total 14 Rings)													
I		Unpoisoned	wd	wd	wd	wd	wd?	wd	wd	wd?	wd	wd	d
J	1-3	1 wt % HfH_2	wd		wd				wd				
K	1-7	1 wt % HfH_2	wd		wd				wd				
L	1-8	1 wt % HfH_2	wd	wd	wd	wd?	wd?	wd?	wd		wd?		
M	1-3	0.1 wt % Eu_2O_3	wd		wd				wd				
N	1-6	0.1 wt % Eu_2O_3	wd		wd				wd				
O	1-8	0.1 wt % Eu_2O_3	wd	wd	wd	wd?		wd?	wd		wd?		
P	1-8	0.5 wt % HfH_2^*	wd		wd				wd				

*Central fuel rod is ring 1; the surrounding 6 rods form ring 2; etc.

† Formed by alternating poisoned (1 wt % HfH_2) and unpoisoned pellets in fuel rods.

w = water-filled core

d = dry core

w = underlined letter denotes experiment already completed

? = optional experiment

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Figure 2. Effective Multiplication Factor as a Function of Fuel Height for the Small, Water-Filled SCA-5 Core

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TABLE 2
REACTIVITY WORTH OF CONTROL
DRUMS No. 1 AND 3, SMALL
SCA-5 CORE

Configuration	Unpoisoned Fuel		Poisoned Fuel (1 wt % HfH ₂) in Rings 1 to 6	
Drum No. 1 (\$)	1 57	2 06	2 08	2 36*
Drum No. 3 (\$)	1 53	2 00	2 08	2 32*
Drum No. 2 Position	165° 30' Out	Full In†	Full In†	Full In†
Water Height (in)	12-3/8	11-1/16	14-3/4	12-7/8

*Additional fuel rods in cusps between drums for this measurement

†Most reactive position

The results in Table 2 show that in the unpoisoned core, the position of the central drum (Number 2) can affect the reactivity worth of the two outer drums to the extent of about \$0.50. It is also seen that the worths of drums 1 and 3 in the poisoned core (1 wt % HfH₂ in Rings 1-6) are essentially the same as in the clean core. The poisoned core in this case, however, is over 2.5 in. higher than the clean core (see Figure 2). The last column shows the effect on drum worth of adding fuel rods to the small cusp regions between the drums. The effect here is to increase drum worth by an average of \$0.26 while lowering the water level slightly to limit excess reactivity.

The results of measurements of fission densities and the episcadmium activation of gold foils in the small SCA-5 core are given in Figures 3 and 4. Above cadmium cutoff, 90% of the gold activation is due to the 4.95-ev resonance. The shape of the episcadmium gold activation agrees well with the calculated neutron flux in the 3- to 10-ev region. There is less agreement between the calculated and experimental fission densities. The calculated fission density is based on the summation of fission rates in all neutron energy groups. The

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Figure 3. Relative Fission Densities in Small, Unpoisoned SCA-5 Core

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Figure 4. Relative Fission Densities and Episcadmium Activities in Small SCA-5 Core with 1 wt % Poison in Fuel Rods of Rings 1-6

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measured fission density is also based on fissions occurring at all neutron energies. It is measured by attaching uranium-zirconium alloy pins to the exterior of the fuel rods and measuring the post-irradiation gross fission product gamma activity for gammas of energy greater than 0.45 Mev.

2. Cross Section Development

A number of changes were made in the PRISM code to improve its accuracy and increase its efficiency. These changes have decreased the running time of the code by approximately 25%. Checkout of modifications made in the PRISM code to calculate P-1 energy transfer matrices was continued.

The programming of the transformation matrices which allow transformation of Legendre coefficients between the center-of-mass and laboratory frames of reference was completed and checked out. The presently available matrices allow for transformations of arrays of up to 29 by 29.

The TCP (Tape Conversion Program) code completed its first successful run this quarter. Data from the Aldermaston Library Tape for sodium and iron was intercompared with similar data from the AItape. Several differences in cross section data were noted.

The modification of the nonspherical optical model code, 2 PLUS, was completed with the addition of a capability to produce graphical displays of experimental and computed angular distributions as well as an option to punch cards for use in the CHAD code. The scattering of neutrons from U^{238} in the region between 10 kev and 1.5 Mev was analyzed with encouraging results.

Using parameters computed by 2 PLUS, preliminary cross section data were calculated for Pu^{238} and Cm^{244} .

The mechanisms by which multigroup libraries may be rapidly constructed were improved and refined. Comparison, of such libraries, with a series of critical experiments provides means for evaluating the library group constants directly and the basic microscopic data indirectly. Continued progress in the programming and improvement of the basic programs, CHAD, PRISM, and TCP, is bringing the concept of a highly automated cross section program closer to fruition. Modifications to existing nuclear model programs, graphical display of results, and punch-out options to provide card data for use with processing programs have enhanced the usefulness of these programs.

3. Method Development

Variational techniques provide an efficient method for approximating the solutions to integral-differential equations in cases where the general characteristics of at least a part of the solution are known beforehand. The application of these techniques to the multidimensional time- and energy-dependent transport equations for problems of SNAP reactor calculations were investigated during the past quarter.

The basis of the variational method is that a function, $u(\vec{r}, E, \vec{\Omega}, t)$, which is the solution to the transport equation,

$$G(u) = \left[\vec{\Omega} \cdot \vec{\nabla} + \frac{1}{v} \frac{\partial}{\partial t} + \Sigma(\vec{r}, E) \right] u(\vec{r}, E, \vec{\Omega}, t) \\ - \int_E \int_{\vec{\Omega}'} S(\vec{r}, E, E', \vec{\Omega}, \vec{\Omega}') u(\vec{r}, E', \vec{\Omega}', t) \\ \cdot dE' d\vec{\Omega}' = 0 \quad , \quad \dots (1)$$

with appropriate boundary conditions, renders stationary an integral function,

$$I(u, u^+) = \int_r \int_E \int_{\vec{\Omega}} \int_t u^+ G(u) d\vec{r} dE d\vec{\Omega} dt \quad \dots (2)$$

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In Equation 1, $\Sigma(\vec{r}, E)$ is the total cross section at position $\vec{r}$ and energy E . The kernel, S , includes both scattering and fission sources. The effect of delayed neutrons is neglected in Equation 1. In Equation 2, integration is performed over all energies, E , directions of neutron travel, $\vec{\Omega}$, the whole reactor volume, and the time of interest, t . For the moment, $u^+(\vec{r}, E, \vec{\Omega}, t)$ may be considered an arbitrary weight function. If the exact solution u is known, $I(u, u^+) = 0$, regardless of the form of u^+ , however, if we replace the exact solution u , u^+ by approximate solutions

$$\bar{u} = u + av, \quad \bar{u}^+ = u^+ + bv^+, \quad \dots (3)$$

where v, v^+ are arbitrary functions (except perhaps at the boundaries), and a, b are "small" constants, we find that

$$I(u + av, u^+ + bv^+) = I(u, u^+) + aI(v, u^+) + bI(u, v^+) + abI(v, v^+), \quad \dots (4)$$

since $I(u, u^+)$ is linear in both u and u^+ . Note that $I(u, v^+) = 0$ from Equations 1 and 2. We integrate differential terms in the integrals of $u^+G(v)$ by parts noting that v, v^+ vanish on the surface of the reactor and at $t = 0$, where t_0 is the final time of interest. This results because $\bar{u}, \bar{u}^+$ are exactly specified by the boundary conditions. We also change limits on the integration terms to obtain the adjoint equation:

$$\begin{aligned} I(v, u^+) &= \iiint u^+ G(v) d\vec{r} dE d\vec{\Omega} dt \\ &= \iiint v G^+(u^+) d\vec{r} dE d\vec{\Omega} dt, \quad \dots (5) \end{aligned}$$

where

$$\begin{aligned} G^+(u^+) &= \left[-\vec{\Omega} \cdot \vec{\nabla} - \frac{1}{v} \frac{\partial}{\partial t} + \Sigma(\vec{r}, E) \right] u^+(\vec{r}, E, \vec{\Omega}, t) \\ &\quad - \int_E \int_{\vec{\Omega}'} S(\vec{r}, E', E, \vec{\Omega}, \vec{\Omega}') u^+(\vec{r}, E', \vec{\Omega}', t) \\ &\quad \cdot dE' d\vec{\Omega}'. \quad \dots (6) \end{aligned}$$

If we now specify that the "exact" weight function (the adjoint) u^+ satisfies the equation $G^+(u^+) = 0$ (with appropriate boundary conditions), we see that

$$\begin{aligned} I(u + av, u^+ + bv^+) &= I(u, u^+) \\ &\quad + abI(v, v^+), \quad \dots (7) \end{aligned}$$

so that first-order errors in u, u^+ cause only second-order perturbations in I .

A direct application of Equation 7 occurs in the time-independent case where we seek an eigenvalue k_{eff} which multiplies the fission term in S . Then Equation 2 can be solved directly for k_{eff} with the knowledge that errors in k_{eff} caused by approximations in u, u^+ will be only of second order.

A potent application of the variational technique is the synthesis method, which we illustrate by space-time synthesis for a super-prompt critical transient. We approximate u, u^+ by

$$\begin{aligned} u(\vec{r}, E, \vec{\Omega}, t) &= \sum_1 R_1(\vec{r}, E, \vec{\Omega}) T_1(t) \\ u^+(\vec{r}, E, \vec{\Omega}, t) &= \sum_1 R_1^+(\vec{r}, E, \vec{\Omega}) T_1^+(t), \quad \dots (8) \end{aligned}$$

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where the R_i, R_i^+ are several flux shapes from quasi-stationary calculations of the asymptotic flux at a few (3-5) points during the transient. Equations 8 are inserted in Equation 2, and integrations over space, angles, and energy are performed directly. Requiring that the resulting integral be stationary in time leads to equations

$$\frac{dT_i(t)}{dt} = \sum_j A_{ij} T_j(t) \quad \dots (9)$$

for the T_j , where the matrix, A_{ij} , is given by

$$\begin{aligned} \|A_{ij}\| = & - \left\| \int_{\vec{r}} \int_E \int_{\vec{\Omega}} R_i^+ R_j \frac{1}{v} d\vec{r} dE d\vec{\Omega} \right\|^{-1} \\ & \cdot \left\| \int_{\vec{r}} \int_E \int_{\vec{\Omega}} \left[R_i^+ (\vec{\Omega} \cdot \vec{v} + \Sigma) R_j \right. \right. \\ & \left. \left. - R_i^+ \int_E' \int_{\vec{\Omega}} S R_j dE' d\vec{\Omega}' \right] d\vec{r} dE d\vec{\Omega} \right\|. \end{aligned} \quad \dots (10)$$

A similar use of the synthesis method may be used in the time-independent case to obtain a two-dimensional flux for a cylindrical reactor:

$$u(r, z, \xi, \mu) = \sum_i R_i(r, \xi, \mu) Z_i(z, \mu).$$

Here $\xi = \vec{\Omega} \cdot \hat{z}$ and $\mu = (\vec{\Omega} \times \hat{z}) \cdot (\hat{r} \times \hat{z})$.

The R_i are taken from one-dimensional radial calculations at various planes through the cylinder perpendicular to the Z-axis.

Applications of the variational technique to reactor problems reported in the literature are mainly confined to the diffusion equations, because the reactor systems described were large enough for the diffusion approximation to be valid. There is no a priori reason why the

variational technique cannot be applied to the transport equations with similar accuracy and savings in computation time. Details of applications to R-Z space synthesis and to space-time synthesis will be investigated during the next quarter.

4. SNAP Basic Nuclear Technology Advisory Committee

Based on the beneficial results derived from the SNAP interproject meeting on common nuclear analysis problems sponsored by the General Supporting Technology Program in the preceding quarter, a permanent committee was organized. This committee is known as the SNAP Basic Nuclear Technology Advisory Committee. Its purposes are to provide a broadly based forum for the exchange of current nuclear analytical and experimental information between projects and to review current nuclear analysis problems and needs and recommend appropriate action. Three meetings of the committee were held during the past quarter. Topics which were considered included: (1) committee purpose and functioning, (2) current nuclear analysis activities and needs, (3) inspection of the SCA-5 critical assembly, and (4) review of the SCA-5 experimental program.

B. NUCLEAR EXPERIMENTS

The experimental program using the small (nominal 9-in. diameter) SCA-5 core was continued. Preliminary calibration experiments with the clean, wet core were completed. These included measurements of fission densities and gold detector epicadmium fluxes, control drum differential and total reactivity worths, and control drum interaction effects. The six central rings of fuel consisting of 91 fuel rods were then replaced by fuel containing 1 wt % HfH_2 poison. Using this partially poisoned core, the value of k_{eff} was measured for several fuel heights. Measurements of fission densities,

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epicadmium fluxes and control drum reactivity worths, and drum interactions effects were also completed for this configuration. Some of these results are presented and compared with analytical predictions in Section III. A. 1. (Also, see Table 2 and Figures 2, 3, and 4.)

Preliminary measurements were made to determine the effect of reproducibility errors on reactivity measurements in the SCA-5 critical assembly. The results of these measurements indicated that the uncertainty in reactivity determinations may be about 4%. The causes of this uncertainty are being investigated. Meanwhile, experiments are being designed so that expected changes in reactivity will be large compared with the uncertainty. The parts comprising the fuel rods used in the SCA-5 assembly are shown in Figure 5. The rod at the bottom of the figure shows the lower fitting which is used to position the fuel rod in the lower grid plate. The fitting is fluted to permit the passage of water up through the circular hole in the lower grid plate and into the fueled region of the core. The center rod in the figure shows the top of the fuel rod with the seal plug removed. The upper rod shows the top of the fuel rod with the seal plug assembled.

C. SHIELDING ANALYSIS

1. Cross Section Development

A description of the new 21-group, cross section library for use with the DTF code in performing shielding calculations was completed. A description of the use of this library in calculating the neutron distribution around a point neutron source in water was also completed. The discrepancy in the results predicted for the low energy range by the new 21-group and the 16-group libraries and by moments method calculations has not yet been resolved. In an effort to explain these differences, additional simple problems will be

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Figure 5. SCA-5 Fuel Rod Details

run with finer mesh point spacing. Also, the DTF code results will be compared with DTK code results to determine whether errors may exist in the newer DTF version.

2. Code Development

The SLAP code, which provides a two-group, P_3 treatment for neutron calculations used in the revised slab geometry SHOE shield optimization code, was satisfactorily checked out. It is now being revised to incorporate four neutron energy groups. It is also being converted from FORTRAN-II to FORTRAN-IV machine language in anticipation of forthcoming changes in the AI computer system. A simpler optimization code that treats radiation attenuation as the sum of a series of exponentials was also written, checked out, and is now in use in SNAP 8 shielding studies.

The ANNEX-3 Monte Carlo code was used to check the results of neutron attenuation experiments in lithium hydride, which were performed at the STIR. Very good agreement with experiment was obtained, indicating that the code may be quite useful in this type of analysis.

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3. Space Radiation Effects

A preliminary study of the effects of space radiation on SNAP reactor systems was completed. The results of this study indicate that on the basis of those components considered, viz., transistors, high reflectivity coatings, and high emissivity coatings, space radiation would cause a degradation in electrical output to about 1 to 3%/yr for a SNAP 10A system in a 700-n.mi, circular polar orbit. The primary cause of degradation is reduction in the efficiency of high emissivity coatings due to low-energy proton irradiation.

D. SHIELD EXPERIMENTS

1. STIR Shield Experiments

Startup tests for the Shield Test and Irradiation Reactor (STIR) were completed, and the reactor was placed in full operational status. Neutron fluxes and gamma dose rates were measured in the vicinity of the thermal column and fission plate. Following this, the attenuation of fast neutrons by a set of lithium hydride slabs from ORNL was measured. The neutron collimator and heavy shield slab carriage were then installed, and attenuation measurements using depleted uranium and tungsten slabs were begun.

2. Radiation Effects on Semiconductors

As reported in the preceding quarterly report, a review and analysis of radiation effects data for semiconductors has indicated that the damage from SNAP reactor leakage gammas may be up to four times that produced by an equal dose of Co^{60} gammas. The study also indicated that the optimum shield weight for a given total dose would be achieved by allocating a greater percentage of the shield weight to gamma shielding. If these preliminary findings are confirmed, a revision of present payload shielding criteria is indicated.

In order to experimentally test this finding, 15 silicon transistors of a single type were exposed to a dose of 4×10^6 rad of hard Cd gammas produced by neutrons from the thermal column of the STIR. Preliminary results qualitatively support the predicted "hard" gamma effect. At the close of this report period, preparations were underway to perform a similar irradiation using Co^{60} gammas.

As another part of this preliminary study, equal numbers of two types of silicon transistors (2N717 and 2N744) were irradiated in two different neutron spectra. The mountings for the transistors are shown in Figure 6. All of the mountings were placed inside temperature-controlled, oil-bath capsules. The "In-Pool" unit was subjected to the thermal neutron spectrum in the water adjacent to the STIR core. The erroneously labeled "Thermal Column" unit was in fact subjected to the fast neutron spectrum emerging from the fission plate which was located on the outer surface of the STIR thermal column. The "Control" unit was not irradiated but was otherwise maintained under identical environmental conditions. Data obtained from these tests are currently being analyzed.

E. CONTROL AND STABILITY

Digital Simulation of SNAP System Transient Behavior

During this quarter, several digital subroutines were prepared for use with the UPSTART-II program. These new subroutines permit digital simulation of the dynamics and heat transfer of the SNAP radiator-condenser, the SNAP thermoelectric pump, and the heat losses and flow delays associated with SNAP coolant pipe sections. In addition, it was determined that the fuel cladding could not be lumped with the fuel in the analytical model without

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Figure 6. Mounting of Silicon Transistors for Irradiation in Different Neutron Spectra

adversely affecting the accuracy of the simulation. The core thermal model was therefore modified to provide separate nodal representation of the cladding. Several of the original UPSTART subroutines were modified to conform with the more versatile and efficient format used in developing the newer subroutines.

With the development of the new subroutines listed above, the UPSTART-II program could now be used to simulate the transient behavior of SNAP 10A/2 systems as well as SNAP 8 systems. The subroutines were assembled to

simulate a SNAP 10A configuration, and the startup transient behavior of this system was analyzed. Examination of the results of this simulation revealed that, while the program operated satisfactorily, the subroutine for the thermoelectric pump was not utilizing the most efficient form for the fluid flow equations and that this was adversely affecting computer running time. The equation form for fluid flow was modified and the transient was rerun. The program ran in a satisfactory manner and analysis of a 30,000-sec startup transient was completed in 1 min and 42 sec of computer time.

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Figures 7 through 14 show the results of UPSTART-II simulation of a SNAP 10A system startup for the period between 22,000 and 30,000 sec after startup. Figure 7 shows the behavior of the reactor thermal power. The power rises to about 60 Btu/sec before heat generation terminates the rise, after which the thermal power undergoes rapidly convergent minor oscillations as it approaches the steady-state power level. The corresponding changes in reactivity are shown in Figure 8. The fuel temperature changes which provide the primary effect on reactivity during the initial rise to power are shown in Figure 9. For this analysis, the fuel temperature was averaged over five equal axial regions. Region 1 represents that 20% of the fuel at the inlet end of the core; Region 5 represents that 20% of the fuel at the outlet end of the core.

The behavior of the control drum is shown in Figure 10. Drum position is controlled through the drum controller by the coolant out-

let temperature. During the portion of the transient shown in Figure 10, the coolant outlet temperature has not yet reached the set-point value. At a result, there is no feedback and the drum is shown to be in the process of being driven to a more reactive position at a constant rate. The irregularities in the curve are the result of the program formulation and time sequencing and do not represent actual drum steps.

The variations in the reactor coolant inlet and outlet temperatures are shown in Figure 11. These are seen to closely parallel the averaged fuel temperatures. The variation of the coolant flow, as provided by the thermoelectric pump, is shown in Figure 12. As can be seen by reference to Figure 11, flow rate is a sensitive function of coolant outlet temperature. The effect of flow rate on coolant temperature rise through the core is also apparent upon careful examination of Figures 11 and 12.

Figure 7. Reactor Thermal Power vs Time After Startup

Figure 8. Net Reactivity vs Time After Startup

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Figure 9. Regionally Averaged Fuel Temperatures vs Time After Startup

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Figure 10. Control Drum Angle vs Time After Startup

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Figure 11. Inlet and Outlet Coolant Temperatures vs Time After Startup

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Figure 12. Coolant Flow vs Time After Startup

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Figure 13. Average Temperature of Thermoelectric Material vs Time After Startup

Figure 13 shows the time behavior of the average temperature of the thermoelectric material. This is seen to closely parallel the variations in the coolant outlet temperature. The same is true for the average radiator temperature shown in Figure 14.

F. HEAT AND MASS TRANSFER

1. One-Phase Hydraulics

Effort under this project has been directed toward completion of data reduction for the single channel, crossflow hydraulic model, modification of the cross-correlation analyzer to improve accuracy and reduce analysis time, and identification of error sources in velocity measurements. A summary report was in preparation describing the development and application of noise analysis techniques to the measurement of reactor coolant velocities.

Data obtained from the single channel, crossflow model tests indicate that the factor relating

Figure 14. Radiator Temperature vs Time After Startup

the measured velocity (i.e., distance between electrodes in adjacent dummy fuel elements divided by the measured transit time) to the average crossflow throat velocity in the gap between elements is approximately 0.76 for the geometry used in the test model. Since this geometry is patterned after that used in SNAP 8, the crossflow velocities measured in the SNAP 8 core can be converted to average crossflow throat velocities for use in heat transfer and hydraulic analytical codes.

The modification of the cross-correlation computer was accomplished by replacing the smoothing amplifier with a true integrating amplifier. This modification improves the accuracy of the correlation coefficient calculation and permits more rapid and efficient analysis of data. A block diagram of the modified cross-correlation computer is shown in Figure 15.

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Figure 15. Modified Cross-Correlation Computer Data Reduction System

An error analysis of the velocity data obtained using the salt injection noise analysis technique was performed. A block diagram showing probable sources of error, their interrelationships, and estimated magnitudes is given in Figure 16. The main source of error is due to differences in electrode response characteristics. To verify this conclusion, tests were performed using a special, single channel hydraulic model where transit times were measured with flow in one direction and then in the opposite direction. Differences in transit time values for a given pair of electrodes for forward and reverse flow were as large as 78%. One half of this difference represents a measurement of the actual difference in response times between the pair of electrodes. Other possible sources of systematic error associated with the hydraulic apparatus are difficult to evaluate in the presence of these electrode problems. The results of this study indicate that the conductivity cell used in the past has a large error associated with it, particularly when used with close spacings and for high flow rates.

2. Two-Phase Hydraulics

Based on the high surface tension of mercury, theoretical analysis indicates that mercury vapor can exhibit an exceptionally high degree of supersaturation in the process of expansion. There is little direct experimental evidence, however, to quantitatively substantiate this prediction. As a result, considerable uncertainty exists as to the exact thermodynamic behavior of mercury vapor in a turbine. This uncertainty is reflected in the design of mercury turbines with the consequence that maximum efficiency is not assured. In addition, reliability is possibly jeopardized due to the difficulty in accurately predicting the point at which condensation, and hence erosion, will occur. The purpose of this study, therefore, is to experimentally investigate mercury supersaturation behavior, and thereby provide the basis for the design of mercury-vapor turbines exhibiting greater efficiency and reliability.

This experimental study is being conducted with two concurrent efforts: (1) cloud chamber basic studies of the expansion and condensation

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Figure 16. Probable Sources of Error in Noise Analysis Measurements of Fluid Velocity

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Figure 17. Diagram of Basic Cloud Chamber Apparatus with Short Tube, High-Pressure Chamber

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Figure 18. Short (8-1/4 in.) and Long (37-3/4 in.) Cylinder High-Pressure Chambers

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processes, and (2) engineering analytical studies utilizing a continuous flow, nozzle expansion loop.

a. Cloud Chamber Studies

The cloud chamber studies are being performed in conjunction with the NAA Science Center. During this report period, a glass cloud chamber apparatus has been assembled and operated. This apparatus, using methyl alcohol (CH_3OH) as a working fluid, is intended to provide basic data for use in the design of a high-temperature, high-pressure chamber for use with mercury. Methyl alcohol was chosen for this preliminary apparatus because of its high vapor pressure (2 psia) at room temperature.

A sketch of the basic cloud chamber device is shown in Figure 17. It is designed to function in a manner similar to that of a shock tube, and shock tube analytical relationships are used to predict vapor behavior during the rapid expansions. During initial tests of the cloud chamber, a long (37-3/4 in.) high-pressure chamber was employed, as shown in Figure 18. With this configuration, only a small amount of condensation was observed. The lack of abundant condensation was attributed to the low cooling efficiency of the system, where cooling efficiency is defined as the ratio of the observed cooling to the theoretical cooling that would occur under adiabatic conditions for the same expansion ratio. One of the major factors affecting the value of the cooling efficiency is the quantity of heat transferred to the vapor by the walls of the chamber. This, in turn, is affected by the surface-to-volume ratio (S/V) of the high-pressure region and the expansion time, t . To investigate this effect, three other high-pressure chambers of differing geometries were tested; these were the short (8-1/4 in.) tube shown in Figure 18 and the 0.5- and 2.0-liter flasks shown in Figures 19 and 20, respectively.

The S/V values and experimentally determined expansion times, t , of these systems are shown in Table 3.

TABLE 3
EXPANSION CHARACTERISTICS FOR
VARIOUS CHAMBERS

Chamber	S/V (in. ⁻¹)	Expansion Time, t (msec)
Short tube	3.50	65
Long tube	3.25	65-75
0.5-liter flask	1.65	83.88
2-liter flask	0.98	100-110

A plot of cooling efficiency vs the product $(S/V)(t)$ is shown in Figure 21 for the latter three configurations. Examination of the figure shows the good correlation provided by the index $(S/V)(t)$. Figure 22 shows the oscilloscope record of temperature and pressure variations during an expansion using the short tube, high-pressure chamber.

The improved cooling efficiencies provided by the 0.5- and 2-liter flask geometries resulted in substantially increased condensation. Further operational experience revealed, however, that the condensation associated with a given expansion ratio decreased or disappeared over a period of time involving a number of successive expansions. Theoretical analysis based on the well defined thermodynamic properties of pure methyl alcohol vapor also showed that the degree of condensation observed exceeded that which would be predicted. Accordingly, it was concluded that the observed condensation was heterogeneous in nature and was probably due to impurities in the system which were gradually being scrubbed out in the course of chamber operation. Present efforts are directed toward obtaining faster expansion times by replacing the present 3/4-in. ID ball valve with a 2-in. ID unit. It is expected that

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Figure 19. 0.5-Liter High-Pressure Chamber
Installed on Expansion Device

this will permit higher cooling efficiencies which, when combined with better elimination of impurities, will permit observation of true homogeneous nucleation, i.e., true supersaturation. These data and experience will then be applied to the design, fabrication, and operation of a high-temperature, high-pressure cloud chamber for use with mercury.

If the problems associated with the construction and operation of a mercury vapor cloud chamber appear too difficult, a diffusion chamber of a type currently being built at the NAA

Science Center will be utilized. A sketch of this diffusion chamber is shown in Figure 23. The chamber consists of a heated lower plate, where evaporation of the vapor of interest occurs, and a cooled upper surface, where condensation of this vapor occurs. A noncondensing gas is also used in the chamber to maintain a constant chamber pressure and to avoid convection by providing a decreasing density from the bottom of the chamber to the top. Supersaturated conditions are obtained, since the saturation pressure is an exponential function of temperature while the partial

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Figure 20. 2.0-Liter High-Pressure Chamber
Installed on Expansion Device

pressure and temperature are linear functions of height. Figure 24 illustrates the predicted supersaturation pressure ratio as a function of height in a diffusion chamber under the specified conditions.

b. Continuous Flow Nozzle Expansion Loop

The second aspect of the study of mercury vapor supersaturation involves the design, construction, and operation of a mercury vapor, continuous flow, nozzle expansion loop. In this study, the nozzle will be equipped with transparent walls to permit optical measurement of mercury droplet size and concentration during the expansion process. Preliminary studies of loop arrangement, nozzle design, and optical instrumentation requirements were completed. A report describing the preliminary design of

the loop, nozzle, and instrumentation was written. Figure 25 shows the planned arrangement of the nozzle expansion loop within the safety enclosure. (The safety enclosure is required due to the high toxicity of mercury vapor.) Figure 26 is a schematic diagram of the optical system that will be used to measure droplet size and concentration prior to and during expansion.

3. Silver-Mercury Diffusion Experiment

This experiment, designed to determine the effect of mercury vapor (at 700 to 900°F and pressures up to 1 psia) on the electrical resistivity silver wires, was unintentionally terminated near the end of this report period. Failure of the furnace controls resulted in overheating of the furnace and damage to the test capsules and specimens. The failure occurred after

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Figure 21. Chamber Cooling Efficiency vs Product of Surface-to-Volume Ratio Times Expansion Time

Figure 22. Oscilloscope Record of Pressure and Temperature Variations in High-Pressure Region During Expansion Using Short Tube, High-Pressure Chamber

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ACCESS TUBE FOR
EVACUATING, PRESSURIZING,
AND INTRODUCING LIQUID
OF INTEREST INTO CHAMBER
FOUR TUBES 90° APART

VAPORIZED COOLANT
TO CONDENSER

ACCESS TUBE FOR
THERMOCOUPLE TWO
TUBES 180° APART

ACCESS TUBE
FOR
THERMOCOUPLE
FOUR TUBES
90° APART

Figure 23. Diffusion Chamber Under Construction at NAA Science Center

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Figure 24. Supersaturation Profile Theoretically Attainable in a Diffusion Chamber Containing Mercury and Helium Gas

about 3 mo of exposure. The initially planned duration of the test was 1 mo; however, when no change in silver resistivity was observed after an exposure of 1 mo, it was decided to continue the test until a change in resistivity was observed. At the time of failure, no significant change in resistivity had yet been observed. Data obtained in the course of the test are being assembled for inclusion in a final report. Unfortunately, the damage to the capsules will prevent post-test metallographic examination of the specimens.

4. NaK-Hydrogen Partial Pressure Experiment

All six capsules for the NaK-hydrogen partial pressure experiment were completely assembled this quarter. All welds were helium leak-checked, dye-penetrant-tested, and x-rayed where practical. One such completed assembly consisting of a 9-in., 321 stainless-steel formed bellows with appropriate end closures and internal tubing is shown in Figure 27. When loading the capsules, it is necessary to maintain a 100%, void-free NaK system inside the bellows assembly at test temperature so that all hydrogen entering the NaK will remain in the atomic state. It was therefore decided to load the capsules at test temperature and, additionally, to circulate the hot NaK through the capsules to obtain an ultraclean system. A special vertical NaK loading manifold was designed so that

stresses on the capsules would be minimized and good NaK flow during cleanup would be assured. Sealing of the capsules could not be accomplished at temperature, as planned, due to safety considerations. A revision to the loading manifold design was therefore necessary. In the revised version, the capsules were mounted and supported horizontally on the same centerline as the inlet and outlet valving to assure no crossflow of NaK either into or out of the capsules while cooling. Thus, the void-free capability was not sacrificed and stresses were minimized. Due to the delay involved in redesigning the loading manifold, the NaK loading cart was diverted to another commitment, causing further delay in the NaK loading phase of the program. Calibration of the RGA ion pump system was completed and furnace profiles were obtained during this delay period. When the NaK loading cart again became available, the six capsules were loaded with NaK at temperatures of 1075, 1170, 1260 (2 capsules), and 1320°F (2 capsules). The 1400°F temperature goal previously set could not be attained due to insufficient temperature capability of the NaK loading cart. Figure 28 shows the capsules in place in the NaK loading manifold. As of the close of this report period, one capsule had been installed in the testing apparatus and the system was being pumped down and checked out. Following this, the capsule will be heated to 1260°F (the NaK loading temperature), out-gassed, and hydrogen testing will begin.

5. Duplex Tubing Materials Study

The study of the properties of niobium-lined, Type 316 stainless-steel tubing was conducted in three areas: (1) study of the diffusion kinetics of the metallurgical bond between the niobium and the stainless steel, (2) study of the effect of mechanical deformation on bond integrity, and (3) study of the effect of hydrogen on the niobium liner.

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NOTES:

1. (P) DENOTES A PRESSURE MEASUREMENT
(T) DENOTES A TEMPERATURE MEASUREMENT
2. ALL Hg PIPING BETWEEN THE BOILER AND THE CATCH TANK WILL BE OF HAYNES-25 ALLOY
3. Hg VAPOR LINES WILL BE 2in. PIPE
Hg LIQUID LINES WILL BE 1/2in. SCH. 10 S, 316 STAINLESS-STEEL PIPE

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Figure 25. Preliminary Equipment Arrangement for Continuous Flow Nozzle Expansion Loop

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NOTES :

P = PHOTOMULTIPLIER TUBE

I_n = INTENSITY FROM P-n

POWER SUPPLIES FOR LAMPS AND PHOTOMULTIPLIERS
NOT SHOWN

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Figure 26. Schematic Diagram of Optical System for
Continuous Flow Nozzle Expansion Loop

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Figure 27. Capsule Assembly - NaK-Hydrogen
Partial Pressure Experiment

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Figure 28. NaK Loading Apparatus and Capsules

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a. Diffusion Kinetics

Sections of the duplex tubing, approximately 3/4 in. long, were prepared and metallographically examined. Each transverse section was scanned for defects. A representative area was selected and marked, and the general characteristics of the bond were noted and measured. The specimens were then sealed in a vycor capsule at a pressure of less than 10^{-5} mm Hg.

After being placed in furnaces, the temperatures of the capsules were recorded once each minute for the duration of the test, with the exception of the 1960°F specimen which was monitored continuously. The time at temperature was defined as beginning when the specimen reached the desired temperature and concluding when the specimen had been removed from the furnace and quenched in water. After removal from the capsule, the transverse face of each specimen was ground and metallographically examined. In some instances after metallographic examination, the specimens were reencapsulated, given additional heat treatment, and examined again.

The results of these studies to date are shown in Table 4. The thicknesses shown are the average of ten measurements made from a 500X photomicrograph of each specimen. Calculations will be made from these results to determine the life of the bond at various temperatures.

b. Bond Deformation

After the diffusion couples described in paragraph a. had been examined, specimens with different thicknesses of bond diffusion layers were selected for mechanical testing. The tubing was compressed between two parallel plates until the bond failed or the tubing was flattened. A microscope was used during compression to determine the onset of bond failure.

TABLE 4

SUMMARY OF RESULTS OF DIFFUSION TESTS OF NIOBIUM-LINED, TYPE 316 STAINLESS-STEEL TUBING
(Diffusion couples were sections of coextruded Nb/316 stainless-steel tubing)

Test Temperature (°F)	Time at Temperature (hr)	Thickness of Metallographically Observed Diffusion Zone (10 ⁻³ in.)
1960	2	0.50
1960	4	0.76
1960	8	0.82
1960	12	0.98
1870	5	0.49
1870	8	0.50
1870	16	0.68
1870	25	0.76
1760	15	0.50
1760	30	0.60
1760	60	0.69
1760	177	1.09
1650	75	0.38
1650	150	0.43
1650	350	0.61
1550	100	0.17
1550	200	0.22
1400	200	0.06
1400	400	0.10
1300	1000	0.08

None of the specimens with metallographically observed diffusion zones of less than 0.0003 in. thickness at the bond interface failed, even when the tube was completely flattened. Bond failure occurred after only small deformation when the zone was greater than 0.0005 in. thick. A void in the "as-received" bond did not appear to propagate upon deformation.

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1960	12	0.98
1870	5	0.49
1870	8	0.50
1870	16	0.68
1870	25	0.76
1760	15	0.50
1760	30	0.60
1760	60	0.69
1760	177	1.09
1650	75	0.38
1650	150	0.43
1650	350	0.61
1550	100	0.17
1550	200	0.22
1400	200	0.06
1400	400	0.10
1300	1000	0.08

None of the specimens with metallographically observed diffusion zones of less than 0.0003 in. thickness at the bond interface failed, even when the tube was completely flattened. Bond failure occurred after only small deformation when the zone was greater than 0.0005 in. thick. A void in the "as-received" bond did not appear to propagate upon deformation.

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c. Hydrogen Effects

To investigate the effect of hydrogen on the niobium liner, a 10-in.-long section of the co-extruded tubing was sealed at one end with a welded plug 1/2 in. long, and a 1/4-in. vacuum lead tube was welded to the other end. The specimen was placed in a furnace which imposed a thermal gradient (600 to 1200°F) along its length. The inside of the specimen was evacuated to a pressure of about 0.3 mm Hg, and 1 atm of hydrogen gas was injected into the furnace around the outside of the specimen. A hydrogen mass spectrometer on the vacuum line monitored the permeation of the hydrogen from the outside of the specimen to the inside. The hydrogen permeation rate was measured approximately daily. The temperature profile along the specimen was periodically measured to assure that the gradient had not changed.

Calculations were performed to estimate the hydrogen permeation rate through each of the components of the duplex tubing and to estimate the hydrogen partial pressure at the Nb/SS interface. Calculation of permeation rates treated the niobium and stainless-steel membranes separately and assumed a hydrogen driving pressure of 1 atm in each case. The results of these calculations are given in Table 5. Calculation of the hydrogen partial pressure at the interface assumed a hydrogen pressure of 1 atm at the outside surface of the stainless steel and 5×10^{-4} atm at the inside surface of the niobium. The calculated hydrogen partial pressures at the Nb/SS interface are also given in Table 5.

After a soaking period of over 1000 hr, the specimen was removed from the test rig and examined. Metallographic and hardness measurements indicated that brittle Nb-H phase or

TABLE 5
CALCULATED INDIVIDUAL PERMEATION RATES, AND HYDROGEN PARTIAL PRESSURE AT THE NIOBIUM/STAINLESS-STEEL INTERFACE

Temperature (°F)	Calculated Hydrogen Permeation Rate (cm ³ /hr/cm ²) (H ₂ ΔP = 1 atm)		Calculated H ₂ Partial Pressure at Nb/SS Interface (atm)
	Through SS	Through Nb	
600	0.003	0.056	2.8×10^{-3}
700	0.008	0.36	4.8×10^{-4}
800	0.020	3.0	4.4×10^{-5}
900	0.040	800	2.5×10^{-9}
1000	0.075	960	6.1×10^{-9}
1100	0.135	1184	1.3×10^{-8}
1200	0.22	1380	2.5×10^{-8}

phases had not formed. The specimen and test conditions are under further study.

A topical report is being written to summarize all results obtained during fiscal year 1965. It is planned that this report will be published during the next reporting period.

6. SNAP Technology Handbooks

The topical report, "SNAP Technology Handbook, Vol. III, Refractory Fuels and Claddings," NAA-SR-8617, Vol. III, was published and distributed. It contains currently available property data on high-temperature, nuclear reactor compounds (UC, UN, and UO₂), ceramic dispersions (Al₂O₃-UO₂ and BeO-UO₂), and cermet (Mo-UC, -UN, and -UO₂) fuels, and potential claddings (Cb, Ir, Mo, Re, Ta, Ti, V, and W). The unclassified portion of the report, "SNAP Technology Handbook, Vol. II, Hydride Fuels and Claddings," containing properties of potential cladding materials for SNAP hydride reactors was reprinted and published as NAA-SR-MEMO-11149, "Properties of Cladding Materials for SNAP Hydride Reactors."

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IV. EVALUATION OF EFFORT

Initial experimental data generated with the SCA-5 critical assembly are considered to be of substantial value in meeting the objectives of this program. These experiments show promise that through proper regional poisoning of the core, significant improvements in radial power flattening may be achieved. These experiments also indicate that present analytical techniques are reasonably accurate in predicting epicalcium fluxes in the present SCA-5 system. Serious deficiencies, however, are also revealed, emphasizing the importance of and need for this investigation. These include the current inability to predict k_{eff} for this assembly to better than 6.5%, and a very poor prediction of the radial fission density distribution. Further efforts in this area will be directed toward identifying the causes of these discrepancies and incorporating the necessary corrections.

In connection with the above, the completion and successful operation of the Tape Conversion Program (TCP), which permits utilization and comparison of cross section data from other laboratories with those of Atomic International, will permit general improvement of cross section data used in analysis of SNAP and other type reactors. This improvement, in turn, will aid in the resolution of the discrepancies noted above. Another facet of this problem is the development of multigroup, cross section libraries which are prepared for the specific system being studied, rather than the use of a general purpose multigroup library which may introduce errors. With the improvement of the PRISM code, which is used for the calculation of scattering matrices, a further step is taken toward the achievement of improved multigroup libraries which in turn result in improved theoretical analysis.

The organization of the SNAP Basic Nuclear Technology Advisory Committee is intended to provide a forum for the exchange of views of those involved in SNAP nuclear analysis and to develop a consensus as to those problems in nuclear analysis which most urgently require attention. Initial indications are that significant improvements in nuclear analysis techniques will accrue as a result of the committee's operation. Similarly, the SNAP Shielding Advisory Committee met several times during the quarter. These meetings also are producing better common understanding of shielding analysis problems and improvements and enhanced utility of analytical techniques.

The Shield Test and Irradiation Reactor (STIR) achieved full operational status and basic shielding studies with lithium hydride and heavy metal slabs were initiated. These tests will provide valuable experimental data for use in improving shield analytical techniques.

Irradiation tests to investigate the effects of different neutron and gamma spectra in producing radiation damage to silicon semiconductors were initiated. Preliminary results appear to support the contention, reported in the preceding quarterly, that reactor leakage gammas are more damaging to silicon semiconductors than the Co^{60} gammas usually employed in radiation damage studies. As noted in the preceding quarterly report, if this finding is confirmed, a reassessment of SNAP shielding criteria will be indicated.

The additional subroutines prepared for the UPSTART-II digital code have increased its utility so that it may now be used for startup transient analysis of both SNAP 10A and SNAP 8 systems.

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The destructive termination of the silver-mercury diffusion experiment will unfortunately prevent metallographic examination of the silver wire specimens for evidence of changes in crystalline structure or silver-mercury reactions. The experiment did contribute information in the gross sense, however, to the extent that there is no significant change in resistivity when the silver wire is exposed to mercury vapor at 700 to 900°F and at vapor pressures up to 1 psia for periods up to 3 mo. This finding will permit greater confidence in assessing the probable performance of electrical equipment (using silver conductors) which is potentially subject to exposure to hot mercury vapor.

A number of unavoidable technical delays were encountered in initiating the NaK-hydrogen

partial pressure experiment. However, with the loading of the capsules proceeding smoothly at the close of this period, generation of experimental data will commence early in the next report period. These data are expected to provide a basis for evaluating the feasibility of the hydrogen getter concept.

The publication of the topical report, "SNAP Technology Handbook, Volume III, Refractory Fuels and Claddings," NAA-SR-8617, Vol. III, marks the completion of the initial publication of the originally planned series of materials handbooks. These handbooks will make a significant contribution to SNAP technology by helping to provide an evaluated, up-to-date and consistent set of data upon which to base performance analyses.

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V. OBJECTIVES FOR NEXT REPORTING PERIOD

The objectives for the next quarter are as follows:

1) Nuclear Analysis:

- a) Publish a report describing the DTF-II code
- b) Modify the PRISM code to provide anisotropic energy transfer matrices
- c) Combine GRAVE, PRISM, and CONTRACT codes to provide automated production of multigroup, cross section libraries.

2) Nuclear Experiments:

- a) Upon completion in early July of work presently being performed for another SNAP project, experimentation with the SCA-5 will be resumed. Experiments will include investigation of another poison configuration for power shaping.
- b) Manufacture of additional fuel pellets for the SCA-5 will be initiated. These pellets are required for the study of some of the extreme configurations planned for inclusion in the SCA-5 experimental program.

3) Shield Analysis:

- a) Complete the evaluation of the suitability of the DTF transport code for use in shielding calculations with the new 21-group cross section library. Initiate modifications or improvements that may be indicated
- b) Issue topical reports on "Space Radiation Damage to Reactor Components," and "Evaluation of Neutron Attenuation in Lithium Hydride Using the ANNEX-3 Code."

4) Shield Experiments:

- a) Continue shielding measurements at the STIR using various combinations of heavy metal slabs and lithium hydride

b) Complete irradiation of silicon transistors by Co^{60} source gammas to determine gamma spectrum effects on radiation damage to semiconductors. Irradiate silicon transistors in STIR pool and with hard cadmium gammas (conversion of thermal column neutrons by cadmium sheet) to investigate spectrum effects on semiconductor recovery. Complete reduction of data from previous tests.

5) Control and Stability:

a) Complete preparation of digital subroutines simulating the SNAP 8 auxiliary heat exchanger and parasitic load resistor. These subroutines will permit study of S8DS startup transients.

b) Complete preparation of a digital subroutine simulating a modified SNAP radiator-condenser unit. This subroutine will permit study of the startup transient behavior of the modified SNAP reactor to be used at the ORNL Tower Shield Facility for space shielding studies.

6) Heat and Mass Transfer:

- a) Publish the report, "Development of Noise Analysis Techniques for Measuring Reactor Core Velocities," NAA-SR-11193
- b) Publish the report, "Deterioration of Silver in Mercury Vapor," NAA-SR-MEMO-11192
- c) Publish the report, "Metallurgical Study of Nb/316 SS Duplex Tubing," NAA-SR-11191
- d) Complete construction and checkout of the continuous flow, nozzle expansion loop to be used in studying mercury vapor supersaturation during rapid expansion through a typical nozzle.

NAA-SR-11096

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