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PROGRESS REPORT FOR SNAP
GENERAL SUPPORTING TECHNOLOGY
NOVEMBER 1966 - JANUARY 1967

AEC Research and Development Report

ATOMICS INTERNATIONAL

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By
L. S. MIMS
Project Manager

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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. 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 materials studies of phenomena associated with the use of metals, metal hydrides, and liquid-metal coolants and working fluids at high temperatures.

II. SUMMARY OF EFFORT

An improved set of photoneutron decay constants is required in order to predict photoneutron population levels for use in startup studies. A study was therefore made to determine the beryllium photoneutron decay constant sets which best fit existing SNAP experimental data. The study showed that the Rotter sets excel for all available SNAP data. The Rotter "reflector" set is most suitable where only extremely short-lived decay is present.

The efforts on transport-theory flux synthesis were completed for FY 1967.

The LOAFER code was documented.

NaK-78 corrosion testing of 24 ceramic-braze-metal samples was completed.

The ANISN code was used to calculate the neutron levels and secondary gamma source strengths within each STIR heavy metal/lithium hydride shield configuration. Good agreement was shown by preliminary comparisons of ANISN calculations of the thermal neutron flux in LiH with the shape of recent STIR foil measurements.

The QAD code, used to calculate primary gamma dose rate, shows good agreement with STIR measurements.

The objectives of the Composite Material Shielding program have been met and an extensive library has been accumulated of standard configuration measurements for the evaluation of analytical methods and cross-sections, for space reactor system shields.

Chemical analysis was completed of the four samples of LiH irradiated in the STIR, and documentation is being prepared.

The effort of the S10FS-3 post-test examination task was directed toward the post-irradiation examination of selected reactor components, including internal and external reflectors, core hold-down springs, handlebars, and the reflector band. The evaluation was completed and will be documented.

The activity of the cloud chamber supersaturation study consisted of documenting previous investigations of cloud chamber operation and in preparing for studies of fluids calculated to have critical nuclei containing less than 50 molecules.

III. AREAS OF EFFORT

A. REACTOR PHYSICS ANALYSES

1. Comparison of Beryllium Photoneutron Decay Sets Using Experimental SNAP Reactor Data

The portion of photoneutrons which return to a subcritical SNAP reactor core from its beryllium reflector constitute its "unmultiplied" photoneutron source strength, P_o . Once they reach the core, these "first generation" photoneutrons are multiplied in successive generations by the reproduction factor k . In the absence of other neutron sources, the neutron population in a subcritical reactor converges then to,

$$P_o \left(1 + \sum_{n=1}^{\infty} k^n \right) = \frac{P_o}{1 - k} \quad \dots (1)$$

The magnitude of P_o will depend on the time and power level of previous reactor operation (fission product buildup and magnitude) and on the time since shutdown (fission product decay). Knowing the unmultiplied photoneutron source strength and an associated set of suitable decay constants and yields, it is possible to predict a given reactor's neutron population for all operating histories, time of shutdown, and degrees of subcriticality of interest.

Sets of photoneutron decay constants are extracted by analysis of careful experiments. Each set is therefore representative of a particular reactor. Keepin has shown^{(1)*} his set to better fit available data than do sets of earlier workers. No comparison was found for two later sets of Rotter.⁽²⁾

The differences in the Rotter and Keepin sets (Table 1) are listed below.

- 1) Rotter uses twice as many decay groups as Keepin
- 2) Rotter accounts for precursor photoneutron-production in appropriate decay groups

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

TABLE 1
BERYLLIUM PHOTONEUTRON DECAY SETS

Keepin Sets					
Group Index	Gamma Emitters	Half-Life	λ (hr ⁻¹)	Normalized Group Fraction	
1	Ba-140	12.8 days	2.246×10^{-3}	3.80×10^{-3}	
2	Te-132	77.7 hr	8.928×10^{-3}	2.50×10^{-3}	
3	Te-131 + others	12.1 hr	5.724×10^{-2}	1.71×10^{-2}	
4	I-135, La-142, Kr-87, Kr-88	3.11 hr	2.232×10^{-1}	2.108×10^{-1}	
5	Te-134, I-134, Br-84, Cs-138	43.2 min	9.612×10^{-1}	2.37×10^{-2}	
6	Xe-138, Rb-89, Mo-101	15.5 min	2.6712×10^0	2.425×10^{-1}	
7	Sb-133, Kr-89, Rb-90	3.2 min	1.296×10^1	1.219×10^{-1}	
8	I-136, Br-87	1.3 min	3.186×10^1	2.412×10^{-1}	
9	Kr-90, Se-87	0.51 min	8.136×10^1	1.364×10^{-1}	
Rotter Sets					
Group Index	Gamma Emitters	Half-Life	λ (hr ⁻¹)	Normalized Group Fraction in Br 02 Reactor	
				In Core	In Reflector
1	Rh-106	1 yr	7.993×10^{-5}	0.7292×10^{-4}	0.6875×10^{-4}
2	Pr-144	285 days	1.013×10^{-4}	0.2356×10^{-4}	1.968×10^{-4}
3	La-140	12.8 days	1.723×10^{-2}	-1.234×10^{-4}	-1.114×10^{-4}
4	La-140 Precursor	40.2 hr	2.256×10^{-3}	9.423×10^{-4}	8.487×10^{-4}
5	I-132	77.7 hr	8.918×10^{-3}	11.89×10^{-4}	9.957×10^{-4}
6	Zr-97, Y-93	17.0 hr	4.076×10^{-2}	9.479×10^{-4}	8.037×10^{-4}
7	I-135, Y-93	6.68 hr	1.040×10^{-1}	1.649×10^{-2}	10.27×10^{-3}
8	Y-92	3.6 hr	1.93×10^{-1}	0.2008×10^{-2}	1.546×10^{-3}
9	Kr-88, Rb-88, Y-92, Mn-55	2.77 hr	2.50×10^{-1}	1.784×10^{-2}	1.510×10^{-2}
10	Kr-87, La-142	77.0 min	5.40×10^{-1}	3.579×10^{-2}	3.139×10^{-2}
11	I-134	52.0 min	9.45×10^{-1}	-1.728×10^{-2}	-1.091×10^{-2}
12	I-134 Precursor	44.0 min	8.00×10^{-1}	2.131×10^{-2}	1.351×10^{-2}
13	Br-84, Cs-138	32.0 min	1.299	1.352×10^{-2}	1.178×10^{-2}
14	Xe-138, Y-94, Rb-89 Mo-101, Rb-91, Rb-88	15.0 min	2.772	5.301×10^{-2}	4.706×10^{-2}
15	Sb-133, Ru-107, Kr-89	4.1 min	1.0142×10^1	5.699×10^{-2}	5.033×10^{-2}
16	Rb-90, Br-84, Al-28	2.3 min	1.8078×10^1	9.704×10^{-2}	21.48×10^{-2}
17	I-136	86.0 sec	2.9008×10^1	7.404×10^{-2}	6.638×10^{-2}
18	Br-87	55.6 sec	4.4872×10^1	1.122×10^{-1}	1.098×10^{-1}
19	Kr-90	33.0 sec	7.5957×10^1	2.855×10^{-1}	2.458×10^{-1}
20	Br-88, Te-100, Y-94	16.0 sec	1.5594×10^2	2.283×10^{-1}	1.904×10^{-1}

3) The Rotter "core" set is based on photoneutron measurements in the BR 02 core; the Rotter "reflector" set is based on photoneutron measurements in the reflector region. The Keepin set reflects photoneutron measurements of reactor-irradiated samples in a shielded neutron detector.

The first two of these differences are independent of the particular experiment.

The suitability of each set was measured by correcting all data to the same point in time: at shutdown from infinite operation at one thermal kilowatt. The corrected data which exhibited the least deviation from the mean value of P_o should then indicate the set best able to fit the SNAP experimental data. This deviation may be expressed as the variance:

$$\left[\frac{\sum_{i=1}^N \left(\frac{P_o^i}{P_{o\text{avg}}} \right) - N}{N - 1} \right]^{1/2} \quad \dots (2)$$

Table 2 presents the results. Both Rotter sets show clear advantages over the Keepin set. There are insignificant differences between the two Rotter sets for most of the data. The Rotter "reflector" set appears to be the most suitable for the extremely short-lived SNAP 10A decay data.

2. Transport Theory Synthesis: State-of-the-Art Review and Recommendations

FY 67 efforts on transport-theory flux syntheses were completed during the quarter. The planned efforts⁽³⁾ had been to: (1) begin development of a space-time transport-theory flux synthesis code, and (2) develop multi-dimensional (space-space) transport theory flux synthesis. These efforts would have extended previous transport-theory synthesis work^(4,5,6) using parallel developments being made in diffusion-theory synthesis. However, AI development of TSN, an S_n time-dependent transport theory code,⁽⁷⁾ relieved the immediate need for space-time transport-theory synthesis. Also, postponement of the diffusion theory work eliminated the opportunity of using the parallel developments, and time-dependent transport theory code work has thus been postponed.

TABLE 2
ANALYSIS OF SNAP DATA

Reactor	Prior Operation	Measurement Time From Point of Shutdown (hr)	Number Data Points	Percent Variance			Remarks
				Keepin	Rotter Core	Rotter Reflector	
SNAP 8	1400 hr at constant power	13-14	6	16	16	16	
SNAP 8	1400 hr at constant power	234-235	4	19	14	13	These data particularly test the long-lived decay constants (compare to the previous test)
SNAP 8	120 hr at constant power	4-5	6	62	36	38	These data provide a strong test of the <u>short-lived</u> decay constants
SNAP 8	Combination of above data (plus one point at 26 hr after shutdown from 900 hr operation)		17	50	43	37	
SNAP 10A	300 sec transient	1-20	12	46.3	28.1	21.6	These data provide a strong test of the <u>extremely short-lived</u> decay constants
SNAP 2	600 hr at constant power	12-100	94	22.4	18.5	19.8	

a. Space-Space Synthesis Review

Another approach to space-space synthesis has been presented elsewhere and is reviewed below. In this approach to the transport flux synthesis, the multi-dimensional discrete-angle group-fluxes $\phi_{gi}(x, y, z)$ are expressed as:⁽⁸⁾

$$\phi_{gi}(x, y, z) = \sum_{k=1}^K \phi_{gi}^k(x, y) Z_{gi}^k(z) \quad \dots (3)$$

The $\phi_{gi}^k(x, y)$ are the so-called trial functions. They are in general, but not necessarily, discrete-angle group-fluxes of a lower dimensionality than the ϕ_{gi} 's. They are obtained by preliminary S_n calculations at different locations (z in this example) in the system.

The mixing functions $Z_{gi}^k(z)$ are the functions to be determined by the synthesis procedure, as follows:

The expression (3) is inserted into the S_n form of the transport equation, satisfied by the $\phi_{gi}(x, y, z)$. It is then multiplied by a weight function: $N_{gi}^\ell(x, y)$ and integrated over x and y . Equating this expression to zero gives an ordinary differential equation in the functions $Z_{gi}^k(z)$. Repeating the process with each one of the members of a set of weight functions $W_{gi}^\ell(x, y)$, $\ell = 1, \dots, K$, $g = 1, \dots, G$, $i = 1, \dots, N$, a system of $K \times G \times N$ ordinary differential equations in the $K \times G \times N$ unknown functions $Z_{gi}^k(z)$ is obtained. The solution of this system of equations provides the mixing functions and, by means of (3), the solution to the transport problem.

An experimental Fortran program, TRANSY, has been developed to obtain a synthesized solution to the S_2 or S_4 equations of a monoenergetic $x - y$ problem.⁽⁸⁾ It can handle a 100×100 space mesh, 10×10 material regions and 10 trial functions.

The program was applied to calculate fast fluxes in a reactor cell. The results were compared with those obtained with an S_{10} two-dimensional transport approximation and a PDQ-5 diffusion calculation for the same problem. The S_{10} calculation was taken as a standard for comparison. It was found that the integral properties of the cell, e. g., captures by regions and capture ratios for different regions, were calculated with less than 1% error by TRANSY whereas the diffusion results were off by as much as 14%. Computing times for TRANSY were 1/4 of PDQ's and less than 1/40 of S_{10} 's. The point-by-point fluxes were not so well reproduced by TRANSY, but its accuracy was nevertheless far better than that obtained with the two-dimensional diffusion calculations. It is clear from this result that this idea is well worth pursuing for SNAP reactor analysis. The work could be extended to include multigroup problems and higher S_n approximations.

b. Space-Angle Synthesis

A space-angle synthesis method has been very recently proposed by S. Kaplan, J. A. Davis and M. Natelson.⁽⁹⁾ The basic feature is to assume that the neutron flux angular dependence can be well represented by a combination of a few given angular distribution functions. The angular distribution functions used are not, as in a P_n approximation, standard mathematical functions. Instead, they are any angular distribution which intuition or experience suggests for the particular

problem. The synthesis reduces the calculation of the flux angular dependence to the calculation of a set of space-dependent mixing functions. These functions have the same dimensionality as the neutron flux for the problem.

The method used to determine the mixing functions is variational, but differs considerably in principle and formulation from those used in solving the neutron diffusion equation by the authors.⁽¹⁰⁾ It follows ideas proposed by Selengut⁽¹¹⁾ and applied, among others, by Wachspress and Becker,⁽¹²⁾ in a multinode diffusion synthesis.

The problem is first cast into a "canonical" form by expressing the angular flux as a sum of its even and odd angular parts,

$$\varphi(\Omega) = \phi(\Omega) + \chi(\Omega) \quad \dots (4)$$

$$\varphi(-\Omega) = \phi(\Omega) - \chi(\Omega) \quad \dots (5)$$

and then working with the pair of coupled transport equations for ϕ and χ instead of the usual transport equation for φ . A variational principle is then set up, such that it has the transport equations for ϕ and χ among its minimizing conditions. The variational principle proposed has the added feature that the set of functions admissible as its functional variables can have any finite value at the boundaries of the system, and can have finite discontinuities at a finite number of internal surfaces of the system. The boundary conditions and the continuity of ϕ and χ throughout the system are explicitly imposed as part of the set of minimizing conditions. It is possible to choose a set of angular trial functions in one region and a different set in an adjacent region without worrying about boundary discontinuities. The explicit satisfaction of the boundary conditions is also dispensed with when using this principle. In this way, at each region only those trial functions that are suitable for the region are considered. The size of the problem for each region is then reduced with the corresponding savings in computing time.

This synthesis method has been applied to the solution of monoenergetic Milne problems with isotropic scattering and various amounts of absorption. These are convenient first tests of the technique because the synthesis equations can be solved by hand and exact solutions and solutions by other approximations are available for comparison.

Only two trial functions were used in the test cases reported. The amount of calculations needed is roughly comparable to that in a P1 calculation. The results for magnitudes such as extrapolated end point position, angular distribution of the flux at various depths and space variation of the integrated flux can be considerably better than a P3 calculation or clearly worse than a P1 calculation. It all depends on the choice of trial functions. The potential for obtaining excellent approximations is well demonstrated as well as the need to exercise care in the selection of trial functions.

The technique just outlined is still in the development stage. It has yet to be applied to more realistic problems and some remaining theoretical difficulties have to be ironed out.

c. Summary and Recommendations

The space-space and the space-angle transport flux synthesis methods reviewed above have the potential of producing higher order transport approximations at diffusion calculation costs. Both methods are in the development stage, and FORTRAN codes produced are not yet available.

The space-angle synthesis method appears to be of particular interest and most needed in SNAP reactor analysis. However, it is less developed and it is difficult to ascertain how much effort will be necessary to bring it to completion.

3. Nuclear Code Improvement

a. 360 Conversions

Numerous codes, including LOAFER⁽¹³⁾ and 100 GT, were converted for the IBM 360 computer, and debugged.

b. Perturbation Codes

An investigation was made into perturbation codes which might improve SNAP reactor analysis capability. Currently (FY 67) feasible steps were recommended as enumerated below.

1) ANISN Modification

The present AI capability is limited to the first-order diffusion theory PERT⁽¹⁴⁾ code, which is incompatible with the in-house S_n transport code ANISN.

A logical step considered was making ANISN compatible with PERT: i. e., modifying ANISN to punch output fluxes and cross sections in the PERT input format. This could save much computer execution time in perturbation studies. However, PERT inconveniently requires all changes in the form of cross sections, whereas ANISN permits simple changes in atom densities. Thus the only interesting ANISN first-order perturbation capability would be one which requires the same minimum data manipulation as in regular successive cases. This modification did not appear feasible in FY 67.

2) Evaluation of Programs Outside AI

Two laboratories, LASL and Aktiebolaget Atomenergi (Sweden), are known to have second-order transport theory perturbation codes.

A copy of a Swedish internal document has been obtained.⁽¹⁵⁾ It describes six codes intended for use with adjoint and regular S_n group fluxes obtainable from IBM-7090 calculations. Included are second-order perturbation codes in spherical and $r - z$ cylindrical geometries which analyze effective cross section measurements. This will be very useful in evaluating SNAP prepoison cross sections with available spline-worth experiments, and steps were taken to obtain and put into operation the Swedish second-order perturbation codes. The four other Swedish codes did not appear to offer advantages over the PERT code routines.

3) AI Code for Volume Integration of Temperature-Dependent Cross Section

A code is being developed which weights temperature-dependent cross section sets over axial temperature and neutron importance distributions. It will yield for radial calculations a cross section set which is representative of the core and not of a single axial position and will be a desirable tool for the SNAP reactor analysis capability.

4. Studies of Transport-Theory Variables

a. Quadrature Set Comparisons

The studies of quadrature sets and other transport-theory variables for FY 67 were completed. Quadrature sets for SNAP reactor analysis were previously reported⁽¹⁶⁾ for cylindrical geometry. No one quadrature set appeared to have marked advantages over the others.

The comparisons in planar and spherical geometries are made for four sets. These are the Angular Segmentation Set⁽¹⁷⁾(S_n), the Gauss-Legendre Set⁽¹⁷⁾(P_n), the Double-Legendre Set⁽¹⁷⁾(DP), and the Even-Moment Satisfying, Completely Symmetric Set (EM). The comparisons were made by Lee⁽¹⁷⁾ and by Lathrop and Carlson.⁽¹⁸⁾

Figure 1 compares the ability of EM, S_n, P_n, and DP sets to calculate the critical dimensions of unreflected SNAP fuel in planar geometry. The exact values are those determined by Carlson and Bell. Clearly the DP set excels.

Figure 2 presents the same comparison in spherical geometry. For high precision (high n order), the S_n set excels. For low n order the S_n sets are equivalent in accuracy to other quadrature sets.

b. Necessary Significant Figures in Quadrature Coefficients

Investigation by ANISN calculation showed that the difference between four and five significant figures in S8 weights (~ 0.000005 in 0.050000) produced an 0.00031 error in k . This means that, in this example, a one percent error in quadrature weight causes a three percent error in k . Further refinement in weights and cosines produced no additional change in the fifth decimal place of k ,

Number Significant Figures		
In Weights	In Cosines	k
4	5	1.10827
5	5	1.10796
8	8	1.10796

This suggests that: (1) a bare minimum of 5 significant figures be used to obtain 5 decimal place accuracy in calculations, and (2) the percentage error in k is of the same magnitude as that in the quadrature coefficients.

5. Cross Section Updating

Up-to-date temperature-dependent cross-section sets are being prepared for the new 26-group (13 thermal group) library (Table 3). The purpose of the 26-group library is to replace the out-dated error-containing 10-thermal groups of an older 23-group library. The cross sections contributed under GST include the following:

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Figure 1. Critical Half-Thickness of a Bare Slab of SNAP Fuel

Figure 2. Critical Radius of a Bare Sphere of SNAP Fuel

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TABLE 3
GROUP STRUCTURE OF THE NEW 26-GROUP LIBRARY

Group Number	Lower Energy (ev)	Δu	Group Number	Lower Energy (ev)	Δu
1	3×10^6	1.204	14	0.675	0.39
2	1.4×10^6	0.762	15	0.55	0.207
3	0.9×10^6	0.442	16	0.405	0.305
4	0.4×10^6	0.811	17	0.27	1.404
5	0.1×10^6	1.386	18	0.2075	0.265
6	17×10^3	1.772	19	0.735	0.430
7	3×10^3	1.735	20	0.1055	0.246
8	0.55×10^3	1.696	21	0.0775	0.308
9	100	1.705	22	0.0475	0.489
10	30	1.204	23	0.024	0.685
11	10	1.099	24	0.12	0.691
12	3	1.204	25	0.004	1.100
13	1	1.099	26	0.001	1.386

Sm¹⁴⁹ absorption only

Xe¹³⁵ absorption only

Gd^{155, 157} absorption only

Zr^{90, 91, 92, 94, 96} absorption and scattering

H₂O and Zr bound hydrogen differential elastic scattering

U²³⁸ temperature-dependent absorption

These cross sections include Doppler broadening for the zirconium and U²³⁸ isotopes. Table 4 shows the contribution of this effect to a SNAP core temperature coefficient.

Cross sections developed under related SNAP projects include temperature-dependent zirconium-bound hydrogen (previously described)⁽²¹⁾ and U²³⁵, U²³⁸,

B, Zr, O, Fe, Mo, Be, Na, C, Ni, Al, Cr, K, Nb, and 1/V absorber. Documentation of this library and the data sources has begun.

TABLE 4
MEAN CORE TEMPERATURE COEFFICIENT
OVER RANGE 73 TO 1292°F⁽²⁰⁾

Spectral	-0.099¢/°F
Doppler (Zr, U ²³⁸)	-0.034
Expansion	-0.033
Total Prompt	0.166
SNAPTRAN Experiments	-0.163

6. LOAFER Documentation

The LOAFER Code⁽¹³⁾ documentation was completed.

B. DC Conduction Pump Development

Studies⁽²¹⁾ for static nuclear space power systems utilizing liquid metals to transfer reactor or isotopic heat to the thermoelectric converter indicates the DC conduction pump to be the most economical means of pumping the liquid metal in systems up to the 20 kwe level. Significant reduction in the weight of these systems can be effected by relatively small increases in the efficiency of the pump.

A development program initiated in FY '66 under SNAP 10A, is presently being conducted to increase the performance of DC conduction pumps by the methods enumerated below.

1) Employ best permanent magnet materials — A direct decrease in total system weight is realized by the correct choice of permanent magnet material for the pump. The high energy product materials offer the maximum energy per unit volume of material. However, other factors such as

mechanical and temperature stability must also be considered in conjunction with the minimum weight aspect.

2) Reduce "end effect" magnetohydrodynamic flow losses — A significant pressure loss occurs in a DC conduction pump when the liquid metal flows through the fringe magnetic field at the pump ends. This loss can be minimized by proper pole piece shaping and location with respect to the electrode assembly.

3) Reduce bypass, throat wall current — A significant pump loss results from current being shunted around the liquid metal through the throat duct walls. This loss can be eliminated or minimized by use of nonconducting materials. The chief problem in this method lies in obtaining a suitable ceramic-to-metal joint.

The initial, overall program objective was to evaluate these efficiency augmenting techniques through analytical and test data and to demonstrate their effectiveness through performance and endurance testing of a pump utilizing these new features.

Items 1 and 2 were satisfactorily completed and reported.^(22,23) The objectives for this reporting period were to:

1) Perform evaluation of NaK corrosion test specimens from selected ceramic-braze alloy-metal systems to enable selection of an acceptable system for use in pump constructions.

2) Construct a pump for performance and endurance tests in a NaK environment. The pump would utilize a ceramic-metal joint elected on the basis of the corrosion test work.

1. Corrosion Test

The NaK-78 corrosion testing of 24 ceramic-braze-metal samples was completed. Metallographic specimens from the tested, as well as the control, samples have been prepared, examined and evaluated. The test histories of the samples fabricated from four braze alloy systems, three ceramics and one base metal (niobium) are shown in Table 5. Also included in the table is a rating of "satisfactory" or "unsatisfactory" based on the evaluation of the metallographic specimens.

TABLE 5
NaK CORROSION TEST SAMPLE SUMMARY

Braze Alloy	Braze Composition (wt %)	Braze Melt Temperature (°F)	Braze Form	Time at 1100 and 1300°F, Hours and Rating					
				BeO/Nb		Al ₂ O ₃ /Nb (Lucalox)		Al ₂ O ₃ /Nb (Sapphire)	
				Time	Rating	Time	Rating	Time	Rating
Ti-Ni	72/28	1751	Foils	500	Sat*	500	Unsat†	500	Unsat
ZrBe	95/5	1805	Wire	518	Sat	518	Sat-1100°F	518	Sat-1100°F
ZrNbBe	75/15/6	1920	Powder	593	Sat	593	Sat	593	Sat
ZrVTi	56/28/16	2280	Powder	593	Sat	593	Sat	593	Sat-1100°F

*Satisfactory

†Unsatisfactory

As may be seen, only the niobium-BeO combination provided satisfactory results with all four braze alloy systems after exposure for 500 hr in both 1100 and 1300°F NaK. In addition, one Nb-ZrBe-BeO sample that had been placed on test in a 1300°F NaK loop during earlier work in this program was removed and examined after an exposure of 2185 hr. No evidence of corrosive attack was observed.

The ratings were based upon general "brazability," survival of the test exposure, and adequate corrosion resistance in the braze region as determined by metallographic examination. The corrosive attack observed always occurred at the braze ceramic interface.

Cracking was observed in both the ceramic and braze zones of some samples. The former was due in part to the cutting and polishing procedures employed for preparation of metallographic specimens.

Proper metallographic polishing procedures have not been developed for the samples of this program. Therefore, in most instances the niobium appears pitted, and in some instances the braze and ceramic appear chipped or pitted. Even with these defects, the metallographic specimens were adequate for purposes of evaluation and selection of a ceramic-braze-base metal system.

Photomicrographs of the control and exposed specimens are shown in Figures 3 through 34.

2. Pump Fabrication

Because of funding limitations, no work was performed during the quarter under this phase of the program.

C. BASIC SHIELDING ANALYSIS

1. STIR Experiments Planning and Analysis

Analyses were made for STIR heavy metal/lithium hydride shield configurations. The QAD code was selected for the calculation of the dose rate due to fast neutrons, primary gamma rays originating in the disk-shaped fission plate, and secondary gamma rays resulting from neutron interactions within each shield. The ANISN code was used to calculate the neutron levels and secondary gamma source strengths within each shield.

Prior to their use in these heavy metal/lithium hydride cases, the ANISN code and its attendant input data were tested. Preliminary calculations were made for a point isotropic fission neutron source surrounded by both water and lithium hydride. As a result of recent ORNL studies of the effects of the angular mesh description⁽²⁴⁾ and of the order of the Legendre series expansion of the angular dependence of the scattering cross section, S16 quadrature coefficients and the P3 approximation were used.

Table 6 summarizes the ANISN thermal neutron ($E_n < 0.414$ ev) results for the water case. The respective neutron group structures for the 20- and 27-group cross section sets are tabulated in Table 7. Figure 35 presents a graphic comparison of the ANISN calculations with LTF gold foil measurements⁽²⁵⁾ which had been transformed from disk source to point source geometry.

Table 8 summarizes the ANISN fast neutron dose rate calculations in water. The Hurst fast neutron dosimeter response factors used in this analysis are tabulated for the 27-group formulation in Table 7. Figure 36 illustrates the comparison of the ANISN data (transformed to LTF fission plate disk geometry) with the LTF Hurst dosimeter measurements.

BeO

BRAZE

Nb

500X

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Figure 3. Nb/Ti-Ni/BeO Braze Region
in the As-Brazed Condition

BeO

BRAZE

Nb

500X

7697-2599

3-2-67 UNC

Figure 4. Nb/Ti-Ni/BeO Braze Region Following
500-hr 1100°F NaK Exposure

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Figure 5. Nb/Ti-Ni/Sapphire Braze Region Following
500-hr, 1300°F NaK Exposure

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Figure 6. Nb/Ti-Ni/Sapphire Braze Region - Control

SAPPHIRE

BRAZE

Nb

500X

3-2-67 UNC

7697-25102

Figure 7. Corrosion at Sapphire-Braze Interface
of Nb/Ti-Ni/Sapphire Specimen Following
500-hr, 1100°F NaK Exposure

BeO

BRAZE

Nb

200X

3-2-67 UNC

7697-25103

Figure 8. Nb/ZrBe/BeO Braze Region in As-Brazed Condition

3-2-67 UNC

7697-25104

Figure 9. Nb/ZrBe/BeO Braze Region Following
518-hr, 1100°F NaK Exposure

3-2-67 UNC

7697-25105

Figure 10. Nb/ZrBe/BeO Braze Region Following
518-hr, 1300°F NaK Exposure (Note slight
oxidation at edge of braze.)

3-2-67 UNC

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Figure 11. Nb/ZrBe/BeO Braze Region Following
2185-hr, 1300°F NaK Exposure

3-2-67 UNC

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Figure 12. Nb/ZrBe/Sapphire Braze Region
in As-Brazed Condition

BRAZE

Nb

SAPPHIRE

200X

3-2-67 UNC

7697-25108

Figure 13. Nb/ZrBe/Sapphire Braze Region
After 518-hr, 1100°F NaK Exposure

BRAZE

Nb

SAPPHIRE

200X

3-2-67 UNC

7697-25109

Figure 14. Nb/ZrBe/Sapphire Braze Region After 518-hr,
1300°F NaK Exposure (Note corrosive attack
at sapphire-bronze interface.)

Figure 15. Nb/ZrBe/Lucalox Braze Region
in As-Brazed Condition

Figure 16. Nb/ZrBe/Lucalox Braze Region After
518-hr, 1100°F NaK Exposure

Figure 17. Nb/ZrBe/Lucalox Braze Region After 518-hr, 1300°F NaK Exposure (Note corrosive attack at braze-ceramic interface.)

Figure 18. Nb/ZrNbBe/BeO Braze Zone - Control

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Figure 19. Nb/ZrNbBe/BeO Braze Region Following
593-hr, 1300°F NaK Exposure

3-2-67 UNC

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Figure 20. Nb/ZrNbBe/Sapphire Braze Region — Control

3-2-67 UNC

SAPPHIRE

BRAZE

Nb

200X

7697-25116

Figure 21. Nb/ZrNbBe/Sapphire Braze Zone After 593-hr, 1100°F NaK Exposure (Dark region in braze is brazing void.)

3-2-67 UNC

SAPPHIRE

BRAZE

Nb

200X

7697-25117

Figure 22. Nb/ZrNbBe/Sapphire Braze Region After 593-hr, 1300°F NaK Exposure

LUCALOX

BRAZE

Nb
200X

3-2-67 UNC

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Figure 23. Nb/ZrNbBe/Lucalox Braze Region - Control

LUCALOX

BRAZE

Nb
200X

3-2-67 UNC

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Figure 24. Nb/ZrNbBe/Lucalox Braze Region
After 593-hr, 1100°F NaK Exposure

LUCALOX

BRAZE

200X

7697-25120

3-2-67 UNC

Figure 25. Nb/ZrNbBe/Lucalox Braze Region
Following 593-hr, 1300°F NaK Exposure

BeO

BRAZE

Nb

200X

3-2-67 UNC

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Figure 26. Nb/Zr/VTi/BeO Braze Region - Control

Figure 27. Nb/ZrVTi/BeO Braze Region
After 593-hr, 1100°F NaK Exposure

Figure 28. Nb/ZrVTi/BeO Braze Zone
After 593-hr, 1300°F NaK Exposure

SAPPHIRE

BRAZE

Nb

3-2-67 UNC

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Figure 29. Nb/ZrVTi/Sapphire Braze Zone Control

SAPPHIRE

BRAZE

Nb

200X

3-2-67 UNC

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Figure 30. Nb/ZrVTi/Sapphire Braze Zone
Following 593-hr, 1100°F NaK Exposure

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Figure 31. Nb/ZrVTi/Sapphire Braze Zone After 593-hr 1300°F NaK Exposure (Note separation at braze-sapphire interface due to corrosion.)

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Figure 32. Nb/ZrVTi/Lucalox Braze Region - Control

LUCALOX

BRAZE

Nb
200X

3-2-67 UNC

7697-25128

Figure 33. Nb/ZrVTi/Lucalox Braze Region
After 593-hr, 1100°F NaK Exposure

LUCALOX

BRAZE

Nb
200X

3-2-67 UNC

7697-25129

Figure 34. Nb/ZrVTi/Lucalox Braze Zone
Following 593-hr, 1300 °F NaK Exposure

TABLE 6
COMPARISON OF ANISN RESULTS WITH EXPERIMENTAL DATA -
THERMAL NEUTRON FLUX DUE TO POINT ISOTROPIC
FISSION NEUTRON SOURCE IN WATER

Radius (cm)	P3, S16 ANISN Results, Thermal Neutron Flux (4.4 ev)		Experimental Thermal Flux	ANISN/Experiment	
	20 Groups	27 Groups		20 Groups	27 Groups
0.25	1.55×10^{-2}	1.55×10^{-2}	—	—	—
0.75	1.54×10^{-2}	1.54	—	—	—
1.5	1.48×10^{-2}	1.48	—	—	—
2.5	1.36×10^{-2}	1.36	—	—	—
3.5	1.20×10^{-2}	1.20	—	—	—
4.5	1.02×10^{-2}	1.02	—	—	—
5.5	8.40×10^{-3}	8.40×10^{-3}	—	—	—
6.5	6.78×10^{-3}	6.78	—	—	—
8	4.84×10^{-3}	4.83	—	—	—
10	2.89×10^{-3}	2.88	2.26×10^{-3}	1.28	1.27
20	1.73×10^{-4}	1.72×10^{-4}	2.06×10^{-4}	0.84	0.84
30	1.39×10^{-5}	1.48×10^{-5}	1.93×10^{-5}	0.72	0.77
40	1.61×10^{-6}	1.84×10^{-6}	2.31×10^{-6}	0.70	0.80
50	2.54×10^{-7}	3.06×10^{-7}	3.81×10^{-7}	0.67	0.80
60	4.89×10^{-8}	6.10×10^{-8}	6.81×10^{-8}	0.72	0.90
70	1.06×10^{-8}	1.34×10^{-8}	1.44×10^{-8}	0.74	0.93
80	2.58×10^{-9}	3.34×10^{-9}	3.51×10^{-9}	0.74	0.95
90	6.79×10^{-10}	8.80×10^{-10}	8.76×10^{-10}	0.78	1.00
100	1.91×10^{-10}	2.51×10^{-10}	2.83×10^{-10}	0.80	1.05
110	5.72×10^{-11}	7.34×10^{-11}	6.86×10^{-11}	0.83	1.07
120	1.80×10^{-11}	2.16×10^{-11}	2.08×10^{-11}	0.87	1.04
130	5.45×10^{-12}	7.01×10^{-12}	6.72×10^{-12}	0.81	1.04
140	1.77×10^{-12}	2.23×10^{-12}	2.23×10^{-12}	0.79	1.00
150	5.76×10^{-13}	7.26×10^{-13}	7.95×10^{-13}	0.73	0.91
160	1.93×10^{-13}	2.44×10^{-13}	2.98×10^{-13}	0.65	0.82
170	6.56×10^{-14}	8.23×10^{-14}	—	—	—
180	2.22×10^{-14}	2.77×10^{-14}	—	—	—
190	2.79×10^{-15}	3.27×10^{-15}	—	—	—

TABLE 7
NEUTRON ENERGY GROUP DATA

Neutron Energy Group	Maximum Neutron Energy (ev)		Hurst Dosimeter Response for 27-Group Structure, ⁽²⁵⁾ (rad/hr per $10^5\text{n/cm}^2\text{-sec}$)
	20 Groups	27 Groups	
1	1.492×10^7	1.492×10^7	1.9648
2	1.000×10^7	1.221×10^7	1.8503
3	7.408×10^6	1.000×10^7	1.7423
4	6.065×10^6	8.187×10^6	1.6788
5	4.493×10^6	6.703×10^6	1.6183
6	3.329×10^6	5.488×10^6	1.5587
7	2.466×10^6	4.493×10^6	1.5518
8	1.353×10^6	3.679×10^6	1.4165
9	9.072×10^5	3.012×10^6	1.2248
10	4.076×10^5	2.466×10^6	1.1308
11	1.111×10^5	2.019×10^6	1.0625
12	1.503×10^4	1.653×10^6	0.9685
13	3.355×10^3	1.353×10^6	0.9035
14	5.829×10^2	1.108×10^6	0.8454
15	1.013×10^2	9.072×10^5	0.6932
16	2.902×10^1	6.081×10^5	0.5813
17	1.068×10^1	4.076×10^5	0.3960
18	3.059×10^0	1.111×10^5	0.1597
19	1.125×10^0	1.503×10^4	0.0215
20	4.140×10^{-1}	3.355×10^3	0.0000
21	—	5.829×10^2	0.0000
22	—	1.013×10^2	0.0000
23	—	2.902×10^1	0.0000
24	—	1.068×10^1	0.0000
25	—	3.059×10^0	0.0000
26	—	1.125×10^0	0.0000
27	—	4.140×10^{-1}	0.0000

Figure 35. Point Isotropic Fission Neutron Source at Center of 190 cm Sphere of Water

Figure 36. Fast Neutron Dose Rate in Water vs Distance from LTF Source

TABLE 8
COMPARISON OF ANISN RESULTS WITH EXPERIMENTAL DATA - FAST
NEUTRON DOSE RATE IN WATER DUE TO LTF FISSION PLATE

Distance From LTF Source Plate (cm)	Fast Neutron Dose Rate (erg/gm-hr)		ANISN/ Experiment
	ANISN Calculation	LTF Measurement	
0.25	1.86×10^5	—	—
0.75	1.45×10^5	—	—
1.5	8.86×10^4	—	—
2.5	6.35×10^4	—	—
3.5	4.52×10^4	—	—
4.5	3.36×10^4	—	—
5.5	2.63×10^4	—	—
6.5	2.07×10^4	—	—
8	1.46×10^4	—	—
10	9.27×10^3	—	—
20	1.36×10^3	1.70×10^3	0.80
30	2.62×10^2	—	—
40	5.80×10^1	6.10×10^1	0.95
50	1.41×10^1	—	—
60	3.70×10^0	4.25×10^0	0.87
70	1.03×10^0	—	—
80	2.98×10^1	3.35×10^1	0.89
90	8.96×10^{-2}	—	—
100	2.78×10^{-2}	3.05×10^{-2}	0.91
110	8.81×10^{-3}	—	—
120	2.86×10^{-3}	3.30×10^{-3}	0.90
130	9.44×10^{-4}	—	—
140	3.17×10^{-4}	—	—
150	1.08×10^{-4}	—	—
160	3.72×10^{-5}	—	—
170	1.30×10^{-5}	—	—
180	4.54×10^{-6}	—	—

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Figure 37. Comparison of P3-S16, 27-GP ANISN Results (Transformed to a STIR Disk Source) with STIR Indium Foil Measurements

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Figure 37 presents a comparison of preliminary ANISN calculations of the thermal neutron flux in lithium hydride with STIR foil measurements.⁽²⁶⁾ The experimental data were not measured absolutely and have been normalized to the calculated results at a lithium hydride shield penetration distance of 10 cm. The discrepancy exhibited in the slopes of the measured and calculated fluxes was attributed to poor lithium cross section data. Subsequently, a new set of lithium cross sections was obtained from ORNL.⁽²⁷⁾ These new data have been demonstrated by ORNL to yield ANISN results which are in substantial agreement with O5R (Monte Carlo) calculations. Furthermore, the O5R calculations themselves compare well with lithium hydride fast neutron measurements.⁽²⁸⁾ These ORNL lithium cross sections were used in all the STIR heavy metal/lithium hydride shield analyses. Preliminary comparisons of ANISN calculations of the thermal neutron flux in lithium hydride exhibit good agreement with the shape of recent STIR foil measurements. Comparison of the absolute magnitudes of the calculated and experimental data is also being investigated.

2. Shield Code Development

A description of the mathematical bases and formulation for the GRAMS code has been prepared in NAA-SR-12337, entitled "GRAMS: A FORTRAN Program to Solve Gamma Ray Penetration Problems in Multi-Slab Shields, Part 1 - Numerical Methods." In addition to space-angle-energy distribution of x-rays, the code computes dose profiles, deposited heat density, integrated energy fluxes, transmitted angular distributions, buildup factors, etc., from the basic solution.

QAD calculations of the variation of the primary gamma dose rate with depleted uranium shield thickness appear to be in close agreement with measured data. Since all QAD calculations have not been completed, however, the secondary gamma dose rate has not been fully evaluated.

D. BASIC SHIELDING EXPERIMENTS (Composite Material Shielding Program)

The attenuation of gamma rays through a two-cycle (four-region) lead-lithium hydride shield has been measured. The shield dimensions were arbitrarily picked to coincide with those of the previously measured uranium-lithium hydride shield.⁽²⁹⁾ The first cycle consisted of 2 in. of lead and 4 in. nominal (3-3/4 in. actual) of lithium hydride, while the second cycle consisted of 4 in. of lead and

6 in. nominal (5-3/4 in. actual) of lithium hydride. Two measurements were made on these configurations; one with the lead regions first and one with the lithium hydride first in each cycle. The results are shown in Figure 38.

The measurements show a 70% increase in dose rate when the lead regions are closest to the fission plate. This effect is due to secondary gamma ray production. By comparison, similar measurements in the uranium-lithium hydride system gave a 700% increase due to the greater secondary production in uranium.⁽²⁹⁾

Neutron dose measurements were made through lithium hydride using a Hurst dosimeter in the collimator as the detector. The measurements were made through a maximum lithium hydride thickness of 74 cm, compared to previous measurements through 38 cm². The results, shown in Figure 39, give an attenuation coefficient of 0.179 cm⁻¹ which is in good agreement with the previous measurement.⁽³⁰⁾ The greater thickness of measurement was made possible by the increase in reactor power from 50 kw to 1 Mw and by the collimator, which reduced the sensitivity of the detector to extraneous neutrons.

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Figure 38. Gamma Attenuation in Two-Cycle Lead Lithium Hydride Shield

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Figure 39. Hurst Dosimeter Measurements in LiH Slabs

The distribution of neutron flux through lithium hydride was measured using a number of foils to detect thermal, resonance, and threshold reactions. Gold and indium were used in the thermal and resonance region, while sulfur and aluminum were used as threshold detectors. The results of these measurements are shown in Figure 40. The longer attenuation length as shown by the aluminum threshold detector indicates that high energy neutrons (>8 Mev) are more penetrating and may become the dominant component in thick shields.

E. BASIC SHIELD MATERIALS TECHNOLOGY

1. Gamma Shield Development

The twelve 1000-, 2000-, and 4000-hr compatibility test capsules containing aggregates of tungsten alloy-lithium hydride, or zirconium hydride-lithium hydride at 800 and 1000°F were opened and the specimens visually examined.* No significant changes were observed. Figures 41 and 42 show the two sets of 4000-hr specimens. The only change noticeable was a darkening, shown in Figure 43, of the normally metallic-appearing $\text{ZrH}_{1.6-1.7}$ in the 1000°F specimen. It is believed that the darkness is caused by a thin layer of black ZrH_2 .

The specimens were broken into two pieces. One piece was submitted for chemical analysis and the other piece used for a density measurement. After the density of the specimen was measured, the sample was reacted with water to remove the lithium hydride. The residual gamma shield material was washed, dried, and weighed. From these weights the composition and the mixture density of the samples were calculated for comparison with the measured densities.

Table 9 summarizes the data currently available. The results of the chemical analysis and the remainder of the density measurements will not be available until next quarter.

2. Lithium Hydride Technology

Chemical analysis of the four samples of lithium hydride irradiated in the STIR was completed. Four unirradiated control samples also were analyzed. Table 10 shows the results of the analysis. Unfortunately some of the control samples were accidentally contaminated with water vapor during preparation for analysis, preventing comparison of the pre- and post-irradiation values for

*A Topical Report (NAA-SR-9400, Vol. IV), now in preparation, describes the complete aggregate neutron-gamma shielding program in detail.

a. Thermal and Resonance Energy

b. Fast

Figure 40. Neutron Flux Distributions in LiH

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Figure 41. 800 and 1000°F, 4000-hr Lithium Hydride-Tungsten Alloy Compatibility Specimens

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Figure 42. 800 and 1000°F, 4000-hr Lithium Hydride-Zirconium Hydride Compatibility Specimen

TABLE 9
DENSITY AND COMPOSITION OF LiH/W ALLOY AND
LiH/ZrH_x ROD SPECIMENS

Sample	History	Composition		Sample Void Percent	Density g/ml	Density	Mixture Density g/ml
		LiH Wt %	Aggre- gate Wt %			g/ml Corrected to 100% Dense	
W/LiH No. 1	As cast in stainless steel tube	4.0	96.0	1.5	9.37	9.52	9.53
W/LiH No. 2	As cast in stainless steel tube	ND	ND	—	8.96	ND	ND
W/LiH No. 1C	Untested control specimen*	6.0	94.0	0.7	7.92	7.97	7.89
W/LiH No. 2C	Untested control specimen*	4.7	95.3	1.3	8.80	8.93	9.02
W/LiH No. 3C	Untested control specimen*	5.3	94.7	1.1	8.31	8.42	8.45
W/LiH No. 4C	Untested control specimen*	5.1	94.9	1.5	8.43	8.57	8.58
W/LiH No. 5C	Untested control specimen*	5.0	95.0	3.9	8.29	8.63	8.68
W/LiH No. 6C	Untested control specimen*	5.3	94.7	1.5	8.39	8.52	8.42
W/LiH No. 1A	Heated 800°F, 1000 hours*	ND	ND	ND	ND	ND	ND
W/LiH No. 2A	Heated 800°F, 2000 hours*	5.3	94.7	2.3	8.14	8.33	8.45
W/LiH No. 3A	Heated 800°F, 4000 hours*	5.0	95.0	2.6	8.23	8.45	8.67
W/LiH No. 1B	Heated 1000°F, 1000 hours*	5.2	94.8	1.8	8.11	8.26	8.57
W/LiH No. 2B	Heated 1000°F, 2000 hours*	4.8	95.2	1.5	8.73	8.89	8.92
W/LiH No. 3B	Heated 1000°F, 4000 hours*	4.4	95.6	2.0	9.00	9.18	9.33
ZrH _x /LiH No. 1	As cast in stainless steel tube	ND	ND	ND	3.55	ND	ND
ZrH _x /LiH No. 2	As cast in stainless steel tube	ND	ND	ND	3.88	ND	ND
ZrH _x /LiH No. 3	As cast in stainless steel tube	ND	ND	ND	3.71	ND	ND
ZrH _x /LiH No. 1A	Heated 800°F, 1000 hours*	IP	IP	IP	IP	IP	IP
ZrH _x /LiH No. 2A	Heated 800°F, 2000 hours*	IP	IP	IP	IP	IP	IP
ZrH _x /LiH No. 3A	Heated 800°F, 4000 hours*	10.3	89.7	1.3	3.30	3.35	IP
ZrH _x /LiH No. 1B	Heated 1000°F, 1000 hours*	IP	IP	IP	IP	IP	IP
ZrH _x /LiH No. 2B	Heated 1000°F, 2000 hours*	16.2	83.8	2.0	2.77	2.86	IP
ZrH _x /LiH No. 3B	Heated 1000°F, 4000 hours*	7.9	92.1	3.0	3.16	3.26	IP

* = Cast in Titanium Tube
ND = Not Determined
IP = In Process

TABLE 10
SUMMARY OF LiH - STIR TEST CHEMICAL ANALYSIS

Sample Number	Weight Percent of Constituent							
	Hydroxide*		Carbonate		Lithium		Hydrogen	
	Control	Irrad.	Control	Irrad.	Control	Irrad.	Control	Irrad.
1. Cast LiH, High Purity Minimum Contamination	1.4	2.4	ND	0.002	85.4	85.4	12.36	12.18
2. Cast LiH, Surface Layer of Contamination†	ND	2.4(a)	ND	0.002(a)	84.1	84.8(a)	11.54(a and b)	12.19(a)
	ND	27.5(b)	ND	3.05(b)	ND	21.4(b)		11.03(b)
3. Cold Pressed LiH, High Purity, Minimum Contamination	ND	6.2	ND	0.14	85.8	80.9	12.30	11.67
4. Cold Pressed LiH, plus about 11 wt % LiOH Contamination†	ND	11.2	ND	0.25	77.1	76.1	10.11	9.82

*Chemical analyses methods cannot differentiate between lithium oxide and lithium hydroxide.

†A portion of the LiOH contaminant probably exists as the hydrate, $\text{LiOH} \cdot \text{H}_2\text{O}$. No analysis was made for the weight percent of water.

ND = Not Determined

a = Center of Specimen

b = Surface of Specimen

a and b = Combined

hydroxide and carbonate in Samples 2, 3, and 4. In general, no significant changes in composition were noted, except in the irradiated Sample 3. The relatively low values for lithium and hydrogen, as compared to the controls, indicate that the specimen was contaminated after fabrication. Since no other specimen showed a change of this magnitude, it is suspected that the contamination occurred after irradiation.*

F. SNAP 10A FS-3 POST-TEST EXAMINATION

As explained in the preceding quarterly progress report, the purpose of the detailed examination of the 10FS-3 fuel elements and selected components is to thoroughly evaluate the operational performance of fuel, cladding, and hydrogen barrier, and to a lesser degree that of the internal reflectors, core hold-down springs, NaK plumbing (handlebars), external reflector assembly, reflector band, and core vessel.

During the report period, the entire hot laboratory effort was directed toward the examination of selected reactor components. The examination of these components was completed with the exception of metallography of samples taken from the core hold-down springs, NaK plumbing, and internal reflectors. A brief summary of the components examined and the extent of their examination follows below.

- 1) Internal reflectors — visual inspection and density measurement.
- 2) External reflectors — visual inspection, measurement of backlash in fine control drum drives, measurement of drum rotation and shaft oscillation torques, inspection of bearings, ball and sockets, and gears, measurement of actuator winding resistances, actuator torques and brake torques, and measurement of position sensor torques.
- 3) Core hold-down springs — measurement of free length and spring constant.

*A Topical Report (NAA-SR-9400, Vol. III), now in preparation, describes the complete test in detail, and correlates the results of this test with other lithium hydride irradiations.

4) Handlebars (NaK outlet plumbing) – visual inspection, and sectioning for metallography.

5) Reflector band – visual inspection and disassembly.

1. Internal Reflectors

a. Visual Inspection

The six beryllium internal reflectors were removed from the core, cleaned of NaK oxide, and visually inspected. Typical reflector appearance is shown in Figure 43. The reflector surfaces were free of defects, but did exhibit a number of small black stains; the stains appear to have occurred during cell handling and residency. Some of the spots were removed or altered by wiping.

b. Density

The density of each of the six reflectors was determined by the liquid displacement method using distilled water as the fluid. Each density measurement was made three times and averaged. The data are summarized in Table 11.

2. External Reflectors

a. Visual Inspection

Both reflector block assemblies were visually inspected and photographed by means of the Kollmorgen wall periscope. The assemblies were identified by the serial number of the fine control drum actuator on each (2817 and 2834). The block assembly bearing the 2817 actuator is shown in Figure 44. No defects were observed. The gears appeared to be in excellent condition.

The block assembly bearing the 2834 actuator was similarly free of defects except for the scratches on the inside surface, shown in Figure 45. The scratches suggested tool marks which might have been made during fabrication.

b. Drum Drive Gear Backlash

Backlash measurements were taken with a dial indicator held in a specially designed fixture keyed to the upper drum support bracket. Nine measurements were taken at approximately equal increments of fine control drum travel from full out to full in for each reflector block assembly. The data are listed in Table 12 for each fine control drum as identified by the corresponding actuator serial number.

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Figure 43. S10FS-3 Typical Internal Reflector

TABLE 11
INTERNAL REFLECTOR DENSITIES

Reflector [*] Number	Density (gm/cc)			
	Measurement No. 1	Measurement No. 2	Measurement No. 3	Average
1	1.8578	1.8578	1.8584	1.8580
2	1.8584	1.8585	1.8584	1.8584
3	1.8585	1.8584	1.8585	1.8585
4	1.8581	1.8581	1.8580	1.8581
5	1.8572	1.8572	1.8572	1.8572
6	1.8574	1.8575	1.8575	1.8575

*Reflectors are numbered in a clockwise direction looking down on the core, starting with the first reflector to the right of the lower grid plate indexing pin.

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Figure 44. External Reflector Block Assembly
(S/N 2817)

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Figure 45. External Reflector Block Assembly
(S/N 2834)

NAA-SR-123 16

TABLE 12
FINE CONTROL DRUM BACKLASH DATA

Drum Position	Assembly Backlash (in.)	
	2817	2834
1 (full out)	0.0020	0.0040
2	0.0030	0.0050
3	0.0030	0.0045
4	0.0020	0.0040
5	0.0030	0.0065
6	0.0025	0.0050
7	0.0020	0.0060
8	0.0020	0.0055
9 (full in)	0.0020	0.0045

c. Fine Control Drum Rotation Torque

The actuators and position sensors were removed from both fine control drum assemblies which allowed the drums to turn, restrained only by the bearing drag. The torque required to overcome this drag and start the drums rotating was measured at several drum positions with a small torque meter. Care was exercised to assure that the drum shafts were positioned vertically. The test data are presented in Table 13.

d. Drum Shaft Oscillation Torque

Shaft oscillation torques for each fine control drum in the intermediate position were measured three times. Torques for the drum associated with the 2817 actuator were 8 to 9 oz-in. and 11.5 to 12 oz-in. for that associated with the 2834 actuator.

e. Bearing, Ball and Socket, and Gear Inspection

The drum bearings were disassembled and visually examined. Ball and socket surfaces showed evidence of contact and pressure in the form of small,

TABLE 13
FINE CONTROL DRUM ROTATION TORQUE

Drum Position	Torque (oz-in.)	
	Drum Corresponding to Actuator 2817	Drum Corresponding to Actuator 2834
1 (out-rotating in)	4.0	7.5
2 (intermediate)	8.0	12.0
3 (in)	11.0	11.0
4 (in-rotating out)	6.0	4.0
5 (intermediate)	8.0	7.5
6 (out)	9.5	11.0

apparently burnished areas, but there was little indication of severe wear or other damage to the surfaces. Figures 46 and 47 illustrate typical ball and socket condition.

The shafts were removed from the balls and examined. Again, although the surfaces showed some evidence of use and were somewhat darkened, there was no indication of significant wear or damage. Figure 48 shows a typical shaft surface compared to an unused shaft.

It should be noted that the bearings turned very freely during all phases of in-cell handling and disassembly. Drum drive gear teeth were examined through the Kollmorgen periscope. No evidence of damage or severe wear could be detected. Tooth faces on the drum sector gear were slightly burnished but to a barely detectable extent. Typical appearance is shown in Figure 49.

f. Actuator Winding Resistances

Actuator winding resistances were measured by connecting a cable of known resistance to each actuator, measuring cable plus winding resistance out-of-cell, and then subtracting the cable resistance from the gross value. The cell temperature was 102°F. At the time the resistances were measured, the actuators

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Figure 46. Fine Control Drum Shaft Ball—
External Reflector

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Figure 47. Fine Control Drum Typical Socket
Surface — External Reflector

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Figure 48. Fine Control Drum Typical Shaft Bearing Surface Compared to New Shaft

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Figure 49. Typical Appearance of Fine Control Drum Drive Gear

had been in this cell environment for several days with no electrical power applied to the windings. It was therefore assumed that the winding temperature was also 102°F when the measurements were made. The resistance measurements are presented in Table 14.

g. Actuator Torques

Stalling torques for actuators 2817 and 2834 were measured with currently calibrated torque meters which covered the range from 4 to 108 oz-in. torque. Actuator brakes were energized (released) for each stalling torque determination. Test data are presented in Table 15.

Torque required to override the de-energized brake for each actuator was as follows:

<u>2817</u> <u>(oz-in.)</u>	<u>2834</u> <u>(oz-in.)</u>
36 clockwise	35 clockwise
37 counterclockwise	34 counterclockwise

TABLE 14
ACTUATOR MOTOR WINDING
RESISTANCES

Winding	Actuator Resistances (ohms)	
	2817	2834
A to E	8.304	8.102
B to E	8.219	7.234
C to E	8.007	7.607
D to E	8.627	7.602

TABLE 15
ACTUATOR STALLING
TORQUES

Voltage	Stalling Torque (oz-in.)	
	2817	2834
1.5	—	—
3.0	8	5
4.5	10	9.5
6.0	33	22
8.0	42	35
12.0	79	49
16.0	104	61
23.0	105	87
25.0	108	93
26.0	108+	—
77.0	—	100
30.0	—	107
33.0	—	108

Actuator bearing torques measured with the brakes released and no power to the motors were as follows:

<u>2817</u> <u>(oz-in.)</u>	<u>2834</u> <u>(oz-in.)</u>
16 clockwise	8 clockwise
17 counterclockwise	10 counterclockwise

h. Position Sensor Torques

Only one of the two position sensors could be identified by a serial number. This was sensor 1030 and was the sensor associated with the drum driven by actuator 2834.

Varying mechanical torques in the 4 to 7 oz-in. range were measured for both sensors with no electrical power applied. With a 26 volt, 400 cps power supply, the torque increased slightly but results were not conclusive. Apparently the varying mechanical drag of the sensors masked the small electrically-induced torque to the extent that reliable measurements could not be made. It should be noted that removal of the sensors from the reflector assemblies was accomplished with difficulty. It is possible that mechanical damage such as some distortion to the sensor shafts may have occurred which precluded the accurate measurement of electrically-induced torques.

3. Core Hold-Down Springs

Spring deflection tests were performed in-cell on 10 unirradiated and 12 irradiated core hold-down springs. Each Rene 41 spring was placed between the jaws of a remote Instron TT-C testing machine which was used to apply equal increments of compressive load. The deflection produced at each load increment was subtracted from the original spring free length to obtain successive spring absolute lengths (see Tables 16 and 17). Data from Spring 2B is not presented, since misalignment in the test fixture produced excessive torque which eventually broke the spring. Figure 50 shows a plot of absolute length versus load, in which each point is the average of all spring values at a particular load. Both the unirradiated and irradiated curves show a linear decrease of spring length with increasing load until a point is reached at which smaller deflections for the same increment of load were measured. At this point, the springs were

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Load (lb)	Spring No. 1A		Spring No. 1B		Spring No. 2A		Spring No.. 2B		Spring No. 3A		Spring No. 3B		Spring No. 4A		Spring No. 4B		Spring No. 5A		Spring No. 5B		Spring No. 6A		Spring No. 6B		Average	
	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Deflection	Absolute Length
	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)
0	0.000	0.585*	0.000	0.598*	0.000	0.592*	0.000	0.591*	0.000	0.600*	0.000	0.603*	0.000	0.581*	0.000	0.602*	0.000	0.577*	0.000	0.601*	0.000	0.583*	0.000	0.597*	0.000	0.593
5	0.019	0.566	0.024	0.574	0.016	0.576	Broken ↓		0.020	0.580	0.016	0.587	0.024	0.557	0.024	0.578	0.015	0.562	0.027	0.574	0.019	0.564	0.019	0.578	0.020	0.572
10	0.035	0.550	0.039	0.559	0.032	0.560			0.035	0.565	0.031	0.572	0.037	0.544	0.037	0.565	0.028	0.549	0.041	0.560	0.034	0.549	0.034	0.563	0.034	0.557
15	0.052	0.533	0.053	0.545	0.046	0.546			0.050	0.550	0.046	0.557	0.051	0.530	0.051	0.551	0.042	0.535	0.054	0.547	0.047	0.536	0.049	0.548	0.049	0.543
20	0.067	0.518	0.067	0.531	0.059	0.533			0.064	0.536	0.060	0.543	0.070	0.511	0.066	0.536	0.055	0.522	0.070	0.531	0.063	0.520	0.063	0.534	0.064	0.529
25	0.083	0.502	0.081	0.517	0.075	0.517			0.077	0.523	0.075	0.528	0.079	0.502	0.079	0.523	0.070	0.507	0.084	0.517	0.077	0.506	0.076	0.521	0.077	0.514
30	0.099	0.486	0.094	0.504	0.089	0.503			0.094	0.506	0.089	0.514	0.094	0.487	0.094	0.508	0.084	0.493	0.099	0.502	0.091	0.492	0.091	0.506	0.092	0.500
35	0.114	0.471	0.107	0.491	0.103	0.489			0.105	0.495	0.103	0.500	0.104	0.477	0.106	0.496	0.098	0.479	0.113	0.488	0.105	0.478	0.104	0.493	0.104	0.487
40	0.130	0.455	0.121	0.477	0.117	0.475			0.119	0.481	0.118	0.485	0.114	0.467	0.121	0.481	0.112	0.465	0.126	0.475	0.119	0.464	0.128	0.469	0.120	0.472
45	0.144	0.441	0.139	0.459	0.128	0.464			0.137	0.463	0.131	0.472	0.121	0.460	0.135	0.467	0.124	0.453	0.139	0.462	0.138	0.445	0.133	0.464	0.133	0.459
50	0.155	0.430	0.146	0.452	0.138	0.454			0.144	0.456	0.141	0.462	0.124	0.457	0.146	0.456	0.134	0.443	0.152	0.449	0.145	0.438	0.142	0.455	0.142	0.450
55	0.161	0.424	0.157	0.441	0.142	0.450			0.153	0.447	0.147	0.456	0.126	0.455	0.156	0.446	0.142	0.435	0.162	0.439	0.152	0.431	0.143	0.454	0.149	0.443
60	0.164	0.421	0.163	0.435	0.144	0.448			0.157	0.443	0.148	0.455	0.128	0.453	0.167	0.435	0.146	0.431	0.173	0.428	0.154	0.429	0.144	0.453	0.153	0.439
65	0.166	0.419	0.166	0.432	0.146	0.446			0.160	0.440	0.150	0.453	0.129	0.452	0.171	0.431	0.146	0.431	0.178	0.423	0.155	0.428	0.144	0.453	0.155	0.437
70	0.167	0.418	0.167	0.431	0.147	0.445			0.163	0.437	0.150	0.453	0.130	0.451	0.173	0.429	0.146	0.431	0.181	0.420	0.156	0.427	0.144	0.453	0.156	0.436
75	0.168	0.417	0.168	0.430	0.147	0.445			0.164	0.436	0.151	0.452	0.131	0.450	0.174	0.428			0.181	0.420	0.156	0.427				
80	0.168	0.417	0.169	0.429	0.147	0.445			0.164	0.436	0.151	0.452	0.132	0.449	0.175	0.427			0.181	0.420	0.156	0.427				
85	0.168	0.417	0.169	0.429					0.164	0.436	0.153	0.450	0.133	0.448	0.175	0.427										
90			0.169	0.429							0.153	0.450	0.133	0.448	0.175	0.427										
95											0.153	0.450	0.134	0.447												
100													0.135	0.446												
105													0.135	0.446												
110													0.135	0.446												

*This value was measured and did not vary more than ± 0.002 in. between the pre- and post-test measurements.

NOTE: Springs reach nested condition (constant length) at different loads because of variations in free length and number of turns.
Therefore, average lengths are not computed, or plotted in Figure 51, beyond the 70-lb load point.

Table 16. S10FS-3 Irradiated
Spring Data

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Load (lb)	Spring No. 1		Spring No. 2		Spring No. 3		Spring No. 4		Spring No. 5		Spring No. 6		Spring No. 7		Spring No. 8		Spring No. 9		Spring No. 10		Average	
	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Measured Deflection	Calculated Absolute Length	Deflection	Absolute Length
	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)	(in.)
0	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619*	0.000	0.619
5	0.024	0.595	0.023	0.596	0.017	0.602	0.019	0.600	0.024	0.595	0.021	0.598	0.016	0.603	0.015	0.604	0.019	0.600	0.016	0.603	0.019	0.600
10	0.041	0.578	0.036	0.583	0.031	0.588	0.032	0.587	0.039	0.580	0.036	0.583	0.031	0.588	0.031	0.588	0.033	0.586	0.031	0.588	0.030	0.584
15	0.056	0.563	0.050	0.569	0.045	0.574	0.046	0.573	0.052	0.567	0.049	0.570	0.045	0.574	0.046	0.573	0.049	0.570	0.046	0.573	0.048	0.570
20	0.071	0.548	0.064	0.555	0.059	0.560	0.061	0.558	0.065	0.554	0.064	0.555	0.059	0.560	0.060	0.559	0.062	0.557	0.060	0.559	0.062	0.556
25	0.086	0.533	0.079	0.540	0.073	0.546	0.075	0.544	0.080	0.539	0.077	0.542	0.074	0.545	0.075	0.554	0.076	0.543	0.074	0.545	0.076	0.542
30	0.101	0.518	0.093	0.526	0.088	0.531	0.089	0.530	0.094	0.525	0.091	0.528	0.088	0.531	0.090	0.529	0.091	0.528	0.086	0.533	0.091	0.527
35	0.115	0.504	0.107	0.512	0.101	0.518	0.102	0.517	0.110	0.509	0.106	0.513	0.101	0.518	0.105	0.514	0.106	0.513	0.101	0.518	0.105	0.513
40	0.129	0.490	0.122	0.497	0.115	0.504	0.115	0.504	0.124	0.495	0.120	0.499	0.115	0.504	0.119	0.500	0.120	0.499	0.115	0.504	0.119	0.499
45	0.144	0.475	0.136	0.483	0.130	0.489	0.127	0.492	0.139	0.480	0.134	0.485	0.130	0.489	0.134	0.485	0.135	0.484	0.128	0.491	0.133	0.485
50	0.159	0.460	0.149	0.470	0.144	0.475	0.143	0.476	0.155	0.464	0.149	0.470	0.144	0.475	0.148	0.471	0.149	0.470	0.142	0.477	0.148	0.470
55	0.173	0.446	0.163	0.456	0.158	0.461	0.159	0.460	0.169	0.450	0.162	0.457	0.159	0.460	0.164	0.455	0.164	0.455	0.156	0.463	0.163	0.458
60	0.187	0.432	0.176	0.443	0.172	0.447	0.173	0.446	0.184	0.435	0.176	0.443	0.175	0.444	0.177	0.442	0.176	0.443	0.170	0.449	0.177	0.442
65	0.200	0.419	0.184	0.435	0.186	0.433	0.185	0.434	0.196	0.423	0.189	0.430	0.185	0.434	0.186	0.433	0.180	0.439	0.186	0.433	0.187	0.431
70	0.213	0.406	0.200	0.419	0.192	0.427	0.187	0.432	0.197	0.422	0.199	0.420	0.187	0.432	0.189	0.430	0.181	0.438	0.190	0.429	0.193	0.425
75	0.217	0.402	0.202	0.417	0.193	0.426	0.188	0.431	0.199	0.420	0.200	0.419	0.187	0.432	0.190	0.429	0.181	0.438	0.190	0.429	0.194	0.424
80	0.217	0.402	0.202	0.417	0.193	0.426	0.188	0.431	0.199	0.420	0.200	0.419	0.187	0.432	0.190	0.429	0.181	0.438	0.190	0.429	0.194	0.424
85	0.217	0.402	0.202	0.417	0.193	0.426	0.188	0.431	0.199	0.420	0.200	0.419			0.190	0.429						

*This value was measured and did not vary more than ± 0.003 in. between the pre- and post-test measurements.

NOTE: Springs reach nested condition (constant length) at different loads because of differences in the number of turns.
Therefore, average lengths are not computed, or plotted in Figure 51, beyond the 80 lb load point.

Table 17. S10FS-3 Unirradiated
Spring Data

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Figure 50. S10FS-3 Core Hold-Down Springs, Length vs Load

partially nested. Where no increase in deflection was measured with an increase in load, the springs were fully closed. The slope of the linear portion of each curve represents the spring constant. This was determined to be 344.8 lb/in. for both unirradiated and irradiated springs. The design spring constant was 357.4 lb/in. The average nested or solid length of the spring was calculated to be 0.424 in. for unirradiated springs and 0.436 in. for irradiated springs. Variations in nested length are probably due mainly to differences in initial lengths, and are dependent on the number of coils in each spring. Springs could vary considerably in initial free length, since the fabrication tolerance was ± 0.032 in.

Measurements of spring free lengths were taken before and after testing. Unirradiated springs averaged 0.618 in. (range 0.616 to 0.620 in.) before testing, and 0.620 in. (range 0.618 to 0.622 in.) after testing. Irradiated springs averaged 0.593 in. (range 0.577 to 0.603 in.) before testing and 0.594 in. (range 0.573 to 0.607 in.) after testing. In no case did the post-test measurement vary by more than ± 0.002 in. from the pre-test measurement.

It should be concluded, therefore, that the Rene 41 core hold-down springs were not damaged during reactor operations, since no difference in spring constant was noted in unirradiated and irradiated springs, and the springs returned to the original free lengths, indicating elastic behavior.

4. Handle Bars

No defects were seen during the visual examination of the NaK outlet plumbing. Samples for metallographic examination were cut from fill-tube pinch-off areas, and at the junction of instrumentation wells with the main line, to show weld condition and look for evidence of flow impingement. Results of this examination will not be known until the completion of the metallography report.

5. Reflector Band

Only a short length of the reflector band with the spring and spring-housing attached was received at the AIHL. The assembly is shown in Figure 51. One tensile specimen, at most, could have been cut from the band sample and when tested would have given only one data point. Such statistically poor prospects for data appeared to eliminate the need for the previously planned tensile tests

Figure 51. S10FS-3 Reflector Band
Spring and Housing

slated for the band. Similarly, the determinations of spring rate and free length of the band spring would have been limited to only one specimen. Therefore, because of the very few data points available, and because the measurements were not part of the originally planned program, the planned tensile tests and spring tests were omitted from the examination.

G. SUPERSATURATION STUDY

1. Cloud Chamber

As reported previously, the cloud chamber study (funds and facilities provided by the NAA Science Center) has shown that the experimental critical supersaturation data for methanol, ethanol, and hexane agree within $\sim 5\%$ with the theoretical data. For these fluids, in the temperature range investigated, calculations indicate that the smallest of the critical nuclei, those collections of molecules which will continue to grow to a liquid drop, contain at least 50 molecules. Since the theory predicting the critical supersaturation depends strongly on surface tension, which is only known for an effectively infinite plane surface, substances with critical nuclei calculated to contain fewer than 50 molecules may not have agreement of theoretical and experimental critical supersaturation.

Activity during this reporting period was directed toward documenting the previous investigations and preparing for studies with fluids calculated to have smaller critical nuclei. An article summarizing the experimental work and experiment operation was prepared and submitted to the Journal of Chemical Physics. Preliminary attempts were made at photographing the droplet "rain" in the chamber and, with improved lighting facilities, appear feasible. The larger chamber was disassembled for examination and cleaning prior to resuming operations with heptane and higher hydrocarbons having critical nuclei calculated to contain as few as 30 molecules. Assembly and fabrication of parts for the mercury chamber were continued (at $\sim 68^\circ\text{F}$ the mercury critical nucleus is calculated to contain 5 atoms). The Freon condenser was completed by the glass shop, the Freon chamber cover plate reworked, thermocouples were installed in the chamber condensing plate and a pool of mercury was poured onto the condensing surface and left standing to provide, by amalgamation, a surface which mercury will wet.

TABLE 18
SUPERSATURATIONS ATTAINED

Run **	Nozzle Inlet Pressure (psia)	Estimated Supersaturation†	
		$10^{-5} P/P_{\text{sat}}$	Temperature (°F)
1	15.5	14.0§	13
2	13.0	14.7§	10
3	12.2	24.9	-2
4*	40.3	2.12	65
5*	39.8	2.09	65
6*	33.8	2.81	56
7*	47.7	1.07	85
8*	56.8	0.98	91
9*	47.8	1.07	85
10A*	54.7	0.87	93
10B*	47.2	1.11	84
11*	44.4	1.19	81
12	33.5	0.56	88
13*	33.8	2.67	57
14*	44.8	1.20	81
15*	43.7	1.23	80
17*	45.2	1.16	82
18*	49.0	1.06	86
19*	41.1	1.95	67
20	28.9	0.66	81
21	30.2	5.61	39
22	30.5	3.17	50

*Run with pressure rise indicating condensation occurred.

†Supersaturations calculated at last pressure tap which did not indicate a pressure increase over the noncondensing gas flow values.

§Final taps in nozzle did not indicate because of possible plugged lines. Values computed for last tap which indicated.

**Runs 16 and 23 omitted since they were unsuccessful attempts at repeat runs.

2. Nozzle Expansion

No further experimental work was done with the nozzle expansion system. Review of the experimental report (NAA-SR-12049) was completed and the report is in process of publication. In review of the report, the point in the nozzle at which supersaturation begins was redefined as that point in the nozzle monitored by the pressure tap upstream of the first pressure tap to show a condensation pressure rise. The data previously presented (in NAA-SR-12193) were based on locating the advent of supersaturation as the intersection of the pressure rise curve and the dry gas expansion curve. The change should result in an improvement because actual data from an existing tap are used instead of data interpolated between measured points. Table 18 summarizes supersaturations attained.

IV. EVALUATION OF EFFORT

The tasks listed below were enumerated in the FY 67 Project Proposals report⁽³⁾ for reactor physics analyses. Tasks 1, 2, 3, 7, and 9 were completed by the end of the report period. Tasks 5 and 8 represent a continual updating and maintenance process. Tasks 4 and 6 are underway. Effort on Task 10 has not yet been initiated.

Tasks for FY 67

- 1) Publish LOAFER code as a topical report.
- 2) Study S_n transport quadrature coefficients.
- 3) Determine SNAP photoneutron source strength from experimental data.
- 4) Evaluate SNAP intermediate neutron energy spectra and select optimum neutron-energy group structure.
- 5) Update nuclear code cross section libraries.
- 6) Add Be-free gas kernel to 100 GT code and publish 100 GT code.
- 7) Develop a space-time flux-synthesis code.
- 8) Maintain existing codes and procure new codes.
- 9) Develop multi-spatial dimensional flux synthesis codes.
- 10) Evaluate feasibility of adapting an improved method for checking analytical calculations with experiment.

The analysis of SNAP experimental photoneutron data was completed. The study showed that the Rotter "core" set of photoneutron decay sets best fit the data. An external document describing the comparison of photoneutron decay sets has been prepared. An external document translation of the original Rotter article⁽²⁾ is being published. Internal documentation of a photoneutron decay code and of the SNAP experimental photoneutron data analysis has also been prepared.

The FY 67 effort towards flux-synthesis development was completed. Related in-house and external developments prompted state-of-the-art review and future

program recommendation. Provisions for FY 68 and 69 space-angle synthesis have been proposed.

Existing nuclear codes were maintained; currently feasible code improvements were initiated. Cross section updating included additions to a new 26-group (13-thermal) library. The importance of zirconium-isotopes and U^{238} Doppler-broadening was shown.

Studies of transport theory variables were completed. FY 67 studies showed definite advantages for Double-Legendre Quadrature Sets in Planar Geometry. Optimization of quadrature order with computer time was presented. The necessary number of significant figures in the quadrature sets was also studied.

The LOAFER code was documented and will be published.

In furtherance of the objectives of the DC conduction pump development task, the NaK-78 corrosion testing of 24 ceramic-braze-metal samples was completed. Metallographic specimens from the tested, as well as the control, samples were prepared, examined and evaluated. Only the niobium-BeO combination provided satisfactory results with all four braze alloy systems after exposure for 500 hr in both 1100 and 1300°F NaK.

All ANISN calculations have been completed in the analysis of STIR heavy metal/lithium hydride shielding experiments. Preliminary tests of the cross section data and order of approximations used throughout the study have indicated favorable comparison with experimental measurements in water and in lithium hydride. Initial QAD results for the primary gamma dose rate have exhibited close agreement with STIR measurements.

The work reported herein completes the program objectives of the composite material shielding program. Neutron flux, neutron dose rate, gamma ray spectra, and primary and secondary gamma dose rates have been measured through shields of lithium hydride, lead, uranium, tungsten, LiH-Pb, LiH-U, LiH-Pb-LiH-Pb, and LiH-U-LiH-U. In many cases special techniques were developed to overcome measurement problems. An extensive library now exists of standard configuration measurements for the evaluation of analytical methods and cross-sections for space reactor system shields.

The post-irradiation examination of selected reactor components, with the exception of metallography of samples from internal reflectors, NaK plumbing and core hold-down springs, was completed. This task did not include responsibility for the evaluation of components performance but was limited to the acquisition of data. Components evaluation and comparison with pre-irradiation performance will be reported under another subaccount.

Progress was made in documentation of the cloud chamber operation and in assembling the mercury chamber.

V. OBJECTIVES FOR THE NEXT REPORTING PERIOD

A. REACTOR PHYSICS ANALYSES

The objectives for the next quarter of the reactor physics analyses efforts are listed below.

- 1) Continue the development of SNAP reactor decay and burnup chain information for use with the transport theory burnup code.
- 2) Continue evaluation of SNAP intermediate neutron energy spectra and selection of an optimum neutron energy group structure.
- 3) Obtain and put into operation the Swedish second-order perturbation codes.
- 4) Document the 26-group (13 thermal) cross section library and its data sources.
- 5) Begin addition of the free gas kernel for Be to 100 GT. (This will supplement the present single phonon model in providing adequate representation at higher temperatures.)

The next step in (4) is the replacement of the 13 Hansen-Roach fast and intermediate cross sections with 23 from the Bondarenko set, yielding 36-group sets. This will provide an up-to-date cross section set, always reducible to fewer groups, which should serve until the Brookhaven and/or AI Evaluated Nuclear Data Files are ready. It should be a couple of years yet before the latter are capable of providing checked-out input cross sections for all isotopes of SNAP interest. The 36-group library is expected to be completed this fiscal year. In the interim, the Hansen-Roach cross sections were previously updated for some of the materials with modifications from LASL and GE-Evendale.

B. DC CONDUCTION PUMP DEVELOPMENT

The test pump will be assembled and put under 500 hr test.

C. BASIC SHIELDING ANALYSIS

The approximate spatial distribution of the neutron flux in shield planes parallel to the plane of the STIR fission plate will be determined. This will

be used in combination with the ANISN results in infinite plane geometry to specify the overall spatial distribution of the secondary gamma source strength throughout the shield volume. The QAD code will then be used to determine the corresponding external dose rate. Comparison of final calculations with STIR measurements will be made and reporting of results completed.

D. BASIC SHIELDING EXPERIMENTS

No further effort in this program is planned for the next report period.

E. BASIC SHIELD MATERIALS TECHNOLOGY

NAA-SR-9400, Vol. III on lithium hydride characteristics, and Vol. IV on aggregate shielding rechnology will be printed.

F. SNAP 10AFS-3 POST-TEST EXAMINATION

Metallography for the components examination will be completed.

Tensile tests of irradiated fuel element cladding and as-received and as-processed non-irradiated cladding will be conducted. The samples will be tested at 1400°F employing an Instron testing machine, and strain will be measured directly with an extensometer. This effort has been delayed by unavailability of tooling and manpower.

Electron microscopy of fuel and cladding replicas will be performed.

G. SUPERSATURATION STUDY

The experimental effort to obtain data on the agreement of theory and experiment for substances with calculated critical nuclei containing less than 50 molecules will continue. The task will include the operation of the mercury chamber and further work in the larger chamber with heptane and the higher hydrocarbons.

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