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GENERAL SUPPORTING TECHNOLOGY
FEBRUARY-APRIL 1966
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AEC Research and Development Report

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Program Feldman, Div of Class. Work.
by *OG* DTIE, date *5/24/73*

Exempt from CCRP Re-review Requirements
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GENERAL SUPPORTING TECHNOLOGY
FEBRUARY-APRIL 1966
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L. S. MIMS

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Page 11, 4th paragraph, lines 3 and 7

Hallstrand's should be Hellstrand's

Page 54, 4th paragraph, last two sentences should read:

Comparison with calculations has disclosed that the analytical techniques are seriously inadequate for predicting the source distribution of secondary gamma rays in uranium. Further measurements may be necessary so that adequate models can be developed which yield proper analytically determined functions.

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I. PROGRAM OBJECTIVE

The objective of the General Supporting Technology Program is to develop in a continuing effort basic technology required to meet SNAP (Systems for Nuclear Auxiliary Power) program goals. This 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 study of SNAP applications to mission requirements. The General Supporting Technology Program accomplishes its objective through basic research and development in the fields of compact reactor physics, radiation environment definition, advanced shielding studies, heat and mass transfer studies, and applications engineering.

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II. SUMMARY OF EFFORT

A paper entitled, "The Sensitivity of SNAP Reactor Calculations to Uncertainties in the Zirconium Absorption Cross Section," was presented at the Conference on Neutron Cross-Section Technology in Washington, D.C. The paper reported the need for refined measurements of the zirconium absorption integral.

The checkout of the routines for the 100-GT code which is necessary to calculate cross sections in the SUMMIT group structure was completed. A numerical inaccuracy affecting the present 23-group Zr and Be cross sections was investigated and removed from the routines.

A survey has been made of the literature to determine what work has been done in the field of perturbation theory as applied to transport theory calculations. Works by several authors were reviewed.

The investigation of cylindrical quadrature coefficients with monoenergetic P_0 scattering calculations of a SNAP reactor was completed. The study will be extended somewhat to include P_1 scattering calculations in cylindrical geometry.

The updating and expansion of the Atomic International Evaluated Nuclear Data File (AIENDF) was continued. Data for U^{235} , Pu^{239} , oxygen, aluminum, deuterium, and hydrogen were punched on IBM cards.

The total gamma ray dose rate behind several thicknesses of lead shielding was calculated and compared to the Shield Test and Irradiation Reactor (STIR) experiments. The neutron flux through several thicknesses of lead was also calculated, using a newer set of lead cross sections in the DTF-II transport code.

The complete version of 05R Monte Carlo code, including subroutines specifically written for SNAP-Tower Shield Facility (TSF) was compiled in cooperation with personnel at ORNL. Code 6 was run to obtain the 05R system data tape with a more complete description of the media in the reactor. Program SNARLS was written and checked to analyze the 05R collision tapes and to prepare a leakage neutron source tape to be used in 05R calculations of neutron penetrations in the SNAP 2 shield.

Measurements of the attenuation of fast neutrons in uranium-LiH shields were completed for LiH thicknesses of 0, 9.5, and 24.1 cm and from 0 to 14 cm

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of uranium. Similar measurements were initiated in lead-LiH shields. In order to better understand the production of secondary gamma rays in uranium, measurements were made on the source distribution of capture gammas and fission gammas. Preliminary measurements were made of the gamma leakage spectra from shield assemblies, using a scintillation spectrometer.

Molten lead containment materials test capsules were designed. The materials to fabricate the capsules were ordered and the test matrix was established.

The aggregate gamma shield, designated Experimental Shield No. 2, containing a layer of granular tungsten-alloy gamma shield material and two simulated NaK pipes, was visually examined and x-rayed after remelting and recasting operations were successfully completed.

A number of test rods were prepared and were sectioned to provide specimens for long-term (up to 4000 hr) ZrH-LiH compatibility tests. The first test (1000 hr at 800°F) was completed and the specimen was removed from the capsule and visually examined. Chemical analysis of the specimen will be made when other tests are completed.

In the study on the effects of nuclear irradiation of LiH, an analysis was nearing completion on the number of lattice displacements created in LiH by energetic charged particles. Calculations were made of the number of lattice displacements in natural LiH and 0.9999 Li^7H shields, using fluxes obtained from DTF calculations made for the S8DS shield.

In the Li-LiH bonding and corrosion study, metallographic examination was completed on three stainless-steel capsules containing equal volumes of Li and LiH at 600, 900, and 1200°F for 4000 hr. In the compatibility study of the behavior of molten Li metal heated in contact with LiH, the 1000-hr test at 800 and 1000°F was concluded; the sealed, argon-filled capsules were opened; and the cylinder of cast LiH with a central cavity of Li metal was visually examined.

Hydrogen permeation from a NaK capsule at a temperature of 1075°F was measured. The results were compared to the data yielded from the three higher-temperature capsules (1320, 1260, and 1175°F, respectively) as reported in the progress report for the previous period. Prior to removal of the

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1075°F capsule, a completely static experiment was conducted to determine how fast equilibrium would be reached if a finite supply of hydrogen was admitted to the nickel tube and the pressure drop was monitored in conjunction with the pressure rise outside the stainless-steel NaK chamber. The spare 1320°F NaK capsule was loaded into the test port and will be run initially at 1075°F to verify the previous capsule's behavior. Subsequently, the temperature will be raised to 1320°F to complete the test program.

The nozzle expansion system for the mercury supersaturation study was completed and the instrumentation checkout was in progress.

The diffusion cloud chamber was successfully operated at the NAA Science Center, and data were recorded for methanol, water, and ethanol. From these data, a computer program, allowing for mass diffusion and thermal conductivity in the chamber, determined the variation of supersaturation with temperature for each chamber condition. Improvements in operation were made as a result of experience. A system providing a time history of individual thermocouple readings was installed. Insertion of a metering valve on the helium pressure line to the freon system allowed chamber conditions to be varied by changing the condensing surface temperature.

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TABLE 1
ZIRCONIUM ABSORPTION RESONANCE INTEGRAL MEASUREMENTS

Authors	Year Reported	Reference	Infinite Dilution Value (b)
1. R. L. Macklin H. S. Pomerance	1956	Proc. 1st Geneva Conference, Vol 5, p 833	3.0
2. P. Dobrinin G. A. Dorofeev I. E. Kutirov	1957 1957	Sov. Jour. At. E., <u>3</u> , 1139	2.3 ± 0.5
3. V. B. Klimentov V. M. Griazer	1959	J. Nucl. E., <u>9</u> , 20	3.7 ± 0.5
4. R. B. Tattersall et al.	1960	J. Nucl. E., <u>12</u> , 32	1.0
5. E. Hellstrand G. Lindahl G. Lundgren	1962	Phys. of Fast and Inter. Reactors, Vol 1, IAEA, 1962, p SM-18/79	0.85 ± 0.15

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Figure 1. Natural Zirconium Total Cross Section

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III. AREAS OF EFFORT

A. NUCLEAR AND CONTROL ANALYSIS

1. Reactor Theory and Method Development

a. Zirconium Resonance Integral

A paper, "The Sensitivity of SNAP Reactor Calculations to Uncertainties in the Zirconium Absorption Cross Section," was presented at the Conference on Neutron Cross-Section Technology in Washington, D. C., March 22-24, 1966. The paper reported the need for refined measurements of the zirconium absorption integral.

About 30% of the atoms in the zirconium-hydride SNAP reactor cores are zirconium. Sixty percent of the atoms are hydrogen and 1.3% are U^{235} .

Zirconium occurs in nature as isotopes 90, 91, 92, 94, and 96. The absorption cross-section information on natural zirconium and its isotopes is not extensive. The representation for zirconium absorption in cross-section libraries is generally based on a low $1/v$ component and on the small absorption resonance integral (0.4 ev to 2 Mev).

Reported values for the infinitely dilute absorption resonance integral have ranged from 3.7 barns down to a more recent estimate of 0.85 barns (see Table 1). Sher,^{(1)*} evaluating these results at a time when Hallstrand's⁽²⁾ measurement was still preliminary, reported an estimate of 1.0 ± 0.3 barns. There, he tended to discount the earlier measurements because of problems, such as large corrections due to Hf impurities. This amount of uncertainty typifies all such measurements except Hallstrand's.

A large component of the resonance integral is probably present in the 300-ev resonance (see Figure 1).⁽³⁾ This has been assumed in several cross-section libraries.^(4,5) The worth of zirconium absorption per barn at 300 ev in a SNAP U-ZrH reactor is 3.0% $\Delta k/k$ (by multigroup transport theory code calculation). Assuming the uncertainty in the resonance integral to be 0.3 barns, the uncertainty in a SNAP reactor calculation would then be $\pm 0.9\%$ $\Delta k/k$, without considering the degree of resonance self-shielding. With proper treatment of the

*Superscripts refer to the References listed at the end of this report.

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self-shielding it would be less; however, the value of the resonance self-shielding will depend on whether or not the absorption resonance integral is distributed over the resonance energy range or concentrated in a single resonance such as the 300-ev resonance. Without this information the uncertainty cannot be reduced.

This uncertainty is important in reactor design when one attempts to analyze reactors that are markedly different from those for which experimental information is available. It is also a major factor when one attempts to resolve the compensating errors present in the absolute calculation of critical reactors ($k = 1.0$).

b. Beryllium Scattering Law

The investigation into scattering models for beryllium was continued. The 100-GT code presently uses a single phonon model, which is adequate at low temperatures, but is probably inadequate at higher temperatures. A free-gas kernel will be added to the 100-GT code in FY 1967 to provide adequate representation at higher temperatures.

c. Evaluation of Angular Quadrature Coefficients for Transport Analysis of a Cylindrical SNAP Reactor

In using the discrete ordinate approximations to the neutron transport equation, a satisfactory set of angular quadrature coefficients must be employed. Several sets of coefficients have been advanced for use in cylindrical geometry.^(6,7) The purpose of this study was to determine the set best suited for SNAP reactor analysis.

The sets considered were the two recommended by Lathrop and Carlson⁽⁶⁾ and the one given by Lee.⁽⁷⁾ These may be designated as the Even-Moment Satisfying, Completely Symmetric Set (EM), the Legendre-Tschebyshev Set ($P_n T_n$), and the Angular Segmentation Set (S_n), respectively.

Each of the three sets was used in a monoenergetic calculation of k_{eff} for a SNAP 8 reactor. Table 2 describes the nuclear model. Axial leakage terms have been included in the macroscopic cross sections.

The results of the calculations (see Figure 2) show that the $P_n T_n$ and S_n sets give very similar results. The EM set appears to reach its asymptotic

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TABLE 2
SNAP 8 ONE-GROUP NUCLEAR MODEL

Zone	1	2	3	4	5
Radial Width (cm)	11.04	0.67	0.16	0.18	12.00
Description	U-ZrH core	BeO reflector	Core vessel	Gap	Be reflector
Total Mesh Internals	11	3	3	3	18
Macroscopic Cross-Sections:					
σ_a	0.0368	0.0104	0.0261	0.0084	0.0355
$\nu\sigma_f$	0.0535	0.	0.	0.	0.
σ_{tr}	0.4238	0.3306	0.5148	0.0084	0.5962
σ_{gg}	0.3870	0.3202	0.4887	0.	0.5607

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Figure 2. Evaluation of Cylindrical Quadrature Coefficients With Monoenergetic k_{eff} Calculations of a SNAP 8 Reactor

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value much sooner. This would be highly desirable if the asymptotic value were the exact value. The exact value is unknown; it probably lies between the $P_n T_n$ and the EM calculations based on S_n calculations by Lee.⁽⁷⁾

Thus, in contradistinction to what others have shown for slab and spherical geometries,^(6,7,8) none of the three methods shows clear advantages here in cylindrical geometry for the P_0 scattering approximation. This study will be extended to examine deficiencies of coefficients in anisotropic (P_1 scattering) calculations.

Figure 3 compares the accuracy achieved with computer running time (FORTRAN-II language, IBM-7094 computer, DTF-II code). For situations requiring great accuracy, the S_6 approximation may be chosen over the S_4 with very small increase in computer time.

The close proximity of S_6 results to the near-asymptotic S_8 results has also been evidenced in prompt neutron calculations,⁽⁸⁾ as shown below.

Figure 3. Optimization of Computation Accuracy With Time for One-Group Calculations of a SNAP Reactor

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CANEL Critical Assembly	Prompt Neutron Lifetimes (normalized to S ₈ results)		
	S ₂	S ₄	S ₆
1	1.70	1.16	1.02
2	1.27	1.13	1.04
3	1.48	1.14	1.02

d. ANISN, a FORTRAN-IV S_n Transport Theory Code With the P₁ Scattering Option

The ANISN code was obtained from the Union Carbide Corporation and checked out on the NAASYS computing system. This code represents a FORTRAN-IV version of the DTF-II code, which is necessary for operation with the new IBM-360 computer. It also contains a higher-order P₁ scattering option, providing with the GAMLEG⁽¹⁰⁾ code, an important tool for shielding calculations.

e. Zirconium Hydride Thermal Neutron Spectra

The investigation of the spectral temperature coefficient in SNAP reactors was continued. Recent spectral measurements, as well as calculations with the latest Zr-H_x cross sections, indicate significant changes in the spectrum at neutron energies above 0.4 ev. It was concluded that the present 23-group thermalization library does not properly account for these changes.

A new DTF/ANISN library is being prepared, applying the 100-GT code, which will use the Zr-H_x cross sections to full advantage. The recently revised Hansen and Roach fast cross sections are being used as a basis for this work. The 100-GT code accepts these cross sections as input and adds the appropriate thermal cross sections.

Results from this work are expected to be complete within the next reporting period. Numerical difficulties in 100-GT affecting the flux calculations were corrected. This is expected to make a small improvement in the resulting flux spectra.

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2. Cross-Section Development

a. Multigroup Library Development

The production of sets of multigroup constants (multigroup libraries) has been temporarily discontinued in order that the microscopic nuclear data files may be updated and expanded. It is anticipated that the task of updating the nuclear files will coincide with the availability of the advanced automated cross section processing programs so that the production of desired multigroup libraries will fall into a standard operational procedure.

b. Data Processing Program Development

Checkout calculations with the full and complete GRISM code indicated 21 areas where modifications and corrections were desirable. Of these, modifications and corrections for 8 areas have been completed, a 9th area is partially completed, and 12 areas remain to be worked on.

Coding of the PRISM-II program was completed. The written code remained to be checked and debugged, extensively checked out, and documented (user's manual).

c. AI Evaluated Nuclear Data File

Work in updating and expanding the AIENDF was continued. Data for U^{235} , Pu^{239} , oxygen, aluminum, deuterium, and hydrogen have been placed on IBM cards preparatory to updating and expanding the AIENDF.

B. NUCLEAR EXPERIMENTS (Basic Reactivity Effects)

The formulation of an experimental program to be performed using the SNAPTRAN-1 machine, as adapted for service as a high-temperature critical assembly, was completed. The program consists of 11 basic areas of investigation:

- 1) Clean critical experiments at elevated temperatures
- 2) Control drum calibration at elevated temperatures
- 3) Isothermal temperature coefficients of reactivity
- 4) Mechanical effects on isothermal temperature coefficients of reactivity

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- 5) Power distribution and neutron spectra as a function of temperature
- 6) Hydrogen leakage effects
- 7) Hydrogen redistribution effects
- 8) Temperature coefficients of burnable poisons
- 9) Power shaping by burnable poisons
- 10) Effective delayed neutron fractions
- 11) Nuclear properties of power-shaped cores.

Preliminary experimental procedures were prepared for the first ten of the above investigations and, based on these procedures, the feasibility of the measurements was established. The procedures for the eleventh investigation would be an adaptation of those used in the first ten.

A program schedule and cost estimate were also prepared. These, together with the above experimental programs, the previously prepared evaluation of the adaptability of the SNAPTRAN-1 machine to serve as a high-temperature critical assembly, and a justification of the need for this experimental effort, were assembled into a letter for submission to the AEC.

C. SHIELDING ANALYSIS

1. STIR Experiments Planning and Analysis

The gamma ray transmission through lead slabs exposed to the STIR fission plate source was analyzed. The results of the analyses and the measured values are compared in Figure 4.

The dose rate due to gammas emitted by the fission plate is predominant near the plate. This contribution to the total dose was computed with QAD, a ray tracing code. The energy-dependent, linear gamma ray absorption coefficients and buildup factors used in the code are available in the literature. The agreement near the source plate of the calculated fission plate dose rates with the measured STIR dose rates confirms the literature values.

The measured dose rate beyond several inches of lead is due almost entirely to gammas produced by neutron capture and inelastic scattering. To calculate

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Figure 4. Gamma Ray Dose Rate Through Lead

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these contributions with QAD, a secondary source was generated and the literature absorption coefficients and buildup factors were used. The dose rates shown in Figure 4 do not extend to large enough thicknesses of lead to determine how accurate the secondary gamma calculation is. It appears that the calculation may be slightly low. Since the reliability of the literature data was confirmed by means of the fission plate calculation, any disagreement beyond several inches of lead of the calculated dose rate with the measured dose rate can be attributed to the secondary source.

The secondary source is dependent upon the neutron flux through the shield, the interaction cross sections, and the spectra of the produced gammas. A transport code, DTF-II, was used to calculate the neutron flux through several thicknesses of lead. Neutron capture and inelastic scatter cross sections are available in the literature. Data on the spectra of the produced gamma rays from inelastic scattering are quite sparse. The data on gammas emitted from neutron capture are limited to the 7.4-Mev gamma released on thermal neutron capture. In spite of these uncertainties, the dose rate due to secondary gammas appears to be calculated reasonably well.

In view of the apparent success in predicting the dose rate through 7 in. of lead, it is recommended that the measurements be extended a few inches so that

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the dose rate will be due almost entirely to secondary gammas. Then, the reliability of this method can be more fully established. It is also suggested that either a new code be written or that DTF-II or QAD be modified to eliminate the laborious hand calculations and the time delay of several days in getting results from one code to feed into the other.

2. Tower Shield Facility (TSF) Experiments and Analysis

The complete version of 05R Monte Carlo code has been compiled, including subroutines specifically prepared for SNAP-TSF fission distribution calculations with a single velocity thermal group. Program SNARLS has been prepared and checked to read the 05R collision tape and to prepare leakage neutron source distribution output tapes.

The new 05R version contains 16 media and 16 scatterers, providing for adequate description of the SNAP reactor scatterers. A new version of CODE 6 (05R system data tape) has also been prepared. Program NNPCOM was used to process zirconium inelastic scattering data for use in 05R.

ORNL is preparing to run adjoint S_n calculations on the SNAP TSF and SNAP 2 shield to provide proper biasing for 05R.

D. SHIELD EXPERIMENTS

The Basic Shielding Experiments Program was continued during the reporting period with measurements of the attenuation of fast neutrons in laminated shields. A Hurst dosimeter was used in these experiments biased to give a low energy cutoff of 0.5 Mev. A constant source-detector geometry was employed with variable thicknesses of shield material interposed. Measurements in uranium-lithium hydride shields have been completed for LiH thickness of 0, 9.5, and 24.1 cm and from 0 to 14 cm of uranium. The results of these measurements, which include uranium preceding and following the LiH, are shown in Figure 5. The data give an attenuation length of about 6.7 cm for uranium. Placing the LiH in front of the uranium reduces the attenuation length to 5.3 cm. This reduction is probably due to changes in the neutron spectrum produced by the LiH.

A similar set of measurements has been initiated in lead-LiH shields. The measurements are incomplete; but those which were completed through lead

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Figure 5. Fast Neutron Attenuation in Depleted Uranium and Lithium-Hydride Normalized to 1-kw Fission Plate Power

without LiH and with 9.5 cm of LiH show a behavior similar to the uranium measurements. The effect of transposing the lead and LiH appears to be less significant and the initial attenuation in lead only shows a more pronounced inflection. Work is continuing on these measurements.

In order to better understand the production of secondary gamma rays in uranium, measurements were made on the source distribution of capture gammas and fission gammas. Uranium-238 foils were placed between the slabs of uranium shields of varying thicknesses. The foils were irradiated and counted; the activities due to U^{238} (n, fission) and U^{238} (n, γ) were measured separately. The fission activity was determined by counting the gamma activity of the gross fission products above 0.5 Mev shortly after irradiation, while the radiative captures were measured by counting the 103-keV plutonium x-ray resulting from the decay of Np^{239} . The latter was counted after approximately 24 hr of decay to minimize fission product gamma interference. The results of these measurements are given in Figures 6 through 10. In each case, the activity has been arbitrarily normalized to unity at the fission plate. Of particular significance is the change in slope of the radiative capture distribution with increasing thickness. The DTF-II code, a transport theory code using the S_n approximation, was used to calculate similar configurations. Very good agreement was achieved with the

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Figure 6. Neutron Activation in 2.54-cm Depleted Uranium Slab

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Figure 7. Neutron Activation in 5.08-cm Depleted Uranium Slab

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Figure 8. Neutron Activation in 8.89-cm Depleted Uranium Slab

Figure 9. Neutron Activation in 11.43-cm Depleted Uranium Slab

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Figure 10. Neutron Activation in 13.97-cm Depleted Uranium Slab

fission distribution, but the calculations overestimated the radiative capture distribution at large thicknesses by a factor of 3 or more.

A brief test was performed to evaluate the use of a Stilbene crystal scintillation detector for neutron spectroscopy by proton recoil detection. A special circuit was utilized for proton-electron discrimination by means of the shift in the cross-over point for a double delay, line-differentiated pulse. The circuit rejected all gamma ray counts in a 100-mr/hr gamma field without any noticeable shift in the proton recoil spectrum above 500 kev in a laboratory test. There is also considerable interest in using an He^3 proportional control in this application with special circuitry to reject elastically scattered He^3 nuclei.

Preliminary measurements were made of the gamma leakage spectra from shield assemblies, using a scintillation spectrometer. High counting rates necessitated the design and fabrication of a collimator with a small aperture to reduce the signal to the detector. No further tests have been made with the system.

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E. SHIELD MATERIAL TECHNOLOGY

1. Lead Gamma Shielding Studies

Plans and preparations had previously been made to conduct liquid lead compatibility tests using quartz thermal harps. Search of the literature revealed that quartz might introduce trace impurities of a third material and thus invalidate some of the tests. This prompted reassessment of the approach, a decision to use small static capsules, and to increase the number and variations of capsules used in the test.

The test has been redesigned. It will use small capsules that will be inexpensive to fabricate. This allows testing of several material combinations with and without corrosion inhibitors at various temperatures and for three different periods of time. The test matrix is shown in Table 3. The materials for the capsule fabrication have been ordered and the engineering drawing of the basic capsule has been completed. The test capsule configuration is shown in Figure 11.

The design study of the problem of providing for the 10% volume expansion of lead when it is heated in a shield from room temperature to the molten state has been completed. A detailed design of an actual system will require a further thermal analysis to solve the problem of high stresses which might occur in the containment vessel caused by uneven heating or cooling of the shield.

Five shield configurations were derived from the results of the design study and are listed here.

1) To allow for an expansion void in the containment vessel would create a void at all temperatures except at the nominal operating temperature. This would be acceptable if the shield temperature did not vary greatly from the operating temperature or, if an artificial gravity field were present to insure a harmless location of the void during the period when the shield is below operating temperature.

2) To compartmentalize the containment vessel and leave an expansion void in each compartment (see Figure 12) would produce voids at all temperatures except at the nominal operating temperature. The remarks for design 1 apply here except where a zero gravity environment exists at temperatures lower than operating. Tendency for the void to become stationary

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TABLE 3
TEST CAPSULE MATERIAL CONFIGURATIONS FOR
LIQUID LEAD COMPATIBILITY TESTS

Containment Vessel Material	Without Inhibitor			With Inhibitor*		
	500 hr	2,500 hr	10,000 hr	500 hr	2,500 hr	10,000 hr
	Lead					
1020 Carbon Steel	X	X	X	X	X	X
Chrome-Moly Steel (Croloy-2-1/4)	X	X	X	X	X	X
Type 446 CRES	X	X	X	X	X	X
Columbium	X	X	X	X	X	X
	Lead-Bismuth Alloy and BeO					
1020 Carbon Steel	X	X	X	X	X	X
Chrome-Moly Steel (Croloy-2-1/4)	X	X	X	X	X	X

*200 to 300 ppm of Zr or Mg.

Each configuration tested at 1000, 1300, and 1500°F. Total number of capsules - 108.

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Figure 11. Test Capsule for Liquid Lead Material Compatibility Tests

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Figure 12. Shield Containment Vessel
Compartmentalized Design Con-
figuration for Expansion
Compensation

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would be minimized, and consequently the probability of gamma streaming would be lessened.

3) To add an expansion chamber having a bellows to the containment vessel would maintain the shield as void-free throughout the temperature range; however, at normal operating temperature this would deposit a large amount of lead (approximately 8% of the total) in the expansion chamber, which would not localize advantageously to be used as a gamma shield. In addition, the criticality of the melting and freezing pattern in the shield should be considered in this configuration.

4) To allow the containment vessel to breathe could be done by using flexible diaphragms and retaining straps having different thermal expansion coefficients than those of the bellows. Straps would insure proper diaphragm location without the use of springs, and thus eliminate any problem arising from spring relaxation at high temperature (see Figure 13). This design, which can be used in either a zero gravity or gravity environment would permit the lead to expand, but would also prevent voids at any temperature.

5) To eliminate the difference in expansion between the containment material and the material within by using lead-bismuth eutectic instead of lead since lead-bismuth has zero expansion during phase change. If material with a high thermal expansion coefficient is selected for the containment vessel, and a material such as beryllium oxide having a very low thermal expansion coefficient is contained with the lead-bismuth in the proper proportion, any voids in the shield will be minuscule for the entire operating range (see Figures 14 and 15). This plan can also be used in either zero gravity or gravity environment, and is more reliable than the one discussed in 4) since a bellows flexure is not required (see Figure 16).

2. Aggregate Gamma Shield Materials

a. Aggregate Tungsten-Alloy Gamma Shield

The aggregate gamma shield, designated Experimental Shield No. 2, was conceived to demonstrate the versatility of the casting process, and is a combination of two novel concepts.

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Figure 13. Shield Containment Vessel Design Configuration Incorporating Flexible Diaphragms and Retaining Straps for Expansion Compensation

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Figure 14. Shield With Balanced Thermal Expansion
Composed of Specified Proportions of
Croloy-2-1/4, BeO, and Bi-Pb

Figure 15. Shield With Balanced Thermal Expansion
Composed of Specified Proportions of
1020 Steel, BeO, and Bi-Pb

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Figure 16. Shield Containment Vessel Design Configuration Incorporating Materials of Opposite Thermal Expansion Coefficients for Expansion Compensation

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1) A layer of granular tungsten-alloy gamma shielding (98% W - 2% Mo) imbedded in the cast LiH neutron shielding

2) Two insulated stainless-steel pipes, simulating NaK pipes, contained within two structural stainless steel pipes traversing the shield from top to bottom around which the LiH is cast.

During the previous quarter, the shield was fabricated and cast, but not examined. The freezing pattern, indicated by thermocouples within the LiH, appeared to digress from normal during the time the LiH was solidifying at the tungsten alloy layer. It was conjectured that a casting void had formed below the tungsten layer. When the shield had cooled, it was removed from the casting retort and examined. Visually, the shield vessel appeared sound, although slight distortions were found along the conical sides. Several encrustations of white material were found at the NaK pipe penetrations which are indicative of a porous weld. Figure 17 shows one of the two penetrations which had leaked a slight amount of LiH.

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Figure 17. NaK Pipe Penetration on Top of Shield Vessel
Showing White Encrustations

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Radiographic examination of the shield revealed that a large casting void (6 to 8 in. in diameter by 6 in. thick) existed directly below the center of the tungsten layer. The remainder of the casting was exceptionally uniform and free of large shrinkage cracks. It was concluded from these examinations that the LiH had solidified in the granular W-alloy layer before solidification was completed in the lower LiH region. This stopped the flow of molten LiH through the granular W-alloy and created a casting void in the lower region.

The possibility of remelting the LiH was deliberated, and it was concluded that the potential gain in valuable information exceeded the risk involved. The remelting operation never had been attempted previously, because melting LiH creates a large volume expansion which might cause rupture of the vessel. In this particular case, it was complicated further by an imbedded layer of granular W-alloy in the LiH.

The two leaky weld areas on the NaK pipe penetrations were cleaned and examined but no definite leak or crack could be found, so the seepage was attributed to a generally porous weld which probably would not worsen during remelting.

A stainless-steel-sheathed, magnesia-insulated Nichrome heater was placed around the shield vessel (see Figure 18) at the region of the W-alloy layer.

Figure 18. Experimental Shield
No. 2 — View of Tungsten
Region Heater

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This was done to furnish additional heat to the W-layer during remelt and retard the heat loss from the W-layer during cooling, and to prevent the premature solidification which had occurred during the initial casting.

The shield and its casting assembly were reinstalled in the retort and the temperature of the entire shield was initially raised to about 1200°F. The temperature at the top of the furnace was then slowly increased to start melting of the LiH in the reservoir, followed by melting in the filler tube, and subsequently down to the bottom of the shield. This technique permitted the melting LiH to expand upward into a liquid region. About 50 hr were required to remelt the LiH.

Initially, freezing of the LiH proceeded in a normal manner during the cooling cycle; then both filler tube heaters became inoperative. Without these heaters to prevent freeze-out of LiH in the filler tube and subsequent casting voids, the normal casting method could not be used. Instead, the furnace height, temperature, and power to the reservoir and W-alloy layer heaters had to be carefully and slowly manipulated and adjusted to control the cooling rate of the LiH so that solidification would occur in the proper sequence (bottom-to-top).

After cooling, the shield was removed from the retort for examination. Visually, the shield vessel did not appear affected by the reprocessing, except for some distortion on the upper head. Distortions on the side of the vessel, noted after the original casting, appeared considerably reduced. Figure 19 shows the shield vessel after the recasting operation.

The two porous weld areas around the NaK pipe penetration, which had permitted some seepage of LiH during the initial casting, did not seem to have worsened, although a new seepage was observed at a previously sound weld (see Figure 20).

Radiographic examination revealed that the casting was basically sound, except for two areas. A slight high-porosity region, perhaps a small void, was detected immediately below the W-alloy layer. At the top of the shield, there appeared to be a small casting void, but due to the viewing angle, closeness to the surface, etc., it was difficult to determine its exact nature. It is possible that a small void formed in that region during the final stages of freezing caused

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Figure 19. Experimental Shield
No. 2 After Recasting -
Overall View in Inverted
Position

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Figure 20. Experimental Shield No. 2 - View
of Leak at No. 1 Pipe Penetration on
Bottom Head

by the inability to provide close temperature control. Final evaluation of the casting results for this region and the region below the tungsten layer has been delayed until a destructive post-mortem examination can be made.

b. Aggregate Zirconium-Hydride-Lithium-Hydride Test Program

In some shield applications, utilizing a ZrH-LiH aggregate has been analytically presented as a source of potential weight savings. Through laboratory scale fabrication studies it is evident that strong, dense aggregates can be made by melting and casting LiH into a matrix of the granular ZrH gamma shield material. This technique provides an attractive method of fabricating reactor shields into sizes and shapes which otherwise would be virtually impossible to obtain. In addition, the process uses low cost, readily available granular ZrH instead of expensive slabs which are sometimes difficult to procure.

During the reporting period, a number of test rods were prepared and sectioned in the laboratory (see Figure 21) to provide specimens for the long-term

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Figure 21. Zirconium Hydride-Lithium
Hydride Aggregate Specimens
From Cast Rod

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(up to 4000 hr) compatibility tests. After completion of the first test (1000 hr at 800°F) on one specimen, it was removed from the capsule for examination. The LiH crystals had changed from blue-gray to a bright purple; however, the ZrH appeared to be unchanged. Chemical analysis of the specimen will be made when other tests are completed.

c. Aggregate Tungsten-Alloy-Lithium-Hydride Test Program

An aggregate tungsten-alloy (98% W-2% Mo) -LiH materials properties test program was initiated as a parallel effort to the ZrH-LiH program. Granular W-alloy was procured and specimen casting was initiated to prepare samples for test and evaluation.

3. Nuclear Irradiation of Lithium Hydride

An analysis is nearing completion on the number of lattice displacements created in LiH by energetic charged particles. The charged particles are:

- 1) Li and H knock-ons from neutron scattering
- 2) A 2.1-Mev α and a 2.7-Mev H^3 from the $L^6(n, \alpha)H^3$ reaction
- 3) Two 1.5-Mev α and a 13-Mev-maximum β from the $Li^7(n, \beta)2\alpha$ reaction.

The pertinent results of the analysis are presented in the following paragraphs.

The mean number of displacements per colliding neutron $\bar{V}(E)$ vs neutron energy for Li^7 and H knock-ons in LiH is shown in Figure 22. The curves tend to have a plateau because most of the energy lost by the energetic charged particles in slowing down is lost by ionization rather than by displacement-producing collisions. A useful approximation is to assume that above some threshold energy I , most energy transfer is by ionization; and below the threshold, all energy transfer is by Rutherford scattering. I is determined by the lowest excitation potential, I , of the valence electrons, and is roughly given in insulators by:

$$I \approx \left(\frac{M_1}{m} \right) \frac{I_t}{8} \quad , \quad \dots (1)$$

where M_1 is the mass of the knock-on and m is the electron mass. I_t corresponds to the edge of the first main optical absorption band which is 2500Å for LiH or about 5.0 ev. The ionization energy for the LiH knock-ons, as well as

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Figure 22. Mean Number of Displacements per Colliding Neutron vs Neutron Energy for Li^7 and H Knock-Ons in Lithium Hydride

for α and H^3 , is given in Table 4. Then for knock-ons with energies above the ionization energy I , the number of displacements will equal approximately:

$$N_d \approx \frac{bI}{E_d}, \quad \dots (2)$$

where E_d is the displacement energy, usually assumed to be 25 ev, and b is a constant between $1/2$ and $1/4$. Some substitution of numbers into Equation 2 would indicate that the results of the more detailed analysis as indicated in Figure 22 are the right order of magnitude.

TABLE 4
IONIZATION ENERGY
FOR LiH

Particle	I(ev)
H^1	1,150
H^3	3,440
He^4	4,590
Li^6	6,900
Li^7	8,000

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In Figure 23 is shown the lattice displacement coefficient, $\epsilon_s(E)$, for neutron scattering in LiH vs neutron energy which was calculated from:

$$\epsilon_s(E) = \left[NV(E)\sigma_{tr}(E) \right]_H + \left[NV(E)\sigma_{tr}(E) \right]_{Li^6} + \left[NV(E)\sigma_{tr}(E) \right]_{Li^7} , \quad \dots(3)$$

where N is the atom density of H, Li^6 , or Li^7 . $\epsilon_s(E)$ was calculated to be used with 16-group neutron cross sections. The calculated values are for the mean energy of the i-th energy group in which the group flux averaged transport cross sections were used.

ϵ_{Li^6} is the lattice displacement coefficient for the $Li^6(n,\alpha)H^3$ reaction and equals:

$$\epsilon_{Li^6} = (N\nu\sigma_c)_{Li^6} , \quad \dots(4)$$

where N_{Li^6} is the atom density of Li^6 , ν_{Li^6} is the number of displacements per Li^6 capture, and σ_{cLi^6} is the capture cross section for Li^6 for the i-th energy group. ν_{Li^6} was calculated to be 195 displacements per Li^6 capture.

Figure 23. Lattice Displacement
Coefficient vs Neutron Energy
for Scattering in Lithium
Hydride

Similarly, the lattice displacement coefficient $\epsilon_{\text{Li}7}$ for the $\text{Li}^7(n, \beta)2\alpha$ -reaction, equals:

$$\epsilon_{\text{Li}7} = (N\nu\sigma_c)_{\text{Li}7} \quad .$$

$\nu_{\text{Li}7}$ is calculated to be 121 displacements per Li^7 capture, of which an average of 12 displacements are due to the 13-Mev-maximum β decay.

The number of displacements per cm^3 is given by:

$$N_d = \int_0^{\infty} \epsilon_d(E) \Phi(E) dE \quad , \quad \dots (5)$$

where $\Phi(E)$ is the neutron dose (nvt) as a function of neutron energy. $\epsilon_d(E)$ is the probability per unit length that a lattice displacement will be created by neutrons of energy E about dE . $\epsilon_d(E)$ is made up of three parts,

$$\epsilon_d = \epsilon_s + \epsilon_{\text{Li}6} + \epsilon_{\text{Li}7} \quad .$$

For i energy groups,

$$N_d = \sum_i \frac{\int_{E_i}^{E_{i+1}} \epsilon_d(E) \Phi(E) dE}{\int_{E_i}^{E_{i+1}} \Phi(E) dE} \int_{E_i}^{E_{i+1}} \Phi(E) dE$$

$$\approx \sum_i \left[\epsilon_s(\bar{E}_i) + \epsilon_{\text{Li}6} + \epsilon_{\text{Li}7} \right] \Phi_i \quad , \quad \dots (6)$$

where Φ_i is the integral dose for the i -th group. ϵ_s is assumed to be approximately constant over the group energy interval and evaluated at the mean energy

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$\bar{E}_i$. Tables 5 and 6 give the various lattice displacement coefficients that have been evaluated for natural LiH and 0.9999 Li⁷H.

Calculations have been made of the number of lattice displacements in natural LiH and 0.9999 Li⁷H shields using fluxes obtained from DTF calculations made for the S8DS shield. The spectrum used was 8 cm into the shield where an approximate equilibrium spectrum occurs. The percent of lattice displacements from the important sources of displacements is summarized in Table 7. The conclusion is that both fast and thermal neutrons can create damage in LiH and that careful control and measurement of the neutron spectrum must be made for LiH irradiations.

TABLE 5
LATTICE DISPLACEMENT COEFFICIENTS FOR
NATURAL LiH

Group i	Lower Energy, E_i	ϵ_{s_i} (cm ⁻¹)	$\epsilon_{Li^6_i}$ (cm ⁻¹)	ϵ_{d_i} (cm ⁻¹)
1	10 Mev			
	3 Mev	13.96	0.05	14.01
2	1.4 Mev	10.74	0.10	10.84
3	0.9 Mev	11.02	0.12	11.14
4	0.4 Mev	10.17	0.22	10.39
5	0.1 Mev	24.27	0.87	25.14
6	17 kev	9.13	0.38	9.51
7	3 kev	6.02	0.63	6.65
8	0.55 kev	3.03	1.93	4.96
9	100 ev	0	4.35	4.35
10	30 ev	0	8.56	8.56
11	10 ev	0	15.5	15.5
12	3 ev	0	26.9	26.9
13	1 ev	0	48.9	48.9
14	0.4 ev	0	76.7	76.7
15	0.1 ev	0	138.6	138.6
16	0.001 ev	0	375.0	375.0

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TABLE 6
LATTICE DISPLACEMENT COEFFICIENTS FOR 0.9999 Li⁷H

Group i	Lower Energy, E _i	ϵ_{s_i} (cm ⁻¹)	$\epsilon_{Li^6_i}$ (cm ⁻¹)	$\epsilon_{Li^7_i}$ (cm ⁻¹)	ϵ_{d_i} (cm ⁻¹)
1	10 Mev				
	3 Mev	13.99	0	0	13.99
2	1.4 Mev	10.90	0	0	10.90
3	0.9 Mev	11.16	0	0	11.16
4	0.4 Mev	9.95	0	0	9.95
5	0.1 Mev	23.83	0	0	23.83
6	17 kev	8.87	0	0	8.87
7	3 kev	5.73	0.001	0	5.73
8	0.55 kev	2.88	0.003	0.001	2.88
9	100 ev	0	0.006	0.002	0.008
10	30 ev	0	0.012	0.005	0.017
11	10 ev	0	0.021	0.008	0.029
12	3 ev	0	0.036	0.014	0.050
13	1 ev	0	0.066	0.025	0.091
14	0.4 ev	0	0.103	0.041	0.144
15	0.1 ev	0	0.186	0.078	0.264
16	0.001 ev	0	0.505	0.215	0.720

TABLE 7
PERCENT OF LATTICE DISPLACEMENTS IN NATURAL LiH AND
0.9999 Li⁷H

	LiH	Li ⁷ H
Neutron Scatter	42	68
Li ⁶ (n, α)H ³	58	25
Li ⁷ (n, β)2	0	7

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4. Lithium-Lithium Hydride Bonding

a. Lithium Hydriding Tests

The Li metal-backfilled LiH shield concept⁽¹¹⁾ presents several potential problems, one of which is the presence of the H₂ atmosphere within the shield produced by the dissociation of LiH. A test program was initiated to study the rate of hydriding of metallic Li at temperatures of 200 to 1400°F.

The rate of weight gain of specimens of metallic Li in a hydrogen atmosphere has been measured using an Ainsworth Automatic Recording Balance at 200, 400, 600, 800, 1000, 1200, and 1400°F. At 200°F the rate of weight gain was too low to be measured with the balance system during a 100-hr test, and at 1400°F the rate of weight gain was too rapid to be measured accurately. Test data at the other temperatures indicate that the rate of hydriding is diffusion-controlled; that is, the following parabolic equation is obeyed:

$$w = kt^{1/2} \quad , \quad \dots(7)$$

where w is the amount of H₂ reacted per square centimeter of surface, k is the reaction rate constant, and t is the time in seconds.

The variation of the reaction rate constant with temperature is expressed by:

$$k = Ae^{-Q/RT} \quad , \quad \dots(8)$$

where A is the frequency factor, Q is the activation energy of formation, and T is the temperature in °K. Preliminary calculations have produced a Q value of 13.3 kcal/mole of H₂ which agrees quite well with a reported value of 12.0 kcal/mole.

Several tests will be repeated to verify the value of the reaction rate constant at 400 and 600°F.

b. Lithium-Lithium Hydride Corrosion Tests

A second problem associated with the lithium metal-backfilled shield is the corrosion of the austenitic stainless-steel vessel by the Li-LiH system at shield operating temperatures. A group of three capsules which had been

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exposed to Li-LiH (about 50/50 ratio by volume) for 4000 hr at 600, 900, and 1200°F were examined metallographically.

No significant evidence of corrosion was found for the 600 and 900°F stainless-steel specimens; however, some spotting and shallow (less than 0.001 in.) grain boundary penetration was observed in the 1200°F specimen. No difference in compatibility was observed between the base metal and the welded joint. These results indicate that mixtures of Li and LiH perhaps more accurately described as H-deficient LiH systems, do not corrode austenitic stainless steel to any greater extent than was previously found with normal hydrogen high-purity LiH systems.

c. Lithium-Lithium Hydride Compatibility Study

A study was initiated to learn the behavior of molten Li metal heated in contact with LiH for periods up to 4000 hr simulating the conditions to which a Li metal-backfilled shield might be exposed.

After 1000-hr exposure in sealed, argon-filled capsules, the following observations were made of a cylinder of cast LiH with a central cavity containing Li metal:

1) At 800°F, the Li and LiH appeared unchanged.

2) At 1000°F, the lower half of the LiH cylinder appeared to have tiny droplets of Li metal in the surface, as if the molten Li had penetrated the LiH cylinder. Some Li remained in the central cavity.

5. Casting Facility Liaison and Checkout

Planning was continued on the Building 80 Casting Facility which will eventually provide much larger shield fabrication accommodations (approximately twice the diameter and height of the Building 3 casting equipment facility which limits the maximum size to the SNAP 8 instrument-rated shield design). Requests for Proposal (RFP) were prepared to invite vendors to supply the basic process requirements on a minimum cost basis within the established funding limit. The RFP's had not yet been submitted to vendors for consideration at this date.

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Figure 24. Hydrogen Permeation Rate vs Pressure

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Figure 25. Hydrogen Permeation Rate vs Concentration

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F. HEAT AND MASS TRANSFER

1. Hydrogen Partial Pressure Experiments

a. Summary

Hydrogen permeation characteristics through Type 321 stainless steel in a NaK environment have been studied as a function of hydrogen concentration and hydrogen dissociation pressure in dilute solutions of hydrogen in NaK at temperatures of 1320, 1260, 1175, and 1075°F. The three higher-temperature NaK capsules, as reported in the Progress Report for the previous period, yielded consistent data showing linear relationships among all three parameters (permeation rate, concentration, and partial pressure) in the low hydrogen concentration range under study. The 1075°F NaK capsule showed rather anomalous behavior which has yet to be satisfactorily explained. Prior to removal of the 1075°F capsule, an experiment was conducted to determine how fast equilibrium would be reached if a finite supply of hydrogen was admitted to the nickel tube (McLeod gauge) and the pressure drop was monitored in conjunction with the pressure rise outside the stainless-steel NaK chamber. The unexpected result was that about 72 hr were required to reach equilibrium conditions. The spare 1320°F NaK capsule has been loaded into the test port and will be run initially at 1075°F to verify the previous capsule's behavior. Subsequently, the temperature will be raised to 1320°F to complete the test program.

b. Progress During Reporting Period

Testing of the 1075°F capsule was completed. As with the 1175°F capsule, the diffusion kinetics were rather slow causing a slippage of the original schedule.

The results of the 1075°F capsule were in accord with the previous capsules reported in the previous quarter in every respect except one — the temperature dependence of hydrogen permeation to partial pressure. The data, although still linear, fall in the range of the 1260°F capsule in this one parameter only. Both concentration vs permeation rate and concentration vs partial pressure agree with previous data. Figures 24 through 26 show the 1075°F data in reference to previously reported data at the higher test temperatures.

The design of the test capsules used in this experiment lends itself readily to the study of the solubility of hydrogen in NaK. Literature values on the kinetics

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Figure 26. Hydrogen Concentration vs
Dissociation Pressure

for reaching equilibrium pressure-concentration conditions in the NaK-H₂ system are on the order of ~15 min. Since the AI system has the capability of monitoring the partial pressure of hydrogen on both sides of the NaK, a brief test was performed to check the validity of the literature values. A pressure of 10 mm Hg was admitted (static) to the vacuum chamber surrounding the NaK capsule. As the hydrogen went into solution in the NaK, this pressure dropped until equilibrium was reached. At the same time, the pressure inside the nickel tube (in the NaK) gradually increased. A small leak in the high-pressure hydrogen inlet tube invalidated the result; but when the leak was isolated and the hydrogen was injected in the reverse direction (i. e., 10 mm Hg in nickel tube and static vacuum in retort housing capsule), the results were as expected (see Figure 27). The time to reach equilibrium was, however, much longer than that reported in the literature, 72 hr vs 15 min. A large portion of this discrepancy can undoubtedly be explained by the significant difference in surface-to-volume ratio between the two tests. The present test emphasizes the importance of the time required to reach equilibrium.

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Figure 27. Hydrogen Pressure
Change vs Time

2. Mercury Supersaturation Study

a. Nozzle Expansion System

The nozzle expansion system was completed and instrumentation checkout was in progress at the end of the reporting period. In approximate chronology the system completion activities were:

- 1) Installation of pressure instrumentation and vacuum reference system (see Figure 28)
- 2) Completion of transducer cabling from the enclosure to the instrumentation racks
- 3) Wiring and control installation for the dryer superheater and for heater cable wrapped around the piping between the nozzle and boiler (see Figure 29).
- 4) Installation of thermocouples on nozzle walk and system piping
- 5) Insulation of the nozzle and nozzle entrance pipe (see Figure 30)
- 6) Performance of gas calibration runs with a noncondensable gas (argon) to evaluate nozzle performance

Figure 28. View of Nozzle System Prior to Installation of Insulation

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Figure 29. View of Boiler Outlet Portion of Nozzle System Prior to Installation of Insulation

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Figure 30. Nozzle Section With
Insulation Installed

- 7) Installation of insulation on boiler and piping upstream of nozzle
- 8) Wiring of system thermocouples to instrument rack.

The argon gas data (see Figure 31) indicate a Mach number of 2.08 at the last pressure tap in the nozzle. The predicted maximum supersaturations resulting from noncondensing mercury vapor expansions to this Mach number are on the order of 10^5 for the range of pressures to be used (see Figure 32).

b. Diffusion Cloud Chamber

The diffusion cloud chamber, which was built and manned at the NAA Science Center in technical support of this project, was successfully operated. Chamber condition data were recorded for the condensation of methanol, water, and ethanol. From these data a computer program, which allows for mass diffusion and thermal conductivity in the chamber, was used to determine the variation of supersaturation with temperature for each chamber condition (defined by a top and bottom temperature and a chamber total pressure). An experimental curve of critical supersaturation vs temperature was then drawn by fairing in the line

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Figure 31. Comparison of Argon Nozzle Flow Data With Isentropic Flow Parameters

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Figure 32. Mercury Vapor Conditions at Last Pressure Tap Based on Argon Expansion Data

most nearly tangent to the curves for individual chamber states (see Figures 33, 34, and 35). Comparison of experiment and theory indicates good agreement, since values are within 10%.

Improvements in operation were made as a result of experience. A system providing a time history of individual thermocouple readings was installed. Insertion of a metering valve in the helium pressure line to the freon system allows chamber conditions to be more closely controlled. A small airleak was discovered in the main chamber; this leak allowed air to dilute the lighter hydrogen carrier gas and create convection instabilities. This leak is believed to be responsible for two inaccurate water data points and an inaccurate ethanol point. Standard procedure now is to evacuate the chamber and replace the hydrogen carrier gas after each run.

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Figure 33. Methanol — Theoretical and Experimental Variation of Supersaturation With Temperature

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Figure 34. Ethanol - Theoretical and Experimental Variation of Supersaturation With Temperature

Figure 35. Water - Theoretical and Experimental Variation of Supersaturation With Temperature

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IV. EVALUATION OF EFFORT

Spectral temperature coefficient determinations in SNAP systems have been improved through the use of newly acquired Zr-H_x cross sections in the new DTF/ANISN library. Numerical difficulties in 100-GT affecting flux calculations have been corrected.

The GRISM code was made operational; however, optimum operation will be achieved only after certain desirable modifications have been completed.

Theoretical predictions of gamma-ray transmission through lead slabs have shown reasonable agreement with experimental measurements. Secondary gammas from neutron capture and inelastic scattering have been estimated for the slab thicknesses used thus far; however, comparison with measured results for thicker slabs would provide a better check on the estimating method.

With completion of the lead-LiH measurements now in progress, AI will have a rather sufficient description of neutron and gamma attenuation in the basic shield materials and, in combinations of these shields to form a single cycle of a laminated shield. Comparison with calculations has disclosed that the analytical techniques are seriously inadequate for predicting the source distribution of secondary gamma rays. Further measurements may be necessary so that empirical source functions may be substituted for analytically determined functions.

The decision to use small static capsules for the liquid lead materials compatibility tests instead of the quartz thermal harps is expected to produce more highly desirable results. The use of the capsules will allow testing of several material combinations with and without corrosion inhibitors at various temperatures, and for three different periods of time.

The initial casting of the aggregate tungsten-alloy gamma shield, designated Experimental Shield No. 2, was judged as unsatisfactory due to the presence of a large casting void which would be intolerable in a reactor shielding application. A remelting operation, which had never previously been attempted because of the potential danger of rupturing a shield vessel, was successfully completed. The resolidification of the LiH was accomplished satisfactorily, even though two critical filler tube heaters failed. Radiographic examination revealed the presence of two small, low-density regions which may be casting voids, although

their exact nature cannot be determined until a final destructive examination is conducted. The size of the low-density areas would class the shield as satisfactory-to-marginal. In general, the remelt and recast operations are regarded as highly successful, since successful performance in these operations has now been demonstrated for the first time.

In the aggregate ZrH-LiH test program, rod specimens have been cast in the laboratory and the initial ZrH-LiH compatibility test (1000 hr at 800°F) was completed. No significant changes were observed, although chemical analysis had not yet been performed.

The S_n transport code, ANISN, is being checked out with the lattice displacement coefficients used as dummy cross sections so that the number of displacements caused by irradiating LiH may be calculated by the activity option. This will facilitate calculating displacements vs distance through a shield and thus account for the large changes in neutron spectrum which occur in the first several centimeters of a shield.

These calculations not only give an indication of the effect of neutron spectrum, but also give absolute numbers of lattice displacements for given neutron doses. Work is in progress to relate the total number of displacements to the number of atoms in LiH. It is hoped that some conclusions may be made in the next reporting period.

The Li hydriding test results (Li-LiH bonding studies) to determine the hydriding rate constant and its temperature dependence appear fairly reliable, although several low-temperature (low-rate) tests must be repeated.

The results of metallographic examination, following the Li-LiH corrosion tests, reveal that the Li-LiH system at temperatures up to 1200°F for 4000 hr does not corrode austenitic stainless steel to any greater extent (0.001 in. or less) than high-purity LiH.

The strange behavior of the 1075°F NaK capsule in the hydrogen partial pressure experiments does not alter those relationships derived from the previous capsules in any respect. Further examination of this discrepancy will be withheld pending further experimental effort which is presently in progress. As reported in the previous quarter, it appears that an extrapolation of hydrogen permeation data on many different reactor materials can be made which will

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allow closer predictions of hydrogen concentrations in complex reactor systems. This extrapolation involves a square root of pressure dependence from 1 atm to about 10 mm Hg and a linear pressure dependence below that.

The nozzle expansion system for the mercury supersaturation study project was completed about 2 weeks behind schedule. The amount of time required for insulation of piping and connection of thermocouple leads was underestimated. The system, however, is now ready to begin operations with mercury vapor.

The diffusion cloud chamber operation was generally satisfactory. The amount of data obtained was not as large as intended, but sufficient data were gathered to indicate reasonable correlation between theory and experiment.

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V. OBJECTIVES FOR THE NEXT REPORTING PERIOD

The objectives for the next quarter are as follows:

1) Nuclear and Control Analysis:

a) Publish the translation of Rotter's article on photoneutron constants⁽¹²⁾ as an internal document. Continue work on the analysis of SNAP experimental data regarding photoneutron production in Be reflectors.

b) Continue the evaluation of quadrature coefficients for use with S_n transport theory codes.

c) Publish an internal report on previous evaluations of the Be (n,2n) and Be (n, α) cross sections.

d) Assist in the development of a code to make the Bondarenko⁽⁴⁾ library available with the DTF-II code. (This tool is to be used to evaluate the significance of beryllium anisotropic scattering on k_{eff} and to estimate temperature coefficients.)

e) Develop a code to compute spatial neutron flux kinetics by flux synthesis techniques. Results of the computations performed by this method will be compared with detailed time-dependent S_n calculations to evaluate accuracy and relative computer time requirements.

f) Modify the existing UPSTART code as required by SNAP system changes, particularly in the area of transient fuel rod hydrogen diffusion effects.

2) Shielding Analysis:

a) Develop models to describe secondary gamma emission due to neutron capture for introduction to codes used in shielding studies, such as DTF-II.

b) Assist ORNL in the SNAP-TSF experimental planning and analysis of SNAP shield configurations for fast-neutron leakage.

c) Complete development of the GRAMS code, and update neutron cross-section libraries for shielding code usage.

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3) Shield Experiments:

a) Complete measurements on lead-LiH shield slabs.

b) Continue measurements on the sources of secondary gamma rays in both uranium and lead shields, using foil detectors. A scintillation spectrometer will be utilized to determine the gamma ray spectrum changes through the heavy metal shields.

c) Initiate measurements on two-cycle laminated shields.

4) Shield Material Technology:

a) Report partial results of the 500-hr test. Depending on the results of the 500-hr test, it may be possible to eliminate the long-term tests of some material combinations.

b) Begin internal report summarizing the results of the design study and material tests.

c) Perform an extensive destructive examination of the cast shield (aggregate tungsten-alloy gamma shield) to obtain more information on its internal characteristics.

d) Complete the 2000-hr aggregate ZrH-LiH compatibility tests. In addition, cast and test specimens for density and modulus of rupture values.

e) Prepare aggregate tungsten alloy-LiH specimens for long-term compatibility tests and for testing of density and modulus of rupture.

f) Continue work on calculations of lattice displacement created in LiH by energetic charged particles. Complete technical requirements for the irradiation and begin contract negotiations for a reactor to perform the irradiation.

g) Repeat several low-temperature Li hydriding tests to verify the data associated with the Li-LiH bonding project.

h) Incorporate the information gained from the metallographic examination of the corrosion of austenitic stainless steel by a mixture of Li and LiH in an overall internal Li-LiH Compatibility and Corrosion Study report.

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i) Open and examine the 2000-hr Li-LiH compatibility test capsule.

j) Submit the Requests for Proposal on the Building 80 Casting Facility to interested vendors for consideration.

5) Heat and Mass Transfer:

a) Due to project direction, no additional work on the NaK-H₂ partial pressure experiment will be performed in FY 1967. In the next reporting period, therefore, the second 1075°F capsule will be completed, the 1320°F experiment recheck will be performed, and a topical report will be written. This topical report will include all activities conducted in FY 1966 and is scheduled for publishing in August of FY 1967.

b) With reference to the continuous flow nozzle expansion loop for the mercury supersaturation study, the following are the objectives for the next reporting period:

(1) Perform checkout and operation of the system with mercury vapor.

(2) Record pressure and temperature data for the expansion of mercury vapor from pressures of 30, 40, 50, and 60 psia.

(3) Analyze data to determine the degree of supersaturation achieved.

c) The objectives of the cloud diffusion chamber effort during the next reporting period are:

(1) Record additional data to extend the temperature range of critical supersaturation determination.

(2) Modify the cloud chamber for work with mercury vapor.

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