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(SYSTEMS FOR NUCLEAR AUXILIARY POWER)
GENERAL SUPPORTING TECHNOLOGY
AUGUST-OCTOBER 1965
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AEC Research and Development Report

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SNAP REACTOR,
SNAP PROGRAM
M-3679 (42nd Ed.)

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by GG DTIE, date 5/24/73
Wash DC

PROGRESS REPORT FOR SNAP
(SYSTEMS FOR NUCLEAR AUXILIARY POWER)
GENERAL SUPPORTING TECHNOLOGY
AUGUST-OCTOBER 1965
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By

L. S. MIMS

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The Progress Report for the previous quarter was NAA-SR-11496

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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 ob-

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

II. SUMMARY OF EFFORT

Basic equations were developed for application of the space-time synthesis technique to one-dimensional, transport theory reactor kinetics calculations. The WIGLE code, which treats the same problem by using diffusion theory, was obtained from another installation and adapted for use at Atomic International. It will be used as a reference in developing the transport theory program.

In the effort to provide better thermal spectrum estimates for ZrH-moderated reactors, temperature-dependent thermal scattering cross-section data for hydrogen bound in ZrH were processed and assembled on tape. These data are now being processed into a form suitable for spectral coefficient calculations.

A number of the cross sections and other nuclear parameters currently used in SNAP nuclear calculations were critically examined. Several areas were identified where updating, or correction, is called for.

The effects of temperature and microstructure on the neutron scattering properties of sintered beryllium oxide were evaluated in terms of the resulting effects on the nuclear properties of reactors similar to SNAP 8. In most instances the effects were found to be small or insignificant.

The rate of helium production in SNAP fuel elements was estimated. The greatest production was found to be due to the boron (n, α) reaction in the Hastelloy-N cladding (assumed boron concentration of 50 ppm).

Evaluated cross-section data for 34 isotopes/elements was obtained from the Aldermaston cross-section tape, punched on data cards, and plotted. These data will be used in the forthcoming updating and expansion of the AI Evaluated Nuclear Data File (AIENDF).

The problem of calculating realistic dilute resonance integrals and effective group cross sections for fissionable isotopes, using single level resonance theory, was studied. Results obtained to date indicate that the values obtained using this theory are significantly different from the experimental values. The use of a multi-level formalism is being investigated.

The sensitivity of the calculated value of the dilute resonance integral of Pu^{239} to the choice of cadmium cutoff energy (about 0.5 ev) was investigated. It was found that the value of the integral is very sensitive to the choice of cutoff energy, primarily because of the large Pu^{239} resonance at 0.3 ev.

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Elastic scattering angular data was compiled for 18 isotopes/elements in preparation for updating and expansion of the AIENDF.

Improvements were made to the ADT, AROCS and MALINDA cross-section processing codes.

The remaining problems associated with combining the GRAVE, TRIXIE and PRISM cross-section codes into the GRISM cross-section averaging code were resolved, and satisfactory operation was obtained.

The Tape Conversion Program was modified to permit its use with the Los Alamos evaluated nuclear data file.

A computer program to generate Doppler-broadened point cross-section data in the resonance region was written and checked out.

Three additional digital subroutines were prepared for use with the UPSTART-II digital system simulation program. These additional subroutines permit detailed simulation of a three-dimensional transient thermal model of a SNAP reactor core, including fuel and cladding eccentricities.

Operation of the SCA-5 critical assembly with the small core was resumed. The effectiveness of different types of control drum poison backing materials was measured. Interaction effects between adjacent poison-backed control drums were also determined. Approximately 5900 additional unpoisoned fuel pellets were manufactured for the SCA-5.

The DTF-II and 14-0 codes were used to estimate gamma attenuation in depleted uranium for a configuration similar to that employed at the Shield Test and Irradiation Reactor (STIR). The calculated value of the relaxation length for fission-produced primary gammas was in excellent agreement with the experimental value subsequently obtained at the STIR. Much poorer agreement was obtained for the calculated value

of the relaxation length for secondary gammas produced by neutron scattering and capture.

Further experiments and analyses were performed to clarify the cause of anomalous gamma attenuation behavior observed at the STIR. As a result of these investigations, it was concluded that the anomalous behavior was due to secondary gammas produced by neutron interaction with the collimator structure. These effects were essentially eliminated by placing a 12-in. slab of lithium hydride in front of the collimator. This slab was utilized to obtain the gamma relaxation length values referred to above.

The O5R Monte Carlo program was used to calculate neutron attenuation in lithium hydride for a configuration previously employed at the STIR. The calculations gave results which were consistently lower than the experimental values by as much as a factor of two. Means for improving the calculations are being investigated.

A meeting was held with ORNL personnel to coordinate the analytical and experimental shielding programs being performed in connection with the planned SNAP/Tower Shield Facility experiments.

Modifications and improvements were made to the CLUB and SCAR shielding codes. The 21-group cross-section library prepared for DTF-II shielding calculations was partially checked out, and four-group cross-section data for use with the SLAP code were obtained from ORNL. The GAMLEG code was obtained from LASL and adapted to the NAA computer system. It will be used to process gamma ray cross-section data into a format compatible with DTF-II.

Experimentally derived neutron spectra were used to analyze the neutron damage suffered by silicon semiconductors exposed to these spectra. The results were in good agreement with theoretical predictions of damage variations due to spectral effects.

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A draft report was prepared summarizing progress to date in applying electrical conductivity noise analysis to the accurate measurement of velocity profiles and fluid crossflow in hydraulic test models.

The diffusion chamber which was constructed at the NAA Science Center to investigate supersaturation phenomena was placed in operation. Several data points were obtained for the critical supersaturation of methanol as a function of temperature.

As a result of difficulties experienced in efforts to obtain a satisfactory transparent nozzle for use in mercury supersaturation studies, this approach was postponed. Instead, a metal nozzle instrumented with pressure taps and thermocouples will be utilized. Such a nozzle was successfully fabricated and assembly of the test loop was initiated.

Measurements of hydrogen permeation rate as a function of hydrogen pressure in a NaK-containing system were completed for a temperature of 1320°F. Similar measurements were initiated and completed for a test temperature of 1260°F. Preparations were also made for the next series of tests to be conducted at 1175°F. The results of the tests to date indicate that in this temperature and pressure regime, hydrogen permeation and hydrogen concentration in NaK are proportional to the hydrogen pressure to the first power.

A topical report was prepared summarizing the results of the investigation of niobium-lined, stainless steel tubing.

A literature survey of various techniques for liquid metal corrosion testing was completed. The form of the final report on this subject was outlined.

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III. AREAS OF EFFORT

A. NUCLEAR AND CONTROL ANALYSIS

1. Reactor Theory and Method Development

a. Space-Time Synthesis Application of Variational Technique

The development of the variational technique described in the preceding quarterly report involved the application of the technique to a two-dimensional, R-Z, synthesis. A similar development is possible in connection with space-time synthesis — the ultimate object being an efficient, one-dimensional, transport theory, reactor kinetics computational tool, as contrasted with the simple point-kinetics representation now widely employed. The value of such a development to the SNAP program is readily evident when one considers the different neutron kinetic mechanisms at work in the core and in the beryllium reflector of a SNAP reactor. Moreover, the importance of a space-dependent kinetics code will be even greater as consideration is given to 4- π shielded configurations. Because of the immediate importance of this work to the Aerospace Nuclear Safety Program, the current

effort is directed toward development of a satisfactory space-time synthesis code and is jointly supported by that program and the General Supporting Technology Program. Much of the work, however, will also contribute to the development of an efficient two-dimensional criticality program. Such a development will be undertaken upon completion of the space-time synthesis.

(1) Space-Time Basic Equations

Given the time-dependent neutron transport equation, the functional to use as the basis for a variational principle is a matter of choice.⁽¹⁾ Further study has made possible the selection of a more conveniently used functional for the variational space-time flux synthesis than that given in the preceding quarterly report. As has been pointed out in the literature,^(1,2) with this new functional, it is not necessary to place an arbitrary constraint directly on the flux at the end of the pulse. Rather, the pulse-end adjoint is constrained.

The time-dependent neutron transport equations can be written as:

$$L\phi(r, \Omega, E, t) + \sum_{i=1}^6 \lambda_i C_i(r, t) f_i(E) = \frac{1}{v} \cdot \frac{\partial \phi}{\partial t} \quad \dots (1)$$

$$M_i \phi(r, \Omega, E, t) - \lambda_i C_i(r, t) = \frac{\partial C_i}{\partial t} \quad i=1, 2, \dots, 6 \quad \dots (2)$$

where

$$L\phi \equiv -\Omega \cdot \nabla \phi - \Sigma_T(r, E, t)\phi + \int_{\Omega'} \int_{E'} \left[\Sigma_S(r, \Omega, E, \Omega', E') + \frac{\chi(E)}{4\pi} \nu \right. \\ \left. \cdot (1 - \beta) \Sigma_f(r, E', t) \right] \phi(r, \Omega', E', t) d\Omega' dE' \quad \dots (3)$$

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and

$$M_i \phi \equiv \beta_i \int_{\Omega} \int_E \nu \Sigma_f(r, E, t) \phi(r, \Omega, E, t) d\Omega dE \quad \dots (4)$$

with the usual meanings for the rest of the symbols.

The flux ϕ and the delayed neutron precursors C_i satisfy the initial conditions:

$$\phi(r, \Omega, E, 0) = \phi_0(r, \Omega, E) \quad \dots (5)$$

$$C_i(r, 0) = C_{i0}(r) \quad i=1, 2, \dots, 6 \quad \dots (6)$$

The functional chosen is:

$$\begin{aligned} I\{\phi, C_i, \phi^*, C_i^*\} \equiv & \int_0^{t_f} \int_r \int_{\Omega} \int_E \phi^*(r, \Omega, E, t) \left[L\phi + \sum_{i=1}^6 \lambda_i C_i \beta_i - \frac{1}{v} \frac{\partial \phi}{\partial t} \right] dr d\Omega dE dt + \sum_{i=1}^6 \int_0^{t_f} \int_r C_i^*(r, t) \left[M_i \phi \right. \\ & \left. - \lambda_i C_i - \frac{\partial C_i}{\partial t} \right] dr dt + \int_r \int_{\Omega} \int_E \frac{1}{v} \phi(r, \Omega, E, t_f) \phi^*(r, \Omega, E, t_f) dr d\Omega dE + \sum_{i=1}^6 \int_r C_i(r, t_f) C_i^*(r, t_f) dr \quad \dots (7) \end{aligned}$$

If the function $\phi(r, \Omega, E, t)$ and its adjoint, $\phi^*(r, \Omega, E, t)$ satisfy the boundary conditions:

$$\phi(R, \Omega, E, t) = 0 \quad \text{for} \quad \vec{\Omega} \cdot \vec{\eta} < 0$$

$$\phi^*(R, \Omega, E, t) = 0 \quad \text{for} \quad \vec{\Omega} \cdot \vec{\eta} > 0 \quad \dots (8)$$

Where R is a vector describing the reactor external surface and $\vec{\eta}$ is the outward pointing normal to that surface; it can be shown that an equivalent form of Equation 7 is:

$$\begin{aligned} I\{\phi, C_i, \phi^*, C_i^*\} \equiv & \int_0^{t_f} \int_r \int_{\Omega} \int_E \phi(r, \Omega, E, t) \left[L^* \phi^* + \sum_{i=1}^6 \beta_i \nu \Sigma_f C_i^* + \frac{1}{v} \frac{\partial \phi^*}{\partial t} \right] dr d\Omega dE dt \\ & + \sum_{i=1}^6 \int_0^{t_f} \int_r C_i(r, t) \left[M_i^* \phi^* - \lambda_i C_i^* + \frac{\partial C_i^*}{\partial t} \right] dr dt + \int_r \int_{\Omega} \int_E \frac{1}{v} \phi(r, \Omega, E, 0) \phi^*(r, \Omega, E, 0) dr d\Omega dE \\ & + \sum_{i=1}^6 \int_r C_i(r, 0) C_i^*(r, 0) dr \quad \dots (9) \end{aligned}$$

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where

$$L^* \varphi^* \equiv \Omega \cdot \nabla \varphi^* - \Sigma_t \varphi^* + \int_{\Omega'} \int_E \left[\Sigma_s(r, \Omega', E', \Omega, E) + \frac{\chi(E')}{4\pi} (1 - \beta) \nu \Sigma_f(r, E, t) \right] \varphi^*(r, \Omega', E', t) d\Omega' dE' \quad \dots (10)$$

$$M_1^* \varphi^* \equiv \lambda_1 \int_{\Omega} \int_E f_1(E) \varphi^*(r, \Omega, E, t) dE d\Omega \quad \dots (11)$$

The first variation of I with respect to φ^* and C_1^* is easily obtained using the form of Equation 7. The conditions for the first variation to be zero are precisely the Equations 1 and 2, plus the following final conditions:

$$\left. \begin{aligned} \varphi^*(r, \Omega, E, t_f) &= \varphi_f^*(r, \Omega, E) \\ C_1^*(r, t_f) &= C_{1f}^*(r) \end{aligned} \right\} \begin{array}{l} \text{prescribed functions of } r, \Omega, E \\ \dots (12) \end{array}$$

In the linear case, that is, in those cases where the parameters of the system are not implicit functions of the flux in the system (or some function of it, like the energy released) the variation of I with respect to φ and C_1 will vanish when

$$L^* \varphi^* + \sum_{i=1}^6 \beta_i \nu \Sigma_f C_{1i}^* = -\frac{1}{\nu} \frac{\partial \varphi^*}{\partial t} \quad \dots (13)$$

$$M_1^* \varphi^* - \lambda_1 C_1^* = -\frac{\partial C_1^*}{\partial t} \quad i = 1, 2, \dots, 6 \quad \dots (14)$$

and the initial conditions, 5 and 6, are satisfied. The variation of I with respect to φ and C_1 has been performed using Equation 9. Nonlinear cases, for example when cross sections are functions of temperature, would result in a more complex set of equations to be satisfied by φ^* and C_1^* , but no change in the equations for φ or C_1 .

(2) Variational Approximation

To obtain approximate solutions for $\varphi(r, \Omega, E, t)$, using the variational techniques, it is assumed that φ can be expressed in the following way:

$$\varphi(r, \Omega, E, t) \equiv \begin{bmatrix} \varphi_1(r, \Omega, t) \\ \varphi_2(r, \Omega, t) \\ \vdots \\ \varphi_g(r, \Omega, t) \\ \vdots \\ \varphi_G(r, \Omega, t) \end{bmatrix} \quad \dots (15)$$

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and

$$\varphi_g(r, \Omega, t) = \sum_{j=1}^K N_g^j(r, \Omega) T_g^j(t) \quad g = 1, 2, \dots, G \quad \dots (16)$$

where G is the number of energy groups and K the number of trial functions used. K need not be the same value for all groups.

In the treatment that follows, the delayed neutron precursors will not be included. The formalism is somewhat simpler and the resulting equations are appropriate to treat large insertions of reactivity.

For the "adjoint flux" (φ^*) a similar expression is assumed:

$$\varphi^*(r, \Omega, E, t) \equiv [\varphi_1^*(r, \Omega, t); \varphi_2^*(r, \Omega, t); \dots, \varphi_g^*(r, \Omega, t); \dots, \varphi_G^*(r, \Omega, t)] \quad \dots (17)$$

and

$$\varphi_g^*(r, \Omega, t) = \sum_{i=1}^K N_g^{i*}(r, \Omega) T_g^{i*}(t) \quad g = 1, 2, \dots, G \quad \dots (18)$$

The functions $N_g^j(r, \Omega)$ and $N_g^{i*}(r, \Omega)$ are the so called "trial functions." They are approximate solutions of the problem — for example, solutions for the reactor equations in the stationary state or asymptotic solutions corresponding to different states of the reactor.

The functions $T_g^j(t)$ and $T_g^{i*}(t)$ are the unknown "mixing functions," that have to be determined.

Introducing Equations 15, 16, 17, and 18 into the form of Equation 7 of the functional, and calling:

$$\int_r \int_{\Omega} N_g^{i*}(r, \Omega) \Omega \cdot \nabla N_g^j(r, \Omega) dr d\Omega \equiv L_g^{ij} \quad \dots (19)$$

$$\int_r \int_{\Omega} N_g^{i*}(r, \Omega) \Sigma_{Tg}(r, t) N_g^j(r, \Omega) dr d\Omega \equiv A_g^{ij}(t) \quad \dots (20)$$

$$\int_r \int_{\Omega} \int_{\Omega'} N_g^{i*}(r, \Omega) \Sigma_{gg'}(r, t) N_g^j(r, \Omega') dr d\Omega d\Omega' \equiv S_{gg'}^{ij}(t) \quad \dots (21)$$

$$\int_r \int_{\Omega} N_g^{i*}(r, \Omega) N_g^j(r, \Omega) dr d\Omega \equiv I_g^{ij} \quad \dots (22)$$

where Σ_{Tg} and $\Sigma_{gg'}$ are the total and transfer group cross sections, weighted both with a flux and an adjoint spectrum; the functional takes the form:

$$I\{T_g^j, T_g^{i*}\} \equiv \sum_{g=1}^G \sum_{i=1}^K \sum_{j=1}^K \left\langle \int_0^{t_f} \left[-L_g^{ij} - A_g^{ij}(t) \right] T_g^{i*}(t) T_g^j(t) + \sum_{g'=1}^G \left[S_{gg'}^{ij}(t) T_g^{i*} T_{g'}^j \right] \right. \\ \left. - \frac{1}{v} I_g^{ij} T_g^{i*} \frac{dT_g^j}{dt} \right\rangle dt + \frac{1}{v} I_g^{ij} T_g^{i*}(t_f) T_g^j(t_f) \quad \dots (23)$$

The first variation of Equation 23 with respect to the T_g^1 's vanishes if the following set of coupled differential equations are satisfied:

$$\sum_{j=1}^K \left\{ \left[-L_g^{1j} - A_g^{1j}(t) \right] T_g^j(t) + \sum_{g'=1}^G \left[S_{gg'}^{1j}(t) T_{g'}^j \right] - \frac{I_g^{1j}}{v_g} \cdot \frac{dT_g^j}{dt} \right\} = 0 \quad i = 1, 2, \dots, K \quad \dots (24)$$

The following system of differential equations would be obtained for $T_g^1(t)$:

$$\sum_{i=1}^K \left\{ \left[L_g^{1j} - A_g^{1j}(t) \right] T_g^1(t) + \sum_{g'=1}^G \left[S_{gg'}^{1j}(t) T_{g'}^1 \right] + \frac{I_g^{1j}}{v_g} \frac{dT_g^1}{dt} \right\} = 0 \quad j = 1, 2, \dots, K \quad \dots (25)$$

in the linear case (no feedback), through the manipulation of Equation 9. The functions $T_g^j(t)$ satisfy the following initial conditions:

$$\begin{aligned} T_g^1(0) &= 1 & g &= 1, 2, \dots, G \\ T_g^j(0) &= 0 & j &= 2, 3, \dots, K \end{aligned} \quad \dots (26)$$

and the functions $T_g^1(t)$ satisfy the following final conditions

$$\begin{aligned} T_g^G(t_f) &= 1 & g &= 1, 2, \dots, G \\ T_g^1(t_f) &= 0 & i &= 1, 2, \dots, (K-1) \end{aligned} \quad \dots (27)$$

(3) Applications

To actually carry out a numerical application it is necessary to

a) Generate a set of trial functions $N_g^1(r, \Omega)$ and $N_g^j(r, \Omega)$.

Using a code such as DTF, this would mean the generation of sets of values

$$\left. \begin{array}{l} N_g^1 \ell \alpha \\ N_g^j \ell \alpha \end{array} \right\} \quad \text{i. e., the values of } N_g^1 \text{ and } N_g^j \text{ at mesh point } \ell \text{ and in the angular direction } \alpha.$$

b) Using these values and the definitions in Equations 19 to 22, calculate the coefficients for the differential equations in Equations 24 and 25. For example, the coefficient I_g^{1j} of definition in Equation 22 will be given by:

$$I_g^{1j} = \sum_{\ell} \left[\left(\sum_{\alpha} N_{g\ell\alpha}^1 N_{g\ell\alpha}^j \omega_{\alpha} \right) \Delta \ell \right] \quad \dots (28)$$

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where ω_α is the weight given in DTF to the α direction and Δl is the volume element appropriate to the geometry used. Similar expressions can be written for L_g^{ij} , $A_g^{ij}(t)$, $S_{gg}^{ij}(t)$.

c) Solve the system of first order differential Equation 24 with initial conditions of Equation 26.

d) Reconstruct the flux as a function of space and time using Equations 15 and 16, and with it, calculate the reactor characteristics that are desired.

Present efforts in this area are directed towards completion of steps b), c), and d).

The best way to judge the accuracy of the variational approximation is to compare its results with those from a so-called "exact solution" — usually a solution of the finite difference approximation to the problem. Lacking some "exact" reference calculation, another method of evaluation is to add more trial functions and observe the changes in the solutions, in the hope of achieving some sort of convergence. The choice of the type and number of trial functions for a particular problem is necessarily a trial and error process.

For the case of two-group diffusion theory in slab geometry with cross sections varying linearly with time, a comparison between exact calculations and synthesis has been reported. Yasinsky and Henry⁽³⁾ have shown that for a large step insertion, followed by a linear decrease of reactivity, the synthesis method proved quite reliable even with as little as two trial functions. They used WIGLE, a two-group diffusion theory space-time code as the comparison standard. It would be desirable to carry out a similar comparison with the inclusion of a simple temperature feedback in both the synthesis calculation and WIGLE. This possibility is now under study.

A finite difference integration of the space-time transport equation would provide a standard to check the results obtained with the synthesis technique in the more general formulation. The possibility of using the S_n time dependent integration of the transport equation developed at Los Alamos for this purpose is being studied.

b. Zirconium Hydride Thermalization

Thermal scattering cross sections representing thermalization by hydrogen bound in zirconium hydride at temperatures from 68 to 1800°F were assembled on tape. These data are now being processed into a form suitable for spectral coefficient calculations and for inclusion into the DTF cross-section library.

c. Nuclear Parameters and Data for Use With the DTF and Other Nuclear Codes

Comparison of the Hansen-Roach cross sections^(4,5) and other nuclear parameters with more recent data which are currently being used in SNAP calculations revealed a number of important discrepancies. The significance of these discrepancies in SNAP nuclear calculations is usually minimized by the normalization of results to experiments. The discrepancies become important, however, when calculations are made for systems which are somewhat different and for which there are no experimental normalization data.

One major discrepancy which was discovered was the failure to include the 0.9-barn zirconium absorption resonance integral in the Hansen-Roach library. Inclusion of this integral in the library was found to decrease the calculated reactivity of a SNAP 8 reactor by 2.7% $\Delta k/k$. The important effect of this resonance is due to its presence directly above the U^{235} fission resonances.

Two sets of U^{235} cross sections are presently available in the Hansen-Roach cross-section library.⁽⁵⁾ One set is based on a U^{235}

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fission resonance integral of 319 barns, while the other set is based on a value of 294 barns. The latter value is in better agreement with recent measurements.⁽⁶⁾ Only the set based on the 319 barn value, however, has been modified to account for ZrH moderation in the thermal groups, and hence, it is this set which is generally used for SNAP nuclear calculations. Both of these factors, together with the 0.9 barn zirconium absorption resonance mentioned above, will be taken into account in the preparation of an updated cross-section library for general use in SNAP nuclear calculations. Work on the updated library is scheduled to begin early in the next report period.

The Hansen-Roach representation of hydrogen anisotropic scattering (for the P_1 scattering approximation) involves the use of negative absorption cross sections. The effect of this representation on the operation of the DTF-II criticality code was examined. It was determined that the use of the negative cross sections has no effect on the results of the criticality calculation, but does interfere with a summary neutron accounting process. This interference, in turn, results in errors in the neutron balance tables which are printed as an auxiliary output of the code. This interference is considered to be of small importance relative to the other benefits derivable from the arrangement. Accordingly, no further effort on this effect is planned. The findings of this investigation will be incorporated in the documentation of the updated cross-section library mentioned above.

The $\text{Be}(n, 2n)$ reaction occurring in the reflector of a SNAP 8 reactor is estimated to contribute about 2.6% $\Delta k/k$ to the reactivity of the system. A competing nuclear reaction at fast neutron energies is the $\text{Be}(n, \alpha)$ reaction. A refined analysis of beryllium reactivity effects was performed. In this analysis the inclusion of additional $\text{Be}(n, 2n)$ secondary-neutron-

spectrum detail and the updating of the $\text{Be}(n, \alpha)$ and $\text{Be}(n, 2n)$ reaction cross sections produced a net reactivity loss of 0.66% $\Delta k/k$.

Examination of Hansen-Roach group values for the fission neutron spectra⁽⁴⁾ reveals that they are in appreciable disagreement with those obtained from other sources. Table 1 compares the Hansen-Roach values with those obtained from the compilation in the Argonne handbook, "Reactor Physics Constants"⁽⁷⁾ and from the Watt⁽⁸⁾ formula

$$\chi(E) = 0.93544 \left[\frac{2\pi}{e} \right]^{\frac{1}{2}} \exp -1.0363E \sinh(2.29E)^{\frac{1}{2}}$$

TABLE 1
COMPARISON OF HANSEN-ROACH,
WATT AND ANL 5800 FISSION
NEUTRON SPECTRA

Neutron Energy Group	Hansen-Roach	Watt	ANL 5800
1	0.204	0.2076	0.2082
2	0.344	0.3502	0.3501
3	0.168	0.1668	0.1668
4	0.180	0.1762	0.1761
5	0.090	0.0851	0.0849
6	0.014	0.0130	} 0.0139
7	0.0	0.0010	
8	0.0	0.0001	
9-16	0.0	0.0	0.0

The faster spectra of Watt and of the Argonne compilation would reduce the reactivity of a reactor. Since the Hansen-Roach group constants are probably averaged over their fission spectrum, use of a different spectrum with their cross sections will present a problem. Insofar as possible, however, better data on the fission spectrum and updated cross sections for the beryllium reactions will be included in the updated DTF-II cross-section library.

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d. The Effect of Temperature and Microstructure on the Neutron Scattering Cross Section of Sintered Beryllium Oxide

Small BeO fillers are used in the SNAP 8 reactor between the fuel array and the core vessel. Beryllium oxide has also been considered as an alternative reflector material. Figure 1 shows the effect of temperature on the neutron scattering cross section of sintered beryllium oxide. The work of Zhezherun, et al.⁽⁹⁾ evaluated the effects of method of fabrication and grain size on the scattering cross section of sintered BeO for a Maxwellian spectrum. Using these data, the corresponding reactivity effects were estimated by evaluating albedo changes. The estimates are conservative since the cross-section changes would be less for the SNAP 8 spectrum, which is harder than the Maxwellian spectrum used in the analysis.

Figure 1. The Effect of Temperature on the Neutron Scattering Cross Section of Sintered Beryllium Oxide (Reference 9)

Assuming a SNAP 8 reactor with a 5-in.-thick BeO reflector, the maximum possible reactivity effect due to changes in the BeO scattering cross section would be a gain of 24¢. The corresponding maximum contribution to the temperature coefficient would be +2.4¢ per 100° F, constant over the range 68 to 981° F. The maximum possible reactivity variation due to method of fabrication

and grain size would be 3.5 and 17¢, respectively, over the same temperature range.

Based on the foregoing, it is concluded that, with respect to the BeO filler pieces in a SNAP 8 reactor, temperature variations, method of BeO fabrication, and grain size would have a negligible effect on system reactivity. The effects, however, could become marginally significant in a SNAP reactor with a thick BeO reflector.

e. Helium Production in SNAP Fuel Elements

Using the SNAP 8 reactor as a model, the helium production in SNAP fuel elements was calculated. The results are given in Table 2.

2. Cross Section Development

a. Basic Cross-Section Data

Evaluated data from the Aldermaston Data File were retrieved, punched on data cards and plotted. These data will be used in the forthcoming updating and expansion of the AI Evaluated Nuclear Data File (AIENDF). The elements taken from the Aldermaston data tape include:

Tritium	Fluorine	Calcium	Th ²³²
Helium 3, 4	Aluminum	Copper	U ²³³
Lithium 6, 7	Silicon	Cadmium	U ²³⁴
Beryllium	Sodium	Titanium	U ²³⁶
Boron, 10	Chlorine	Zirconium	Pu ²³⁹
Carbon	Chromium	Gallium	Pu ²⁴⁰
Nitrogen	Iron	Xenon	Pu ²⁴¹
Oxygen	Nickel	Lead	

The problem of calculating realistic dilute resonance integrals and effective group cross sections for fissionable isotopes using single level resonance theory is being studied. It is planned to apply an empirical correction to the single level results by adding a suitable smooth background capture and fission cross section which produces agreement with measured values

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TABLE 2
CALCULATED HELIUM PRODUCTION IN SNAP FUEL ELEMENTS

Material	Mechanism	Production Rate (atoms/cm ³ of material/watt-hr of reactor operation)
Fuel	Ternary fission	2.43×10^4
Fuel	Boron (n, α), 2.64 ppm B	0.80×10^4
Hastelloy-N Cladding	Nickel (n, α), may vary $\pm 10\%$	9.80×10^4
Hastelloy-N Cladding	Boron (n, α), 50 ppm B	22.0×10^4

of dilute resonance integrals and cross sections. The current analysis involves the splitting of the resonance region into several energy groups for which resonance integrals calculated with the TRIX code and obtained from studies at Karlsruhe⁽¹⁰⁾ are compared with integrals over the measured fission and capture cross sections. An extensive analysis for U²³⁵ is nearing completion, and some of the results obtained to date are presented in Tables 3 and 4.

Present evidence indicates that resonance integral calculations based on a multilevel formalism tend to minimize the discrepancy between calculated and measured values. Efforts, as yet unsuccessful, are being made to obtain the series of multilevel Fortran programs developed.^(11, 12)

The sensitivity of the calculated dilute resonance integral of Pu²³⁹ to the choice of cadmium cutoff energy (about 0.5 ev) was investigated. The large Pu²³⁹ resonance at 0.3 ev makes a large contribution to the resonance integral, and hence, the calculated value is very sensitive to the choice of cutoff energy. Since the Doppler-broadened width for this resonance is comparable to its natural width, the use of the natural line shape to calculate the dependence of the resonance integral on the cutoff energy introduces an error of only a few percent in the

result. Using the natural line shape, the contribution of the 0.3 ev resonance to the fission resonance integral over the energy interval from 0.45 to 0.50 ev was found to be 44 barns. (The corresponding quantity for U²³⁵ is given by J. J. Schmidt of Karlsruhe as 9.4 barns.) The experimental value for the Pu²³⁹ fission resonance integral above 0.5 ev is quoted as about 327 ± 22 barns. It is evident that meaningful comparison between the experimental and calculated resonance integral values for Pu²³⁹ is strongly dependent on careful evaluation of the cutoff energy of the experiment. In particular, use of this experimental resonance integral value to assess the accuracy of the calculated contribution of resonances located above the cutoff energy appears to be difficult. The use of high resolution measurements of point values of the cross section may be more reliable for this

TABLE 3
CALCULATED FISSION AND CAPTURE
RESONANCE INTEGRALS FOR U²³⁵

Energy Range (ev)	Fission		Capture	
	TRIX (barns)	Karlsruhe (barns)	TRIX (barns)	Karlsruhe (barns)
0.5 to 10 ⁵	238.7	265.7	185.3	169.4
1.0 to 10 ⁵	199.0	223.2	178.4	163.0
60.0 to 10 ⁵	66.5	67.6	38.6	32.6
0.5 to 10 ⁶	241.9	268.7	185.9	171.4

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TABLE 4
MEASURED FISSION AND CAPTURE RESONANCE INTEGRALS FOR U²³⁵
(Cutoff energy is 0.5 ev)

Experimenter	Reference	Resonance Integral Fission (barns)	Resonance Integral Capture (barns)
E. Hellstrand	AE-181 (1965)	278 ± 9	-
J. Hardy et al.	NSE 9, 341 (1961)	274 ± 11	-
Baumann	DP-817 (1963)	276 ± 9	-
E. D. Clayton	AECD-4167 (1955)	269 ± 25	-
C. B. Bigham	EANDC-21L (1964)	263 ± 8	-
F. Feiner et al.	Trans. ANS 7, 78 (1964)	292 ± 18	140 ± 10
H. D. Conway et al.	Trans. ANS 7, 78 (1964)	288 ± 18	158 ± 10

purpose unless the effective cutoff energy of the resonance integral experiment can be determined very accurately.

A large amount of elastic scattering angular data was compiled in preparation for updating and expansion of the AIENDF. In all cases, the differential scattering data were converted into Legendre expansions of order nine or higher. The Legendre coefficients were then used to reconstitute the cross section. The following nuclides have been successfully analyzed by the above process:

Tritium	Fluorine	Copper	Tantalum
Helium	Magnesium	Zinc	Tungsten
Lithium 6, 7	Silicon	Niobium	Lead
Nitrogen	Calcium	Cadmium	Titanium

The initial data were taken primarily from BNL-400;⁽¹³⁾ however, more recent measured information, as well as theoretical calculations, are being used to fill remaining gaps. Documentation of data sources and methods of evaluation and compilation is underway.

b. Cross Section Processing

A number of IBM 7094 computer programs which enhance the automation of the AI cross-section project were written, or modified, and

checked out. These include ADT, a program which maintains, updates, and corrects the AI angular data tape; AROCS, a code to average rapidly oscillating cross sections (e.g., the sodium data of Hibdon) and thus allows the use of an average cross section on the coarser AIENDF energy grid; and the MALINDA code, which performs a number of routine tasks such as rearranging formats for point data, interpolation, etc. The MALINDA code was improved by simplification of input requirements and the addition of log-log and log-linear interpolation subroutines.

The combining of GRAVE, TRIXIE, and PRISM into the comprehensive program GRISM was completed. Problems of intercode compatibility were resolved, and initial runs of the GRAVE-TRIXIE combination have been satisfactory. The addition of CONTRACT, which provides data print-out in the specified format, will permit this package of programs to generate multigroup libraries from start to finish in minimum time with maximum efficiency and reliability. Check-out of this package will begin very shortly.

The tape conversion program, TCP, which permits intercomparison between the Aldermaston and Atomic International Evaluated

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Nuclear Data Files, was modified to provide the capability to compare the AIENDF with the Los Alamos evaluated nuclear data files as well. Further extension of this program to permit comparison of the AI data files with those of the Livermore Radiation Laboratory and Brookhaven National Laboratory is currently underway.

A computer code to generate Doppler-broadened point cross-section data in the resonance region was written and checked out. This program has been used to generate room temperature data for U^{235} and U^{238} based on the zero degree Kelvin Karlsruhe⁽¹⁴⁾ data. The theory and programming of this code is in the process of being documented.

3. Reactor Control and Stability Analysis

The capability of the digital program UPSTART-II was expanded by the development of additional subroutines which allow simulation of a three-dimensional transient thermal model of a SNAP reactor core. Included in the simulation are time- and space-variable power and coolant flow profiles and space-dependent fuel and cladding eccentricities. The temperature dependence of the fuel specific heat, as well as the variation of hydrogen conductivity, are included in the thermal model.

The code can be used to evaluate the detailed temperature distribution within a SNAP core during transients involving core power excursions and/or time-dependent variations of core coolant inlet temperature and flow rate. The effect of fuel-to-cladding eccentricity on local hot spots can be evaluated. Since the flow distribution within the core can be represented as a function of geometric location, as well as fuel and cladding position, the effect of fuel rod bowing can be simulated and evaluated. The code can be used to obtain data necessary for the evaluation of transient thermal stresses within the core.

In order to make the digital program as general as possible, the core simulation was made using three separate subroutines. One subroutine contains the time-dependent solution of the heat transfer within each of the coolant channels. The heat transfer process within a fuel rod is calculated within another subroutine. All the heat transfer is assumed to be radial, with axial conduction within a fuel rod neglected. Each fuel rod is divided into as many as 10 axial nodes. Each of these nodes is connected by thermal conductances to six surrounding coolant channels. The heart of the program is the third subroutine, which supplies the geometric information that couples the individual fuel rods to the appropriate coolant channels. It further calculates the power distribution within each fuel rod, accounts for fuel eccentricity, both axially and radially within each fuel rod, provides data for coolant crossflow, calculates the proper thermal conductances between nodes as a function of spatial location and temperature, and provides all input and output data for the code. The subroutine is quite general and can be easily modified to represent different hexagonal arrays by changing only the geometric pattern.

In the present form of these subroutines, only the SNAP 8 core can be simulated, however, with very simple modifications, any hexagonal array core may be simulated. A report describing this thermal model and digital code was prepared.

B. NUCLEAR EXPERIMENTS

1. Basic Control Experiments

The SCA-5 critical assembly was returned to operational status on September 2, 1965. The 9-in. core went critical with a loading of 331 rods with 29 unpoisoned pellets in each rod. Experiments were begun to study the effects of poison backing and drum interaction on control

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TABLE 5
SMALL CORE SCA-5 MEASURED CONTROL
DRUM WORTHS FOR VARIOUS
POISON BACKINGS

Drum Number 1 Backing	Drum Number 1 Worth (\$)	Drum Number 3 Backing	Drum Number 3 Worth (\$)
Boron Carbide	2.35	Boron Steel	2.00
Samarium Oxide (1.2 g/cm ³)	2.00	Boron Steel	2.00
45% Sm ₂ O ₃	2.00	Boron Steel	2.00
45% Gd ₂ O ₃			
10% other oxides (2.7 g/cm ³)

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Figure 3. Relative Reactivity Worths of
Poisoned Fuel vs Poison Concentration

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Figure 2. Top View of Small Core SCA-5
Showing Poison-Backed Control
Drums

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Figure 4. Reactivity Loss for Poisoned
Fuel vs Radial Position

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drum worth. The control worth achieved with various poison materials is shown in Table 5.

The 9-in. SCA-5 contains three control drums arranged about the surface of one-third of the core in a closely-coupled configuration, Figure 2. Using a backing material of boron containing stainless steel on all three drums, the effect of control drum interaction was studied by calibrating adjacent drums. All drums rotate clockwise when viewed from above (as shown in Figure 2) to increase reactivity. The following results were obtained:

1) With Drum Number 3 in its most re-active position, Drum Number 1 was turned in (increasing reactivity) and Drum Number 2 was turned out. The measured drum worths were:

Drum Number 1: \$1.68

Drum Number 2: \$1.02

2) With Drum Number 1 in its most re-active position, Drum Number 3 was turned in (increasing reactivity) and Drum Number 2 was turned out. The results were found to be:

Drum Number 3: \$1.84

Drum Number 2: \$1.22

These results, when compared with the worth of \$2.00 obtained for calibration of Drum Number 1 against Number 3 and assuming Drum Number 2 would exhibit a similar reactivity worth, show very strong interaction effects, ranging up to 50% reduction in reactivity.

2. Basic Core Configuration Experiments

The reactivity worth of fuel material to which varying concentrations of hafnium hydride and europium oxide had been added was measured in the small SCA-5 core. The fuel contained from 1 to 6 wt % HfH_2 and 0.1 to 0.6 wt % Eu_2O_3 . The reactivity worth of the poisoned fuel normalized to its worth with respect to seven cen-

tral unpoisoned rods is given in Figure 3. The curves show a significant amount of self-shielding initially, followed by an almost constant concentration coefficient.

The reactivity effects were also measured at various radial positions using a single rod. Figure 4 shows the radial dependence for two representative samples of europium and hafnium. Other concentrations exhibited similar behavior.

C. SHIELDING ANALYSIS

1. STIR Experiments Planning and Analysis

a. Gamma Attenuation in Depleted Uranium

Gamma attenuation in depleted uranium (0.20 wt % U^{235}) slabs using a U^{235} fission plate source (6.8 wt % U^{238}) was analyzed in an effort to duplicate experimental results from the Shield Test and Irradiation Reactor (STIR) facility. The test geometry was assumed as shown in Figure 5. It was further assumed that the detector, located 276 cm from the fission plate, "sees" 22.6 in. of the 24-in.-diameter fission plate, which is equivalent to a half angle of 5.9 degrees. The DTF-II and 14-0 codes were used for the analysis.

The results are summarized in Figure 6. The gamma dose rate from the fission plate and the dose rate due to secondary gamma production in the depleted uranium shield are seen to be equal at a uranium thickness of approximately 5 cm. The secondary gamma production in the uranium is essentially completely due to radiative captures in the U^{238} . The fissions and radiative captures in the 0.20 wt % U^{235} portion of the shield were found to be negligible. This is largely attributed to the boral layer between the fission plate and the uranium shield which removes a substantial portion of the neutrons in the low energy range where the U^{235} cross section is high.

Gammas produced by neutron captures in the boron and aluminum of the boral sheet are seen

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Figure 5. STIR Shield Test Geometry

Figure 6. Calculated Gamma Dose Rate vs Depleted Uranium Thickness for STIR Gamma Shielding Experiments

Figure 7. Gamma Attenuation in Depleted Uranium and Lead Using a Fission Plate Source

TABLE 6
O5R RESULTS FOR A 30-cm LiH SHIELD
(8000 Histories)

Source-Detector Separation	Experimental* Relative Dose†	O5R Relative Dose†	O5R Standard Deviation (%)	O5R Deviation From Experiment (%)
16 (in.)	0.0056	0.00651	8.3	+16.2
6 (ft)	0.0089	0.00778	6.6	-12.6

*Interpolated for 30 cm from experimental data
†Normalized to 1 for zero thickness shield

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in Figure 6 to be relatively small. The attenuation of fission plate gammas by the uranium shield results in a slope equivalent to a relaxation length of 1.3 cm.

Preliminary experimental results corresponding to these calculations are presented in Section III-D-1 and Figure 7 of this report. Examination of these results shows excellent agreement between the calculated ($\lambda = 1.3$ cm) and measured ($\lambda = 1.29$ cm) relaxation lengths in depleted uranium for primary gammas (i. e. those originating in the fission plate). Much poorer agreement, however, is obtained for the relaxation lengths of the hard secondary gammas produced by neutron scattering and capture in the uranium shield (λ calculated = 18.5 cm, λ experimental = 8.2 cm). Possible reasons for this discrepancy are being investigated.

b. Anomalous Gamma Attenuation Measurements

The observation of anomalous gamma attenuation behavior in connection with STIR gamma shielding experiments was reported in the Shielding Experiments section of the preceding quarterly. This behavior was tentatively attributed to the production of secondary gammas by fast neutrons interacting with the lead shield which surrounds the gamma detector used in the STIR collimator. The effect of this interaction was analyzed, and it was found to be sufficiently large to contribute significantly to the observed anomalies. To further investigate this phenomenon, various thicknesses of borated polyethylene were placed in the throat of the collimator. The resulting gamma attenuation measurements yielded a slope which was characteristic of neutrons rather than gamma rays — thereby tending to confirm the postulated neutron interaction effects. A number of additional experiments were performed, which are described in Section II-D-1. As a result of these experiments,

neutron interactions in the collimator were effectively eliminated in later measurements by using 12 in. of lithium hydride in front of the collimator to reduce the ratio of neutrons to gammas. This configuration was used to obtain the experimental relaxation length data cited in the preceding paragraph.

c. Neutron Attenuation in Lithium Hydride

The O5R Monte Carlo program was used to analyze the STIR LiH neutron transmission experiments.⁽¹⁸⁾ Cases were run for thicknesses of 4, 10, 20, 30, and 40 cm of LiH, with detectors placed at both 16 in. and 6 ft from the source disk. Source energy biasing and path stretching were employed to improve statistics. The results were consistently lower than the experimental values and were low by as much as a factor of two at the 30- and 40-cm thicknesses. The most probable reason for this discrepancy is that serious undersampling of the important forward-directed neutrons resulted from the selection of neutron directions from an isotropic source. A 30-cm thickness case was rerun using a source direction biasing option in which 70% of the time-selected neutron directions lie in a cone with a half angle cosine of 0.60 and with its axis directed forward and perpendicular to the source disk. This mild source direction biasing gave improved agreement with experiment which is summarized in Table 6. It is believed that with more severe source direction biasing the 4- through 40-cm cases would yield good agreement with the experiment.

2. Tower Shield Facility (TSF) Experiments Planning and Analysis

A meeting was held in September with ORNL personnel for the purpose of discussing the analytical and experimental shielding program being conducted at both AI and ORNL. ORNL

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revealed their plans for immersing the SNAP shield in oil and exposing it to a neutron source. The purpose of this test is to check for internal voids in the shield and to check for separation of the L₁H from the casing. Subsequently, ORNL plans to use a line beam source of neutrons from the TSR-2 to irradiate the SNAP shield with an oil/steel "collar" shield. ORNL also revealed future plans for measurements of the radiation leakage from the unshielded SNAP reactor. AI is currently making preparations for analyzing the fast neutron leakage from the SNAP reactor. The O5R Monte Carlo program will be used in these calculations.

3. Shield Code Development

Documentation was completed for the OPEX shield optimization code. This program uses the method of steepest descent to minimize laminated shield weight, subject to the constraint of a specified dose rate. The variation of radiation levels with shield configuration must be obtained from code calculational methods (or from experiment) and supplied for use by the OPEX code. The code was used successfully in a recent SNAP 8 Manrated Reactor/Shield Design Study.

Corrections were made to the internal document which describes the formulation of the CLUB code. This program, which uses the Monte Carlo method of analysis, was developed to assist in the evaluation of neutron collimator efficiency and in the design of a collimator for use at the STIR facility. Errors in the documentation were recently brought to the attention of AI by ORNL personnel and have now been corrected. Publication of the revised documentation will be made in the next reporting period. It should be emphasized that the errors are confined to the documentation of the computer program, and are not present in the program itself.

Consequently, the code and any results obtained through its use are not affected by these changes.

Revisions are in progress on the SCAR code, a ray-tracing shielding program which computes fast neutron scattering from a thin reflecting surface. These revisions are being made to increase the computational efficiency, and correct some anomalous behavior of the program. This code has been used extensively in the analysis of the SNAPSHOT radiation levels and has been very effective in correlating analytical results with experimental measurements.

Computation of the 21-group neutron cross sections for use in shielding calculations performed by the DTF-II code was completed. Criticality calculations using the 21-group data agreed well with results obtained using the older 16-group data which was based on the Hansen-Roach compilation. For example, for a SNAP 8 type system (a 11.04-cm-radius cylindrical core surrounded by a 7-cm-thick BeO reflector) a calculation using the new data yielded a k_{eff} of 1.102, compared with a value of 1.109 using the 16-group data. Discrepancies were noted, however, between calculations of neutron fluxes in shields performed using the new 21-group library and those made using the 16-group data. To help resolve the discrepancies due to differing group structure from those due to differences in the basic cross-section data, a new 16-group library was generated from the same data that were used to produce the 21-group set. A comparison of shield calculations using these two consistent sets of data will provide a basis for determining the advantages of increased resolution (i. e., more groups) in the high energy region for calculations of neutron flux in shields.

Four-group cross-section data required for use by the SLAP code were obtained from ORNL. These data included the full set of P_3 transfer

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matrices required by the code. Final checkout of the SLAP code can now proceed using a preliminary data tape containing these data for eight elements (H, Li-6, Li-7, C, O, Fe, W, Pb).

A study was initiated to investigate the feasibility of using the DTF-II code to solve gamma ray transport problems. The GAMLEG code was obtained from LASL and is being adapted to the NAA computer system. This code will process gamma ray cross-section data into a format compatible with DTF-II.

D. SHIELD EXPERIMENTS

1. STIR Shield Experiments

Experiments to measure the attenuation of gamma rays in the presence of neutrons were continued. An extensive program to identify sources of secondary gamma rays was completed and reliable values were obtained for the primary and secondary gamma ray attenuations in depleted uranium and lead.

In the effort to identify the sources of secondary gamma rays, the gamma ray detector system was considered to include the ion chamber detector, the lead cask which shields the detector, the borated water tank, and the air-filled throat. Each of these items was selectively removed or changed to study its effect on the apparent attenuation by the uranium. The results of this survey are presented in Table 7.

TABLE 7
DETECTOR SYSTEM PARAMETER SURVEY

Configuration	Change in Gamma Dose Rate Observed Through 5.5 in. of Uranium
1. Standard-full collimator	Reference
2. Remove lead cask	Increase gamma 35%
3. Remove borated water	Increase gamma 70%
4. Flood throat with borated water	Increase gamma 16%
5. Replace ion chamber with NaI scintillation detector	Increase gamma 80%

From the measurements given in Table 7 it was concluded that secondary gamma rays were being produced in virtually every part of the collimator. Borated plastic was investigated as a neutron shield in front of the collimator but was discarded because of its relative ineffectiveness. Lithium hydride slabs were placed in front of the collimator and their thickness was varied using the leakage from 5.5 in. of uranium shield as a source. Those measurements showed that 10 in. of lithium hydride removed essentially all neutron effects. Measurements were made with a pure gamma ray source to verify the gamma attenuation properties of the lithium hydride. A value of 33 cm for this attenuation length for both fission gammas and cadmium capture gammas was obtained. On this basis a 12-in. slab of lithium hydride was selected as a permanent neutron filter, to be placed in front of the collimator.

The position of the detector in the collimator was varied. A position 24 in. behind the front edge of the lead region was found to provide the maximum signal-to-background ratio. Accordingly, this location was selected as the reference detector position.

Using the detector system described above, the attenuation of radiation from the fission plate was measured in depleted uranium and lead. The results are given in Figure 7. The measured relaxation lengths for primary and secondary gamma rays were as follows:

RELAXATION LENGTHS

	Primary Gammas	Secondary Gammas
Uranium	1.29 cm	8.2 cm
Lead	2.05 cm	7.4 cm

These secondary gammas are those produced by neutron capture or inelastic scattering in the heavy metal shield slabs.

It is considered significant that the two attenuation curves in Figure 7 cross at a shield

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Figure 8. Calculated Neutron and Gamma Displacement Efficiencies in Silicon

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Figure 10. Experimentally Derived Neutron Spectrum in STIR Pool

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Figure 9. Experimentally Derived Neutron Spectrum From STIR Fission Plate

TABLE 8
CALCULATED RELATIVE DAMAGE
FOR VARIOUS GAMMA SPECTRA

Gamma Source	Calculated Relative Damage
Co ⁶⁰	1.00
Cd(n, γ)	1.64
SNAP (Typical)	3.68
Fe(n, γ)	10.88
51% Fe(n, γ) 28% Mn(n, γ) 14% Cr(n, γ) 7% Ni(n, γ)	11.36
Ni(n, γ)	14.82

thickness of about 4 in. This indicates that lead can provide a lighter shield than uranium under some conditions, if its lower melting point can be accommodated.

2. Radiation Effects on Semiconductors

Evaluation of the effect of neutron energy spectra upon radiation damage in silicon semiconductors was completed. The results of irradiations in fission and water-moderated fission spectra at the Shield Test and Irradiation Reactor (STIR) were compared with the predictions of a point defect damage model. According to this model, the damage is proportional to the product of the displacement efficiency, $\epsilon_v(E)$, and the neutron spectrum, $\phi(E)$, i. e.

$$\text{Damage} \propto \int_0^{\infty} \epsilon_v(E) \phi(E) dE$$

The displacement efficiency, $\epsilon_v(E)$, is defined as the probability that a silicon atom will be displaced from its crystal lattice position by a photon or neutron of energy, E .

The calculated displacement efficiencies for both neutrons and gammas are shown in Figure 8. The STIR test spectra (Figures 9 and 10) were determined from threshold detector activations by the Weighted Orthonormal Polynomial Method. The calculated fission spectrum compares favorably with the theoretical U^{235} spectrum indicated by a solid line on Figure 9, while the water-moderated spectrum agrees with the predictions from angular-dependent in-pool spectra measured at ORNL. The predictions of damage in each environment, calculated from the preceding equation, were confirmed by the experimental measurements of degradation in the common emitter current gain in large samples of silicon transistors.

Determination of the energy dependence of gamma damage was continued. Preparation for

a STIR irradiation involving three neutron-capture gamma spectra was completed. Together with the previous Co^{60} and cadmium neutron-capture gamma irradiations, these spectra encompass all energies of practical interest for SNAP applications. The predicted relative damage for equal integrated gamma fluxes of each of these spectra are compared with the value for a representative SNAP core axial-leakage gamma spectrum in Table 8.

E. BASIC HEAT AND MASS TRANSFER

1. Noise Analysis

A draft report was prepared which summarizes the progress to date in applying electrical conductivity noise analysis to the accurate measurement of velocity profiles and fluid cross-flow in hydraulic test models. The draft was reviewed by the General Supporting Technology Program staff, and necessary revisions are being incorporated prior to submission for publication.

2. Mercury Supersaturation Study

a. Diffusion Chamber

Initial testing of the diffusion chamber was completed, and minor modifications were incorporated to improve its operation. A vacuum-jacketed condenser was installed on the coolant refluxing system, the upper condensing surface was modified by machining to give a smoother finish, and the single point condensate return was replaced with a gradually curving surface. This film removal surface was provided so that returning condensate would flow down the glass wall instead of dropping into the evaporating pool, causing ripples on the liquid surface and possibly introducing turbulence in the vapor space. Entry points for thermocouples sensing the pool temperature were changed from the top plate to the bottom plate, simplifying

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Figure 11. Modified Diffusion Chamber for Supersaturation Studies

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assembly and disassembly of the chamber. The modified chamber is shown in Figure 11.

The modified chamber was placed in operation and data were obtained on the temperature dependence of methanol critical supersaturation. Preliminary analysis of these data indicates the experimental values are below, but within 10% of, the theoretical values (Figure 12).

Further modification of the chamber is underway. Condensate removal from the upper surface will be improved by providing a conically concave upper surface with six film removal surfaces equally spaced around the circumference. This new upper surface will be machined from copper to decrease the temperature difference between the coolant chamber and the condensing surface.

Operation of the chamber will be resumed after installation of the new condensing surface.

b. Continuous Flow Nozzle Expansion Loop

As a result of a review of the problems attendant to fabrication of a suitable transparent nozzle, as described in the preceding quarterly

report, it was decided to postpone the use of this technique until later in the program. Instead, a metal nozzle fabricated from Haynes-25 alloy will be used for initial experiments. This nozzle will be instrumented with nine pressure taps and ~25 thermocouples. While this will not permit determination of average droplet size and droplet growth rate, as was projected for the optical detection means, the combination of pressure and temperature data will permit thermodynamic evaluation of condensation behavior. The metal nozzle was designed and fabricated, and is shown in Figure 13.

Fabrication of the mercury loop (Figure 14) for use with this nozzle is in progress. The system incorporates two radiantly heated pebble bed dryers. The two parallel dryer sections of different lengths are provided to permit evaluation of the relative effectiveness of the two units in removing droplets which could serve as nucleation sites. The flow in the nozzle will be supersonic in order to approach the degree of supersaturation theoretically required for homogeneous nucleation (i. e. condensation in the absence of nucleation sites). The nozzle

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Figure 12. Critical Supersaturation of Methanol vs Temperature

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Figure 13. Metal Nozzle for Mercury Continuous Flow Loop

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Figure 14. Piping Arrangement for the Mercury Continuous Flow Loop

Figure 16. Hydrogen Concentration vs Hydrogen Permeation Rate

Figure 15. Hydrogen Permeation Rate vs Hydrogen Pressure in Nickel Tube

Figure 17. Hydrogen Concentration vs Hydrogen Pressure in Nickel Tube

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back pressure will be regulated by use of a downstream throttle valve to produce a normal shock wave near the nozzle exit and to decelerate the flow to subsonic conditions before entry into the throttle valve and condenser. Two catch tanks and suitable valving will be provided to allow replenishment of the boiler inventory during operation. An inert gas system will provide the pressure required to transfer the condensed mercury to the boiler.

3. NaK-Hydrogen Partial Pressure Experiment

Measurements of hydrogen permeation rates as a function of hydrogen pressure using the modified 1320°F test capsule were completed. Similar measurements were initiated and completed using a modified 1260°F test capsule. A modified 1175°F capsule was prepared for testing.

The results of the tests to date are as follows:

1) The hydrogen permeation rate through the capsule is proportional to the first power of the hydrogen partial pressure. The permeation rate as a function of pressure and temperature is shown in Figure 15.

2) The hydrogen permeation rate through the capsule is directly proportional to the hydrogen concentration in the NaK. Figure 16 shows this relationship as a function of temperature.

3) The hydrogen concentration in the NaK is proportional to the first power of the equilibrium hydrogen partial pressure. This is shown in Figure 17, which also shows the effect of temperature on this relationship.

Two unexpected phenomena were observed in connection with the testing of the 1260°F capsule. One of these was the observation that the hydrogen permeation rate at a given hydrogen pressure gradually decreased with time as it was measured at intervals over the duration of

the test. This decrease finally asymptotically approached a constant value after about ten days to two weeks. Possible mechanisms which would explain this behavior are being investigated. It is tentatively attributed either to a possible chemical reaction occurring at the permeation surface or to the second unexpected phenomenon which was observed. This second phenomenon, observed when the capsule was removed from the furnace, was the elongation of the capsule bellows by about 2-1/2 in. This elongation is tentatively attributed to stress relief occurring in the Type 321 stainless-steel bellows which was highly stressed during manufacture. This elongation caused one of the ends of the capsule to extend into a nonisothermal region of the furnace, thereby making interpretation of the data somewhat more difficult. It is expected, however, that this nonisothermal condition will alter only the temperature dependence of the parameters and will not influence the pressure and concentration kinetics of the system. Future tests at lower temperatures will clarify this point. Tie rods have been added to the 1175°F capsule which is now being prepared for testing to prevent its elongation while in the furnace.

4. Duplex Tubing Materials Study

Advanced nuclear systems often impose operating conditions that cannot be satisfied by the use of a single material. Sometimes two materials may be bonded or "duplexed" to take advantage of the strength of one and the corrosion resistance of the other. The objective of this study is the evaluation of these composite materials.

The first composite material to be evaluated was coextruded Nb/Type 316 stainless-steel duplex tubing. The final results of this evaluation were reported in the previous quarterly report for this project. Due to an error, however, the regression equation describing the diffusion between niobium and Type 316 stainless steel was incorrect. The corrected equation is

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$$k = 5.588 \times 10^5 e^{-[3.459(10^4/T)]}$$

where

k = diffusion reaction rate [10^{-3} in./hr) $^{1/2}$]

T = temperature ($^{\circ}$ R).

This change in the equation does not significantly affect the calculations reported in the preceding quarterly which predicted the life of the bond in a Nb/Type 316 stainless-steel duplex tube. A topical report, "Metallurgical Study of Niobium/Type 316 Stainless-Steel Duplex Tubing", NAA-SR-11191 was written and is in the process of publication.

Explosive bonding was examined as a technique for obtaining duplex specimens for evaluation. The Dupont Corporation was contacted and preliminary discussions indicated that suitable specimens could be prepared at reasonable cost. Due to project redirection, however, all efforts in connection with this investigation, other than publication of the topical report, have

been terminated for the balance of the fiscal year.

5. Corrosion Testing Techniques

As mentioned in the last quarterly report the objectives of this project are: (1) to critically review the various liquid-metal corrosion-testing techniques used in the study of single-phase and two-phase systems, (2) to identify the techniques capable of producing the largest amount of reliable data at the lowest cost to realize specific objectives, (3) to help define the criteria to be met, and (4) to write a topical report which will serve as a "handbook" for the design and interpretation of corrosion experiments with liquid metals. During this quarter a literature survey was completed and the form of the report was outlined. Several other laboratories will be contacted to obtain information on unique and unusual liquid-metal corrosion testing techniques before the final report is completed.

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IV. EVALUATION OF EFFORT

The development of the basic equations for a computer code utilizing transport theory and space-time synthesis techniques lays the groundwork for the subsequent development of an accurate and efficient tool for reactor kinetic studies. The availability of a program which can properly account for the different neutron interaction mechanisms occurring in the various regions of a reactor system will permit a significant improvement in the ability to accurately predict SNAP transient behavior.

The cross sections being prepared for hydrogen bound in zirconium hydride will permit more accurate prediction of neutron spectra in the thermal region, and will provide a basis for generating temperature-dependent thermal spectra. Both of these capabilities will significantly improve the accuracy of SNAP nuclear calculations.

The critical examination of cross section and other nuclear data and assumptions routinely employed in SNAP nuclear calculations has revealed two areas having significant effects on the calculations, viz, the Zr resonance absorption and the Be (n, 2n) reaction. Certain other effects have been found to be negligible, however, this information is also of value in establishing a sound basis for SNAP nuclear calculations.

Satisfactory progress is being made in the effort to update and expand the AI Evaluated Nuclear Data File. The modification of the tape conversion program to process data files of other installations has greatly increased the utilization by the cross-section project of available nuclear data files. The integration of the various individual processing programs for the production of multigroup libraries has proceeded to the point where optimal production of multigroup libraries, in terms of computer time and personnel effort, will soon be achieved.

The new thermal subroutines prepared for the UPSTART-II program provide a valuable addition to the capabilities of this versatile analytical tool. With these it is now possible to perform detailed studies of time-dependent heat transfer within the reactor core, including the effects of fuel-cladding eccentricity.

The experiments with the SCA-5 critical assembly are progressing satisfactorily in providing basic data on control drum interaction, poison-backed control drum worths, and the relative reactivity effects of various poisons.

While shielding calculations were able to accurately predict the relaxation length in depleted uranium of fission plate primary gammas (based on STIR experiments) the more difficult and important prediction of secondary gamma relaxation length was in considerable error. These results illustrate the need for continued effort in improving shielding analytical techniques.

The determination of means for eliminating extraneous secondary gamma radiation effects at the STIR has permitted the basic measurements program to resume.

The fact that the final experimental evaluation of the effect of different neutron spectra on the radiation damage suffered by silicon semiconductors confirms the postulated damage model and lends support to the view that the model will also be confirmed for different gamma spectra. If future gamma experiments confirm this, it may be necessary to significantly revise SNAP shielding criteria.

Initial operation of the diffusion chamber has confirmed its value as a research tool in the investigation of critical supersaturation of vapors.

The revision of the design of the mercury nozzle expansion loop to utilize a metal, rather

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than a transparent quartz, nozzle ensures that basic mercury supersaturation thermodynamic data at low pressures can be obtained this fiscal year. The more difficult measurements of droplet concentration, size, and growth rate will be postponed till later in the investigation.

With the exception of minor experimental problems, the determination of the interrelationship of hydrogen partial pressure, concentration, and permeation rate in NaK systems is proceeding satisfactorily.

V. OBJECTIVES FOR THE NEXT REPORTING PERIOD

The objectives for the next quarter are as follows:

1) Nuclear and Control Analysis:

- a) Publish a report describing the DTF-II code
- b) Prepare a detailed description of the work done to date in space-time synthesis
- c) Prepare a corrected and updated interim library for use with the DTF-II code
- d) Complete checkout and documentation of the GRISM code (less the CONTRACT format processing code)
- e) Complete a basic description of the UPSTART-II code.

2) Nuclear Experiments:

- a) Initiate experiments with the SCA-5 critical assembly using the small core in a dry (low N_H) configuration.

3) Shield Analysis:

- a) Complete O5R analysis of STIR lithium hydride experiments
- b) Complete modifications to the SCAR code.

4) Shield Experiments:

- a) Initiate measurements of gamma and neutron attenuation at the STIR using combination shields consisting of one layer each of lithium hydride and a heavy metal (lead or uranium)

- b) Perform concurrent gamma irradiations of silicon devices at the STIR thermal column location

5) Heat and Mass Transfer:

- a) Complete assembly and perform initial tests of mercury nozzle expansion loop
- b) Continue diffusion chamber measurements of critical supersaturation, with special efforts being made to optimize measuring techniques so that they can be incorporated into mercury experimental phase. Initiate study of design changes necessary to adapt diffusion chamber to investigation of mercury supersaturation
- c) Initiate and complete measurements of hydrogen partial pressure and permeation in NaK, using 1175°F capsule. Initiate tests using 1075°F capsule.

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