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(SYSTEMS FOR NUCLEAR AUXILIARY POWER)
GENERAL SUPPORTING TECHNOLOGY
MAY-JULY 1965
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By
L. S. MIMS

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I. PROGRAM OBJECTIVE

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

The study of the application of variational techniques to the calculation of SNAP nuclear and control characteristics was extended from space-time synthesis to a cylindrical geometry R-Z synthesis. The basic equations required for transport theory calculations were developed and are being studied. The application of invariant imbedding theory to further simplify the multi-group, two-dimensional calculation was studied.

The 100 GT thermal cross section averaging code was modified and improved. The programming of the code was also converted to FORTRAN IV language in anticipation of forthcoming computer hardware changes.

Aldermaston cross section data for chromium and U^{235} were added to the Atomic International Evaluated Nuclear Data File (AIENDF) using the Tape Conversion Program (TCP).

The CHAD program was used to update the angular distribution data for a number of elements/isotopes in the AIENDF.

A preliminary version of the GRISM program was assembled and operated successfully. When completed, this program will provide a marriage of the GRAVE, PRISM, TRIXIE, and CONTRACT codes. As such, it will provide group-averaged

cross sections and scattering matrices for arbitrary or calculated neutron spectra, in the group structure desired, and in the format specified as needed for subsequent nuclear calculations.

Further improvements were made in the UPSTART II code.

Calculations were performed to estimate the background radiation level due to air and ground scattering of neutrons during shield experiments planned for the Tower Shield Facility at ORNL. The calculations indicated that a special "collar" shield would be required for all planned experiments to reduce the background radiation level to an acceptable value.

The probable sources of errors in the 21-group cross section library prepared for DTF-2 shielding calculations were identified and corrected.

The SLAP code, which performs a P_3 calculation of neutron attenuation in slab shield geometry, was expanded to four energy groups and converted to the FORTRAN IV programming system. This code is a subroutine for the improved SHOE shield optimization code.

Basic gamma attenuation measurements were initiated at the Shield Test and Irradiation Reactor (STIR) using heavy metal slabs of depleted

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uranium and tungsten and the STIR neutron collimator. Initial experimental results exhibited anomalous behavior. This effect is tentatively attributed to secondary gammas produced by neutron inelastic scattering in the lead shield surrounding the gamma detector.

The Co^{60} irradiation of selected silicon semiconductors was completed. Similarly, a STIR in-pool irradiation to investigate the effect of irradiation temperature on radiation damage to silicon semiconductors was completed. Preliminary data from the Co^{60} irradiation indicate that the damage ratio for hard gammas versus Co^{60} gammas is between 4:1 and 10:1. This is in accord with previous predictions.

Further tests and analyses were performed to determine the cause of data scatter in noise analysis measurements of flow velocity. A number of causative factors were identified and used to formulate guidelines for the design of future experiments. A previous test was repeated using the recommended guidelines. Data scatter was reduced from 20% in the earlier test to 2% in the new test.

In the basic study of mercury supersaturation behavior, attempts to construct a prototype

cloud chamber for use with methanol were terminated in favor of construction of a prototype diffusion chamber for use with the same material. The diffusion chamber appears more readily adaptable to a subsequent mercury investigation. Construction of the diffusion-chamber was completed and checkout was begun. Difficulties were encountered in the construction of a suitable glass nozzle for the mercury nozzle expansion loop.

Initial data were obtained in the NaK-Hydrogen Partial Pressure Experiment. These preliminary data points indicate a linear relationship between hydrogen pressure, hydrogen concentration in the NaK volume, and hydrogen permeation rate.

Analysis of data on the rate of diffusion occurring at elevated temperatures in the bond region of niobium-lined 316 stainless-steel tubing provided good correlation between diffusion rate and test temperature. Combining these data with the results of mechanical deformation tests has provided a basis for estimating tube service life at elevated temperatures.

The studies entitled, "Physical and Mechanical Properties of New Alloys" and "Corrosion Testing Techniques" were initiated.

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III. AREAS OF EFFORT

A. NUCLEAR AND CONTROL ANALYSIS

1. Reactor Theory and Method Development

a. R-Z Synthesis Application of Variational Technique

In applying a one-dimensional computer code to the nuclear analysis of a SNAP reactor it is usual practice to perform a one-dimensional radial calculation, while assuming a transverse (axial) buckling derived from the active core length. This approach is valid, only if the reactor is axially homogeneous. Since this is seldom the case, due to axial reflecting elements such as grid plates, plena, etc., calculations using this approach are subject to significant uncertainties. An alternative course is the use of a two-dimensional code, such as DDK. This approach also has disadvantages in that available codes have a limited number of available mesh points and consume large amounts of computer time. The variational technique,

presently being studied, approaches the problem by obtaining a few one-dimensional solutions for one of the dimensions of interest, and synthesizing from these an approximate multidimensional solution. It is expected that the application of the technique will produce multidimensional transport theory calculations which are comparable in accuracy with conventional multidimensional transport theory analyses, but which involve significantly less computer time. Reduction of computing time by a factor of three to five is commonly observed when this technique is applied to multidimensional diffusion theory calculations.

Since SNAP reactors are most conveniently described in the R-Z coordinate system, a synthesis in terms of these coordinates has been studied using the variational technique and S_n neutron transport theory. The approach which has been developed is as follows:

For isotropic scattering in R-Z geometry, the equation for the stationary flux is⁽¹⁾

$$G_r \{ \varphi(r, z, E, \xi, \omega) \} = \left(\mu \frac{\partial}{\partial r} + \xi \frac{\partial}{\partial z} - \frac{\eta}{r} \frac{\partial}{\partial \omega} \right) \varphi(r, z, E, \xi, \omega) + \sigma(r, z, E) \varphi(r, z, E, \xi, \omega) - \int_{E'} \sigma_{iso}(r, z, E, E') \left[\int_0^{2\pi} \int_{-1}^1 \varphi(r, z, E', \xi', \omega') \cdot d\omega' d\xi' \right] dE' = 0 \quad \dots (1)$$

The equation for the time-independent adjoint is:

$$G_r^+ \{ \varphi^+(r, z, E, \xi, \omega) \} = - \left(\mu \frac{\partial}{\partial r} + \xi \frac{\partial}{\partial z} - \frac{\eta}{r} \frac{\partial}{\partial \omega} \right) \varphi^+(r, z, E, \xi, \omega) + \sigma(r, z, E) \varphi^+(r, z, E, \xi, \omega) - \int_{E'} \sigma_{iso}(r, z, E', E) \left[\int_0^{2\pi} \int_{-1}^1 \varphi^+(r, z, E', \xi', \omega') d\omega' d\xi' \right] dE' = 0 \quad \dots (2)$$

Here

$$\sigma_{iso}(r, z, E, E') = \frac{1}{4\pi} \sigma_{s, iso}(r, z, E, E') + \frac{1}{4\pi} \chi(E) \nu \sigma_f(E') \quad ,$$

¹Superscripts refer to the like -numbered References at the end of this report.

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Figure 1. Description of Position Vector $\underline{r}$ and Direction Vector $\underline{\Omega}$ in Cylindrical Geometry

and

$$\sigma_{s,iso}(r, z, E, E')$$

is the isotropic macroscopic cross section for scattering neutrons from energy E' to E . The angle ω and the direction cosines ξ , η , and μ (see Figure 1) are related by

$$\mu = \sqrt{1 - \xi^2} \cos \omega$$

$$\eta = \sqrt{1 - \xi^2} \sin \omega \quad \dots (3)$$

The synthesis approximation for the flux and adjoint are

$$\varphi(r, z, E, \xi, \omega) = \sum_{k=1}^K N_k(r, E, \xi, \omega) Z_k(z, E, \xi) \quad \dots (4)$$

and

$$\varphi^+(r, z, E, \xi, \omega) = \sum_{k=1}^K N_k^+(r, E, \xi, \omega) Z_k^+(z, E, \xi) \quad \dots (5)$$

Here the trial functions N_k and N_k^+ are chosen from radial DTF cases. The expressions (Equations 3 and 4) for the flux and adjoint, and the functional $G_r(\varphi)$ (Equation 1) are inserted into the functional:

$$I(\varphi, \varphi^+) = \iiint \varphi^+(r, z, E, \xi, \omega) G_r\{\varphi(r, z, E, \xi, \omega)\} dz 2\pi r dr dE d\xi d\omega \quad \dots (6)$$

Integrations over r and ω may be performed directly. The following equations are found for the mixing functions Z_k and Z_k^+ by a method similar to that used in Reference 2. Summation over repeated indices is assumed.

$$\begin{aligned} & M_{ij}(\xi, E) \xi \partial Z_j(z, E, \xi) / \partial z + [B_{ij}(\xi, E) + A_{ij}(z, \xi, E)] Z_j(z, E, \xi) \\ & - \int_{E'} S_{ij}(z, \xi, \xi', E, E') Z_j(z, E', \xi') d\xi' dE' = 0 \quad \dots (7) \end{aligned}$$

and

$$\begin{aligned} & -M_{ij}^T(\xi, E) \xi \partial Z_j^+(z, E, \xi) / \partial z + [B_{ij}^T(\xi, E) + A_{ij}^T(z, \xi, E)] Z_j^+(z, E, \xi) \\ & - \int_{E'} S_{ij}^T(z, \xi', \xi, E', E) Z_j^+(z, E', \xi') d\xi' dE' = 0 \quad \dots (8) \end{aligned}$$

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Here the elements $M_{ij}(\xi, E)$, $B_{ij}(\xi, E)$, $A_{ij}(z, \xi, E)$, and $S_{ij}(z, \xi, \xi', E, E')$ of the matrices M , B , A , and S are given by

$$M_{ij}(\xi, E) = \int_0^{2\pi} d\omega \int_0^R 2\pi r dr N_1^+(r, E, \xi, \omega) N_j(r, E, \xi, \omega) \quad \dots (9)$$

$$B_{ij}(\xi, E) = \int_0^{2\pi} d\omega \int_0^R 2\pi r dr N_1^+(r, E, \xi, \omega) \left[\mu(\xi, \omega) \frac{\partial}{\partial r} - \frac{\eta(\xi, \omega)}{r} \frac{\partial}{\partial \omega} \right] N_j(r, E, \xi, \omega) \quad \dots (10)$$

$$A_{ij}(z, \xi, E) = \int_0^{2\pi} d\omega \int_0^R 2\pi r dr N_1^+(r, E, \xi, \omega) \sigma(r, z, E) N_j(r, E, \xi, \omega) \quad \dots (11)$$

and

$$S_{ij}(z, \xi, \xi', E, E') = \int_0^{2\pi} d\omega \int_0^{2\pi} d\omega' \int_0^R 2\pi r dr N_1^+(r, E, \xi, \omega) \sigma_{iso}(r, z, E, E') N_j(r, E', \xi', \omega') \quad \dots (12)$$

In Equation 8 the superscript T denotes the transposition of the matrix, for example $M_{ij}^T = M_{ji}$.

While numerical integration of Equation 7 (with suitable boundary conditions) is simpler than the integration of Equation 1, it still may be too cumbersome, as it stands, for the R-Z synthesis to be an effective design tool, since the mixing functions Z_j are really three-dimensional (that is, dimensions in z , ξ , and E). Equation 7 could be simplified by removing the E dependence from the mixing functions. Then the matrices M , B , A , and S would include integrations over E and E' . The ξ dependence cannot be integrated out of Equations 7 and 8 since:

$$\int_{-1}^1 \xi M_{ij}(\xi, E) d\xi = 0 \quad \dots (13)$$

The differential equation for this case is:

$$M_{ij}(\xi) \xi \partial Z_j(z, \xi) / \partial z + [B_{ij}(\xi) + A_{ij}(z, \xi)] Z_j(z, \xi) - \int_{\xi'} S_{ij}(z, \xi, \xi') Z_j(z, \xi') d\xi' = 0 \quad \dots (14)$$

Solution of Equation 14 is only slightly more complicated than the solution of the one-group (in energy) transport equation in plane geometry. This approach is currently under consideration at Bettis.⁽³⁾

Since the S matrix in either Equation 7 or Equation 14 is dependent on both ξ and ξ' , because of the trial functions, no particular simplification results in the calculation of the mixing functions by restricting the calculation to isotropic scattering. That is, σ_{iso} in Equation 12 can be replaced by

$$\sigma(r, z, E, E', \xi, \xi', \omega, \omega') = \sigma_s(r, z, E, E', \xi, \xi', \omega, \omega') + \frac{1}{4\pi} \chi(E) \nu \sigma_f(r, z, E') \quad \dots (15)$$

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It may be noted in passing that the two-dimensional fluxes and adjoints calculated from Equations 4 and 5 can be used as trial functions for an R-Z-T synthesis.

b. 100-GT Code

The 100-GT Thermal Cross Section program was converted to FORTRAN IV, and checked out. This conversion results in significant savings in operating costs due to the use of buffered input-output. It will also make the program compatible with scheduled changes in NAA computing hardware. The elements/isotopes U^{235} , U^{238} , H (zirconium-bound), B, Zr, O, Fe, Mo, Be, Na, C, Ni, Al, Cr, K, and Nb have been added to the thermal data library. Work is underway to build an up-to-date thermal scattering kernel into the program to allow more accurate calculation of thermal cross sections. The possibility of combining the AIENDF fast cross section data with these thermal data has been studied.

Since this code has not been comprehensively described in previous quarterlies, the following brief description is presented. The 100-GT Computer Program is a tool for investigating thermal scattering kernels and their resulting neutron spectra. The code prepares 100-group tables of thermal cross sections and stores them on tape, calculates thermal scattering cross sections from an analytical kernel supplied in subroutine form and stores these on tape, and uses the data to calculate a 100-Group Thermal neutron spectrum. The spectrum is then used to calculate cross sections for use in DTF or similar reactor analysis codes. The 100-GT Program combines into one operation a process once performed by a combination of the cross section averaging code, the DSDE Thermal Scattering Matrix Code, the ULCER library code, and the QUICKIE thermal spectrum code. The large number of energy groups

was chosen to permit more accurate treatment of scattering from narrow energy bands such as are found in zirconium hydride, and to eliminate the necessity for using an input flux guess to average group-wise cross sections. The code enables the user to see the results of one or several sets of parameters evaluated in a single computer run, where several separate runs on separate days were formerly required. This is a significant advantage when comparing theoretical calculations with experimental data. The program is not intended as a tool for routine reactor analysis, but rather will provide cross section libraries for use in such analyses.

2. Cross Section Development

a. Basic Cross Section Data

The following activities were completed in connection with improvement of basic cross section data:

1) TCP was used to place cross section data for chromium (obtained from Aldermaston) into the AIENDF. U^{235} fission and capture cross sections for $0.001 \leq E \leq 1.0$ Mev was placed on one copy of the cross section tape. These U^{235} data (also obtained from Aldermaston) more closely conform to current best measurements

2) A new set of oxygen scattering data obtained from nuclear model calculations performed at KAPL was used to compute neutron age in water. The results indicate that this set of calculated data is superior to all other sets tested to date.

3) An analysis of new cross section data for oxygen, aluminum, and sodium, based on the new edition of BNL 325, was completed. As a result, the oxygen (n, α) cross sections in the Mev range were modified to conform with the new measured values. Preparations were initiated for a comprehensive comparison in the near future of all of the new

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BNL 325 data (for elements in the range of $Z = 1$ to 20) with the corresponding data presently in the AIENDF.

4) The nuclear model code, NEARREX, was modified to incorporate a number of improvements, and was converted to FORTRAN IV computer language. The modified code was employed to calculate improved cross section data for Pu^{238} and Cm^{244} . These data were subsequently added to the AIENDF.

5) The CHAD program, which automates the handling of angular distribution data, was completed, and was used to update the AIENDF for the elements/isotopes: H^3 , Li^6 , Li^7 , Nb, and W. Cross section angular data for W^{184} was calculated using the 2 PLUS nuclear model program. Calculations were completed for both the elastic and inelastic differential cross sections.

b. Cross Section Processing

The GRAVE code produces group-averaged cross section data from point cross section data for specified input neutron spectra. The PRISM code similarly calculates group-averaged elastic scattering cross sections and inelastic scattering matrices. The TRIxie code is a subroutine within the GRAVE code for handling resonance integral calculations. Since different neutronic codes have different library format requirements for input data, the CONTRACT code is used to convert the output of GRAVE and PRISM to any one of several possible formats. During the past quarter an effort has been made to combine GRAVE, PRISM, and CONTRACT into a single code, GRISM. Availability of such a code will permit complete preparation of cross section input data in a single computer run. In addition to providing greatly reduced overall preparation time, corollary benefits of the GRISM code will include more efficient data handling and less opportunity for programming or operating errors.

Substantial progress was made in the preparation of the GRISM code. A tape of angular data was prepared for use with PRISM which eliminates the need for binary card input to this code. The CONTRACT code was expanded to handle GRAVE-PRISM punched output for the FORM, CAESAR IV and 35-group DTF libraries. A preliminary version of GRISM was assembled and operated successfully in initial trials. Efforts to provide additional refinements and flexibility to GRISM are continuing.

3. Reactor Control and Stability Analysis

The UPSTART II digital system simulation program is being developed to improve the analysis of reactor system transients. The program is composed of self-contained, general-purpose modules which simulate the dynamics of each major component of the reactor system. These components include the reactor core, the control system, a liquid-to-liquid counterflow heat exchanger, a liquid-to-space and/or gas heat exchanger, a SNAP 10A type thermoelectric pump, and piping segments which connect the components forming the system to be studied.

The mathematics describing the neutron kinetics are the conventional monoenergetic, six-group equations. The mathematical model for thermal kinetics of heat transfer to (or from) a moving fluid^(4,5) was developed for the program.

During the past quarter, additional subroutines were prepared or revised and operation of the overall program was further improved. Digital subroutines were prepared for the following components: the auxiliary heat exchanger, the parasitic load resistor, the S8DS drum controller and scram mechanism, and the heat exchanger for the SNAP reactor planned for use in shielding experiments at the ORNL Tower Shield Facility. In addition, the subroutine for the SNAP 10A radiator-condenser was modified and improved.

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The fast kinetics option of UPSTART II was tested and compared with the results provided by the AIREK III digital neutron kinetics code. Good agreement was obtained. The SNAP 10A system was simulated using the latest version of UPSTART II and satisfactory operation was observed. The computer time required for calculation of a given transient was about 40% of that required using a previously checked-out version of UPSTART. The S8DS system startup was also simulated and satisfactory operation was obtained.

B NUCLEAR EXPERIMENTS

1. Basic Core Configuration Experiments

The SCA-5 critical assembly was not operated during the report period. Instead, experiments were conducted on the SCA-4B in support of the Aerospace Safety Program. These experiments have been completed and the SCA-5 is being returned to operational status. Experiments planned for the immediate future include studies of control drum interaction and measurement of the total shutdown margin.

C SHIELDING ANALYSIS

1. STIR Experiments Planning and Analysis

The ANNEX-3 Monte Carlo program was used to analyze the results of STIR experiments on neutron transmission through slabs of lithium hydride. The agreement with experiment for 20 cm of lithium hydride was excellent, however, the 40 cm calculation was lower than experiment by a factor of 2. The analysis is being continued by adjusting biasing techniques in the code.

Analysis of primary and secondary gamma production from slabs of depleted uranium and tungsten is being carried on in conjunction with the heavy metal slab experiments now in progress at the STIR. The DTF and 14-0 codes are being utilized for this analysis.

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

Analyses of background neutron scatter expected to occur in shielding experiments using a SNAP 10A reactor and a SNAP 2 shield configuration at the Oak Ridge Tower Shield Facility have been performed. The first case considered the system suspended in air with the reactor center 13-1/2 ft above the ground. The detector was located on the centerline 12 ft from the center of the reactor and was collimated so that the entire bottom of the shield could be seen.

The analytical techniques were formulated to perform dose calculations involving both single scatter in air, and single scatter in the ground or air, followed by one scatter in the bottom layers of the shield. The collimator was assumed to be perfect, i.e., only those neutrons which approach the detector in the solid angle defined by the collimated cone are counted. The geometry considered is shown in Figure 2, and the results are presented in Table 1.

TABLE 1
CALCULATED AIR AND GROUND
SCATTER, INSTRUMENT-RATED
SHIELD WITHOUT COLLAR
(Transmitted Dose
= 6.5 mrad/hr-kw)

Scatter Source	Air Scatter Dose (mrad/hr-kw)
a) 1 scatter	24.5
b) 1 scatter plus 1 shield scatter	2.5
c) Ground contribution	0.6
Total	27.6

Letters designate rays shown in Figure 2

Since the transmitted dose is 6.5 mrad/hr-kw, it is obvious from Table 1 that supplementary shielding is required. The proposed shielding

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Figure 2. Shadow Shield — Air, Ground, and Shield Scatter Sources

would take the form of a collar around the SNAP shield. It is estimated that the presence of a supplementary collar shield would contribute a background of about 1.0 mrad/hr-kw (estimated from ORNL Monte Carlo calculations for a collar of infinite radius).

The above calculations were extrapolated to study the background for a man-rated shadow shield and a man-rated 4π shield. The results are given in Table 2 (perfect collimator assumed).

It is concluded that supplementary (collar) shielding is required for all shield transmission experiments likely to be performed at the ORNL Tower Shield Facility.

3. Shield Code Development

Analysis of the 21-group DTF-II neutron cross-section library revealed that the calculational discrepancies noted in the previous

TABLE 2
RATIO OF SCATTERED DOSE TO DIRECT DOSE BELOW SHIELD AT CENTER-LINE, MAN-RATED SHIELDS

Man-Rated Shield Type	Test Conditions	Ratio of Scattered Dose to Direct Dose
Shadow	Air, no collar	3000
Shadow	Air, collar	0.15
4π	Air, no collar	7.6
4π	Air, collar	0.15

quarterly report were probably due to errors in the treatment of the $(n, 2n)$ reaction in beryllium and in the calculation of the downscatter transfer matrices. These errors have been corrected and a thorough check of the data is being made before the library is released for general use.

The SLAP code, which performs a P_3 calculation of neutron attenuation in slab shield geometry was expanded to four energy groups and converted to the FORTRAN-IV programming system. This code will serve as the neutron flux calculation subroutine for the expanded SHOE shield optimization code currently being developed. New generalized boundary conditions, developed by O'Reilly,⁽⁶⁾ were added to provide a more exact treatment of the source-shield interface. For the case of a carbon slab with a fission source at one face, the code results agreed well (within about 20%) with Moments Method calculations for the fast group. A check of the three lower energy groups must await final checkout of the PRISM-II code, which will provide full transfer matrices for P_3 calculations.

The OPEX code, a shield optimization code which describes the dose rate as the sum of a number of exponential terms, was satisfactorily checked out. This was done using a sample

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case in which the minimum weight was calculated for a four-region shield which satisfied a specified dose rate. This code, which will handle any basic geometry, requires input from other neutron and gamma codes or from experimental data. Its purpose is to aid in optimization studies while the more sophisticated SHOE code is being revised.

4. Space Radiation Environmental Effects

A review has been in progress to determine existing uncertainties in the contribution of SNAP reactors to the Van Allen radiation belts. The electron intensity of the trapped radiation belts is increased due to the decay of neutrons from an orbiting reactor. The mathematical models and methods of analysis are relatively new and are subject to improvement. The con-

tribution from high power reactors, particularly in non-polar orbits, is potentially significant.

D. SHIELD EXPERIMENTS

1. STIR Shield Experiments

Gamma attenuation measurements were initiated using shields consisting of slabs of depleted uranium and tungsten. Combinations of these materials with lithium hydride as a neutron shield are of interest to the SNAP program. The attenuation of gamma rays from the STIR fission plate was measured over the range from 0.5 to 5.5 in. of uranium using the collimator as shown in Figure 3. The measurements were performed with and without a 4-in. slab of lithium hydride between the fission plate and the uranium. The results showed an initial

Figure 3. STIR Collimator

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attenuation coefficient of 0.91 cm^{-1} which is consistent with published values. A secondary contribution was also observed, however, which had a slope characteristic of the fast neutron removal cross section for uranium. The magnitude of this component varied with the amount of neutron shielding present, but was significantly higher than would have been anticipated on an analytical basis — suggesting the possibility of an extraneous source.

Supplementary experiments were conducted to identify the source of this discrepancy. Borated polyethylene was placed in the throat of the collimator to attenuate neutrons incident on the detector. This experiment showed two regions of apparently different gamma attenuation characteristics: The first region, observed with no gamma shield present, was characterized by a markedly more shallow attenuation slope than the second region which was observed when significant thicknesses of gamma shielding were present. In one arrangement, with 1 in. of uranium as a gamma shield, 5 in. of borated polyethylene were needed to substantially remove the neutron contribution to the dose. Because of the high primary gamma attenuation provided by uranium, proportionately more polyethylene would be required to provide the equivalent neutron attenuation with greater uranium thicknesses. Thus, with a 5.5-in. uranium thickness, about 27 in. of borated polyethylene were required to provide a proportionally equivalent neutron attenuation. Further investigation is required to provide a clearer interpretation of the results.

The power level of the fission plate was measured by comparison of the heating rates using electrical and nuclear heat generation. The experiment indicates that the fission plate power level is 1.58 kw when the reactor is at 1 Mw. This level is higher than that estimated in the design of the new core, but does not introduce any operational problems.

2. Radiation Effects on Semiconductors

Experimental investigations of the influence of neutron and gamma energy spectra and of damage recovery processes upon radiation damage in silicon semiconductors were continued. Selected silicon transistors, exhibiting the full range of sensitivity to radiation, were irradiated at the Shield Test and Irradiation Reactor (STIR) and the AI Co⁶⁰ Gamma Facility. The STIR irradiation exposed samples to the Cd(n,γ) gamma spectrum, which is predicted analytically to be over four times as damaging as an equivalent exposure to Co⁶⁰ gammas. Preliminary results indicate a measured damage ratio between 4:1 and 10:1. Another phase of this experiment involved measuring the effect of irradiation temperature upon the net damage in the range 0 to 150°F, which is representative of probable SNAP payload environments. Spectra for this phase were the Cd-capture gamma spectrum and a highly water-moderated neutron spectrum, which are believed to be very similar in their damaging effects to the neutron and gamma spectra in a SNAP sub-shield environment. For both the Co⁶⁰ and STIR irradiations, semiconductor types, temperature and electrical operating conditions were chosen to permit a correlation with the results of previous irradiations in the Cd gamma spectrum and in fission and water-moderated neutron spectra.

Analysis of the STIR data requires a more accurate knowledge than presently available of the in-pool fast neutron spectrum. A literature review of methods for solving this problem was performed, and the most promising technique was selected — analysis of threshold detector activations by the method of weighted orthonormal polynomial expansions. A computer code, SWOPEX, was written to perform the necessary calculations and the required cross section and trial spectra were assembled. Evaluation of the capabilities of this technique is 50% com-

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complete. Two topical reports, presenting the analytical background for this effort, were issued this quarter. These reports are NAA-SR-10612, "Radiation Damage in Silicon: Neutron Spectrum Considerations," dated June 1965, and NAA-SR-10856, "Comparison of Neutron and Gamma Radiation Damage in Semiconductors," dated June 1965.

E. BASIC HEAT AND MASS TRANSFER

1. Noise Analysis

The objectives of the noise analysis investigation in this period were to increase the accuracy and minimize the scatter occurring in transit time measurements using conductivity cells.

A number of experiments were performed to better define some of the conductivity cell problems. The experimental setup included a container with a solution of known electrolyte concentration and two electrodes capable of being positioned a known distance apart. One electrode was the cross sectional end surface of a wire and the other was a flat plate. The wire electrodes were interchangeable and concentration could be varied. Figure 4 presents a plot of cell impedance at a large gap as a function of frequency. The frequency range of negligible detrimental effects (flat region of curve) in this instance is seen to be above 10 kc/sec.

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Figure 4. Impedance vs Frequency Characteristic of a Conductivity Cell

One of the reasons for the frequency dependence exhibited by the curve in Figure 4 is believed to be electrode polarization (i.e., the formation of an ion sheath around the electrode as occurs with batteries). This is illustrated in Figure 5 for a dilute NaNO_3 solution, a 100 kc/sec modulation frequency and two different wire diameters. Here it is seen that the impedance does not approach a value of zero ohms at zero gap. Rather, it approaches a finite value greater than zero at very close spacings, and discontinuously becomes zero at electrode contact. The effect of low frequency and high concentration is shown by Figure 6. Here it is seen that polarization effects were so great as to cause an increase in impedance as the cell gap was reduced. This is in contrast to the results shown in Figure 5.

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Figure 5. Impedance vs Cell Gap of Conductivity Cell (0.1% NaNO_3 , 100 kc/sec)

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Figure 6 Impedance vs Cell Gap of Conductivity Cell (6% NaNO₃, 1 kc/sec)

A number of experiments similar to those described above were performed. From these, it was concluded that polarization effects increased with electrolyte concentration and with reduced electrode diameter. As would be expected, impedance was observed to decrease with electrolyte concentration. Variation of the amplitude of the modulation frequency was found to have a negligible effect upon the impedance measurement.

Based on the above observations, a number of design guidelines were developed for conductivity cells. Assuming an insulated electrode conducting through a solution to a larger grounded region, the following guidelines may be used:

Electrode Area. The cell electrode area should be large to minimize polarization effects and small to minimize position errors. The optimum size based on past flow studies is about 1/16-in. in diameter.

Insulation. The thickness of the insulation surrounding the cell should be greater than twice the electrode gap.

Modulation Frequency. From Figure 5, the modulation frequency should be greater than 100 kc/sec and less than the break frequency of the cell time constant.

Cell Voltage. The cell voltage should be low to minimize concentration polarization.

yet high enough to overcome electrolysis and overvoltage. A range of between 2 and 4 volts appears to be suitable. Since ac amplitude does not appreciably affect resistance, this is not a primary requirement.

Concentration. The electrolyte concentration should be low to minimize concentration polarization. Concentrations between 0.0001 and 0.001 wt % are recommended.

Electrode Material. The electrode materials should be noncorrosive and nonreacting with the solution. The effect of surface finish is not known, but nothing detrimental has been observed in relation to smooth or rough surfaces.

Based on these design guidelines, a single-channel test was performed with an improved conductivity cell. This test was similar to the single-channel tests reported previously. The improved cell and higher modulation frequency reduced the scatter in the data for this test from >20% to <2%.

An error analysis was performed for previously completed flow measurements. The method used in this analysis was to determine the maximum error and define this as three standard deviations. This standard deviation, or standard error, was then used in the error analysis. Figure 3 of Reference 7 was used to determine velocity differences as a function of cell position errors. Since most of the errors were small, percentage velocity and transit time errors were considered equivalent. Table 3 summarizes the observed flow errors and the expected error of an improved cell. Of all the errors listed, polarization was the most difficult to assess.

From Table 3 it is seen that the improved cell requires closer cell position tolerances (Items 1 and 2), a larger insulator (Item 3), and minimization of the polarization problem (Item 6),

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TABLE 3
ERROR TABULATION OF INSTRUMENTATION USED IN FLOW STUDIES

	Maximum Deviation	Standard Velocity Error	Expected Error for Improved Cell	Notes
1) Longitudinal Inter-Cell Spacing	$\pm(0.0625)/2.5$	$\pm 0.83\%$	$\pm 0.42\%$	Obtained from equipment calibration
2) Rotational Position	$\pm(1/16)/17/64$ of 57°	$\pm 1.2\%$	$\pm 0.6\%$	
3) Feeling Depth	1/3 to 2/3 of gap	$\pm 4.7\%$	$\pm 1.0\%$	
4) Transit Time Computer	-	$\pm 1.0\%$	$\pm 1.0\%$	
5) Graphical Resolution	$\pm 3.0\%$	$\pm 1.0\%$	$\pm 1.0\%$	Assigned Values, See Discussion
6) Polarization	$\pm 50\%$	$\pm 16.7\%$	$\pm 1.0\%$	
Total Standard Error of Measurement	-	$\pm 17.3\%$	$\pm 2.1\%$	Square root of the sum of the squares

An error analysis was also performed for the area variations of a given flow channel. The area variation is directly related to the gap between circular fuel elements. This gap may vary from 0 to 20 mils. The maximum channel area then varies between 0.013 and 0.023 in² or 0.018 ± 0.005 in². The channel area standard error is then 9.3%. The corresponding flow velocity standard error is approximately $\pm 4.6\%$.

Combining the area error with the measurement error yields a standard 18% scatter in the data which is in reasonably good agreement with the 15% scatter observed for previously completed measurements.

2. Mercury Supersaturation Study

a. Cloud Chamber Studies

Efforts to develop a prototype expansion-type cloud chamber to investigate mercury supersaturation behavior were discontinued because a suitable valve could not be found which would provide expansion times sufficiently fast to per-

mit homogeneous nucleation. Instead, it was decided to use a diffusion chamber in this study. Construction of a prototype of such a chamber for use with methanol was completed. Figure 7 shows the experimental apparatus. Checkout of the apparatus is underway.

b. Continuous Flow Nozzle Expansion Loop

The second aspect of the study of mercury vapor supersaturation behavior involves the design, construction, and operation of a mercury vapor, continuous flow, nozzle expansion loop. Fabrication of a prototype glass nozzle was completed. The fabrication technique was found to be unsatisfactory since it produced excessive inside surface roughness. The feasibility of other methods of fabrication is being investigated. Figure 8 illustrates the transparent nozzle planned for use in the experiment. Revision of the design of the optical detection device was completed. Figures 9 and 10 illustrate, respectively, the revised optical and electronic designs for the detection apparatus.

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Figure 7. Diffusion Chamber for Supersaturation Studies

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Figure 8. Transparent Nozzle Design for Mercury Vapor Nozzle Expansion Loop

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Figure 9. Optical Detection Device, Optical Design

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Figure 10. Optical Detection Device, Electronic Design

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3. NaK-Hydrogen Partial Pressure Experiment

Testing was initiated using one of the test capsules prepared for operation at 1260°F. The capsule was heated isothermally and outgassed to a hydrogen background flow rate of 0.2 cc/hr. Hydrogen was then admitted to the SNAP 8 membrane at one atmosphere pressure increments up to a maximum pressure of 8 atm. Hydrogen flow rates through the NaK were measured with a calibrated mass spectrometer. It was determined that the hydrogen flow rates into the NaK with this experimental arrangement were too low to provide significant hydrogen concentration buildup in the NaK. Further, accurate pressure readings could not be obtained from the McLeod gauge because of the large void volume in the system and the resulting time required for the pressure to attain an equilibrium value at the low hydrogen flow rates.

An alternate test procedure was adopted wherein the hydrogen was admitted to the capsule through the nickel pressure-measuring tube connected to the McLeod gauge. While this technique increased the hydrogen flow into the NaK and provided higher hydrogen concentrations, it was found that accurate equilibrium dissociation pressures could not be obtained.

A decision was then made to modify the capsules by removing a portion of the ceramic hydrogen barrier material from the SNAP 8 membrane (hydrogen supply source). A special sandblasting tool was used to remove four circumferential rings of coating from the inside of the membrane of a capsule prepared for operation at 1320°F. This provided about 5 cm² of bare metal, which permitted significantly increased hydrogen flow rates into the NaK. Control of flow was maintained by means of a series of calibrated standard hydrogen leaks.

The modified 1320°F capsule is presently under test and appears to be yielding consistent and valid data. The preliminary results are as follows

- 1) The hydrogen permeation rate through the capsule is proportional to the first power of the hydrogen pressure (dissociation pressure) in the nickel tube. The data are plotted in Figure 11.
- 2) The hydrogen permeation rate through the capsule is directly proportional to the hydrogen concentration in the NaK as shown in Figure 12.

Figure 11. Hydrogen Permeation Rate vs Hydrogen Pressure in Nickel Tube

Figure 12. Hydrogen Concentration in NaK vs Hydrogen Permeation Rate

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Figure 13. Hydrogen Concentration in NaK vs Hydrogen Pressure in Nickel Tube

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Figure 14. Hydrogen Concentration in NaK vs Square Root of Hydrogen Pressure in Nickel Tube

TABLE 4

SUMMARY OF DIFFUSION DATA FOR Nb/316 SS DUPLEX TUBING

Co-Extruded				Explosively Bonded			
Temperature (°F)	Time (t) at Temperature (hr)	Diffusion Zone* = X (10 ⁻³ in. or mils)	K = X/√T (10 ⁻⁴ in./hr ^{1/2})	Temperature (°F)	Time (t) at Temperature (hr)	Diffusion Zone* = X (10 ⁻³ in. or mils)	K = X/√T (10 ⁻⁴ in./hr ^{1/2})
1960	2	0.50	3.53 x 10 ⁻¹	1960	2	0.51	3.53 x 10 ⁻¹
1960	4	0.76	3.80 x 10 ⁻¹				
1960	8	0.82	2.89 x 10 ⁻¹				
1960	12	0.98	2.71 x 10 ⁻¹				
1870	5	0.49	2.16 x 10 ⁻¹	1860	5	0.55	2.46 x 10 ⁻¹
1840	8	0.50	1.78 x 10 ⁻¹				
1840	16	0.68	1.70 x 10 ⁻¹				
1860	25	0.76	1.51 x 10 ⁻¹				
1760	15	0.50	1.26 x 10 ⁻¹	1760	15	0.55	1.42 x 10 ⁻¹
1760	30	0.60	1.09 x 10 ⁻¹				
1760	60	0.69	8.95 x 10 ⁻²				
1760	177	1.09	8.20 x 10 ⁻²				
1650	75	0.38	4.11 x 10 ⁻²	1650	150	0.50	4.08 x 10 ⁻²
1650	150	0.43	3.52 x 10 ⁻²				
1650	350	0.61	3.26 x 10 ⁻²				
1550	100	0.17	1.74 x 10 ⁻²	1550	240	0.35	2.26 x 10 ⁻²
1550	200	0.22	1.55 x 10 ⁻²				
1390	200	0.06	4.38 x 10 ⁻³	1400	400	0.08	4.00 x 10 ⁻²
1400	400	0.10	5.0 x 10 ⁻³				

*Diffusion zone was the thickness of the phases that could be metallographically observed between the Nb and the 316 SS after the diffusion treatment.

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3) The concentration in the NaK is proportional to the first power of the pressure in the nickel tube as noted in Figure 13. This is in disagreement with recently published data of ORNL on the hydrogen-NaK system. Figure 14 compares a long extrapolation of the Oak Ridge data, obtained in a different experimental apparatus at higher hydrogen concentrations (<40 ppm), with the data obtained so far with the 1320°F capsule. In this case, hydrogen concentration is plotted as a function of the square root of pressure.

The above hydrogen concentration data were obtained by graphical integration of plots of hydrogen flow rate versus time. These curves were obtained during both the hydrogen addition and hydrogen removal steps of the experiments. The overall accuracy of the test data is estimated to be $\pm 20\%$.

4. Duplex Tubing Materials Study

The examination of specimens of thermally soaked, niobium-lined, 316 stainless-steel tubing was continued. These specimens had previously been sealed in Vycor capsules at a pressure of less than 10^{-5} mm Hg and exposed to temperatures from 1300 to 1960°F for periods ranging from 2 to 1000 hr. The purpose of the tests was to investigate the diffusion kinetics of the niobium-stainless-steel bond and measure the rate of growth of the diffusion zone adjacent to the bond. Tests were performed with both co-extruded and explosively bonded tubing. The results for the coextruded tubing were presented in the preceding quarterly. The combined results for both types of tubing are summarized in Table 4.

Analysis of the data demonstrated that a functional relationship existed between the diffusion rate and the specimen temperature. This relationship is defined by the equation

$$k = 7.95 \times 10^5 e^{-3.595 (10^4/T)}$$

where

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

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

Further analysis showed that (1) the correlation coefficient for this equation is -0.9943. This correlation coefficient indicates a very strong correlation between the diffusion reaction rate and the inverse of temperature, and (2) the coefficient of determination equals 0.9886, indicating that the regression equation accounts for 98.86% of the variance observed in the experimental data.

Figure 15 is a plot of the regression equation with appropriate confidence limits. The experimental data points are plotted on the same figure for comparative purposes. It should be noted that the diffusion reaction for the coextruded and explosively bonded tubing fit the same equation.

Flattening tests of a specimen with a metallographically observed 0.0002-in. diffusion zone show that the bond did not fail when the tube was crushed flat. The bond of a specimen with a 0.0005-in.-wide diffusion zone did fail after only a slight deformation. It is estimated that the useful lifetime of the tubing under mechanical and/or thermal loading can be defined by the thickness of the diffusion zone. A conservative lifetime estimate can be based on a 0.0002-in. diffusion zone thickness, an optimistic lifetime estimate would be based on a 0.0003-in. diffusion zone thickness.

Figure 16 relates the diffusion zone thickness to time at various temperatures. From this, estimates can be made as to the useful lifetime at temperature. The conservative limit would be 180,000 hr at 1200°F, 50,000 hr at 1250°F, or 15,000 hr at 1300°F. The optimistic limits would be 400,000 hr at 1200°F, 100,000 hr at 1250°F, or 35,000 hr at 1300°F.

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Figure 15. Functional Relationship Between the Diffusion Reaction Rate of Type 316 Stainless Steel and Niobium vs the Inverse of Temperature

5. Physical and Mechanical Properties of New Alloys

The primary objective of this project is to survey and evaluate new structural materials through definition of their properties and characteristics. This study was initiated in FY 66, and initial efforts have been directed toward preparation of the technical plan. The elements of the technical plan are as follows.

Figure 16. Metallographically Observed Diffusion Zone vs Time at Temperature for Nb and 316 SS

Structural materials can be categorized into lightweight alloys such as Be, Be-38 w/o Al (Lockalloy), Al, Mg, and Ti, high strength ferrous materials, such as maraging steels, ausformed steels, precipitation hardened steels, and high-temperature resistant alloys such as the Hastelloys, Inconels, TD-nickel, and Haynes 25.

It is planned to perform materials properties studies on Be-38 w/o Al (Lockalloy) and beryllium. Pure cross-rolled beryllium is currently available with mechanical properties superior to those previously attainable for this material. A limited hardware-oriented study should be initiated to determine its relevance for structural applications.

Laboratory investigations of potential materials are necessary to support and complement data gathered from vendors and literature. For instance, a literature survey on a new alloy will usually yield data on density, thermal expansion, tensile strength as a function of temperature, and methods of Tungsten-Inert Gas (TIG) welding. However, the literature is usually devoid of data on thermal conductivity, emittance, compatibility, fatigue strength, creep and stress rupture strengths, and alternate methods of joining.

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Investigation of the high-strength ferrous alloys will be limited to a literature survey until a thorough study of reactor requirements has been completed.

High-temperature resistant alloys are currently required for certain component applications such as springs and thermoelectric module parts. Investigation of new materials and their applications will be limited to literature surveys until a thorough review of structural requirements is completed.

Materials-oriented reviews of proposed structures, coupled with literature and vendor surveys, complemented by the two laboratory investigations and other limited testing will provide valuable support to advancing materials for structural applications. Information will be

obtained to provide effective utilization of these materials.

6. Corrosion Testing Techniques

The primary 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 which must 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. This investigation was initiated in FY 66. Effort has been started on a literature survey of liquid metal corrosion testing techniques.

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IV. EVALUATION OF EFFORT

The generation of R-Z synthesis basic equations is a necessary prerequisite to the preparation of an improved multi-group, two-dimensional transport theory nuclear code. This code, when completed, should reduce the computer time required for such calculations to a fraction of the time presently required.

Conversion of the 100 GT thermalization code to FORTRAN IV has decreased the required computer time and made the code compatible with scheduled changes in the NAA computer system.

The initial assembly and satisfactory operation of the GRISM cross section averaging code represents significant progress towards achieving improved multi-group cross section libraries. It provides not only the ability to more closely tailor the library to fit the properties of the analytical model, but also provides for more efficient handling of the data with less chance for user error.

The improvements effected in the UPSTART II code have increased the versatility and efficiency of this tool for the analysis of SNAP reactor system transient behavior.

The identification of the probable sources of error in the 21-group cross section library prepared for use with the DTF-2 code in shielding calculations should permit progress to be resumed in this area. If this calculational technique can be perfected, there will be less need for reliance upon expensive Monte Carlo calculations.

The calculation of the scattered neutron dose to be expected in the shield experiments scheduled for the ORNL Tower Shield Facility has demonstrated the need for provision of a "collar" shield in connection with these experiments.

The gamma attenuation experiments underway at the STIR are providing basic data on the attenuation properties of heavy metal slabs.

The preliminary results of silicon semiconductor irradiations in different radiation spectra continue to confirm the prediction that high energy gammas, such as those leaking from a reactor, are much more damaging than the relatively low energy gammas, such as Co^{60} , usually employed in radiation damage studies. If the analytical predictions continue to be borne out, it will be necessary to review present SNAP shielding criteria.

The identification and correction of the causes of data scatter in previous noise analysis measurements of flow velocity remove major obstacles to the application of this sensitive technique to hydraulic studies.

The study of mercury supersaturation behavior is making satisfactory progress considering the difficulty of the experimental problems involved. If the required experimental data can be obtained, it should permit a significant improvement in mercury turbine performance and reliability.

After resolution of initial difficulties associated with obtaining a suitable hydrogen input rate, the NaK-hydrogen partial pressure experiment proceeded in a satisfactory manner. Data are being gathered which will permit evaluation of the yttrium getter concept and better estimation of hydrogen loss rates from a SNAP system.

The study of niobium-lined 316 stainless-steel duplex tubing has provided basic data defining the service limits of this product in terms of time and temperature. Additional effort is needed to develop joining techniques for this material.

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V. OBJECTIVES FOR NEXT REPORTING PERIOD

The objectives for the next quarter are as follows

1) Nuclear and Control Analysis

- a) Publish a report describing the DTF-2 code
- b) Perform preliminary development of a computer program in R-Z synthesis using S_n transport theory and weighting functions obtained from the DTF-2 code
- c) Incorporate the extended Cohen-Vaughn neutron thermalization model into the 100 GT code and compare the calculated spectra with experimental spectra
- d) Develop an experimental program for the basic study of poison-backed control drums
- e) Complete GRISM code development.

2) Nuclear Experiments

- a) Complete experiments on control drum interaction and total shutdown worth using small SCA-5 cores
- b) Initiate fabrication of additional unpoisoned fuel pellets for the SCA-5.

3) Shield Analysis

- a) Complete evaluation of DTF code for shielding studies

- b) Continue analysis of experimental gamma attenuation data generated at the STIR

- c) Continue development of SHOE shield optimization code

4) Shield Experiments

- a) Complete experimental investigation of gamma attenuation by heavy metal slabs
- b) Initiate experimental investigation of gamma attenuation by composite shields consisting of heavy metal slabs and lithium hydride slabs in various combinations
- c) Complete refined determination of the STIR in-pool fast neutron spectrum for analysis of previous in-pool experiments. Complete analysis of preliminary data. Initiate planning for next STIR irradiations.

5) Heat and Mass Transfer

- a) Complete hydrogen partial pressure testing with 1320°F and 1260°F capsules. Initiate tests using 1175°F capsule
- b) Obtain supersaturation data for methanol in the prototype diffusion chamber. Continue assembly and checkout of mercury nozzle expansion loop
- c) Complete literature survey of corrosion testing techniques.

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