PACIFIC NORTHWEST LABORATORY
ANNUAL REPORT FOR 1973
ON CONTROLLED THERMONUCLEAR
REACTOR TECHNOLOGY
AEC RESEARCH AND DEVELOPMENT REPORT

Battelle

Pacific Northwest Laboratories
Richland, Washington 99352

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By

Staff of Battelle Northwest

April 1974

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PREFACE

The Annual Report for 1973 on Controlled Thermonuclear Reactor Technology consists of progress summaries of research conducted by the staff of Pacific Northwest Laboratories (PNL). The AEC Division of Controlled Thermonuclear Research is a major sponsor of the work. However, fusion-related work sponsored by others is also included to provide a more complete perspective of the research being conducted at PNL.

The summaries are presented in four major categories of:

- System Design and Analysis
- Materials Research and Radiation Environment Simulation
- Blanket Design
- Plasma Physics

At the beginning of each section is a brief summary of the reports making up the section. The reports themselves have been kept relatively short and include preliminary results which ultimately are expected to be published elsewhere. Because of this, the reader is cautioned that the results may be modified before they are finalized. In some cases, reference is made to more complete reports that are available now.

Loren C. Schmid,
Program Manager - Fusion

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ANNUAL REPORT FOR 1973 ON
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SYSTEMS DESIGN AND ANALYSIS

Economic studies concerned with systems analysis and preliminary design concepts can be useful in providing direction to the fusion program. To operate any economical controlled thermonuclear reactor one must be able to evaluate cost trends both of the device itself and of the competitive energy sources. Due to different rates of cost escalation, the usual analysis approach does not allow for differential cost factors to appear and influence selection of power plants. An effort is being made to explore the effect this weakness may have by looking at the long-term trends of the cost components of the various competitive fossil and nuclear energy sources.

The fusion-fission hybrid system has been identified as a special design concept of interest to the Pacific Northwest Laboratory. Based on work conducted thus far, this concept offers the most immediate promise for exploration of the fusion principle. One can envision a sub-Lawson plasma driving a sub-critical uranium blanket in which the total machine is generating more energy than is needed for operation. In addition, it should be possible to breed in a short time more fissionable and fusionable material than is consumed. A highlight of the hybrid work was the publication of the results of a review of all of the known studies in which high energy neutrons from a D-T fusion plasma were conceptually used to ultimately produce heavy element nuclear fission. In the review the deficiencies of the hybrid concept and needs for its future development were identified, and a number of potential uses of hybrid devices both before and after the attainment of pure fusion capability have been projected.

The objective of the present systems design work for the hybrid is the identification of major technological problems which must be solved in order to develop this potential source of power. As a result of the preliminary evaluation of the technology problems, it appears that many of the design constraints which limit fusion reactors can be relaxed in the case of the hybrid design. In addition, the problems of hybrid reactor design are in many cases similar to those of fusion reactor design but of reduced magnitude. Thus, the hybrid reactor benefits from progress in pure fusion design and conversely, the fusion reactor benefits from hybrid development.

In addition to the hybrid studies conducted in CY 1973, a cooperative study with Lawrence Livermore Laboratory was begun with the goal of defining the neutronic characteristics of a fission blanket constrained to the characteristics of a mirror confinement controlled thermonuclear reactor (CTR). The blanket concept which is being studied uses currently available gas-cooled reactor technology and a plant power output within the range of present turbogenerator designs. Other criteria embrace technology presently used in the laboratory or which is expected in the next generation of devices; thus, the design is conservative and could in principle be developed in the time span comparable to that needed to develop fission breeder reactors.

The large and intense neutron sources which are projected for the CTR open up the possibility that the storage time, radioactive hazard, or the amount of material resulting from operation of fission power reactors might be substantially reduced by nuclear transmutations. Calculations were made in which the thermal neutron flux was maximized by using a CTR blanket of beryllium since initial calculations demonstrated that maximum nuclear reaction rates might be achievable by this technique. Within the limits of the data and techniques used in this analysis, it appears that CTR transmutation is feasible and attractive. The fact that it is more attractive for the actinides than the fission products is fortunate because the actinides constitute the greater long-term hazard from the fission reactors.

In addition, it was concluded that the transmutation rate of fission products in a fusion reactor could be balanced with the production rate in a fission reactor of the same power. Thus, it was concluded that satisfactory quantities of waste could be transmuted to assist in solving the radioactive waste management problem generated by fission power reactors.

Economic Regimes

J. R. Young

The evaluation of the economic regimes in which any economical controlled thermonuclear reactor will operate depends critically upon many cost trends both of the device itself and of the competitive energy sources. The usual programming approach is limited in the forecasted cost structures because a base year constant dollar projection is used. This approach, due to different rates of cost escalation, does not allow differential cost factors to appear and influence selection of power plants. This fault can be argued to be more apparent than real since any large dislocation will eventually return to nearly the same equilibrium conditions by feedback into the industrial system and thus cause compensating induced differential price changes. It is not clear, however, that the system returns to the same equilibrium and thus the same relative price structure. An effort is being made to explore this problem by looking at the long term trends in the cost components of the various competitive energy sources.

Progress During CY 1973

The determinants of the fossil fuel prices are being explored. Currently, the changes due to air quality controls, mine safety regulations, severance taxes, reclamation requirements and numerous other factors have kept the price of power produced by coal changing too rapidly to permit supportable forecasts of future prices. The oil situation was seemingly more stable, but occurrences such as recent drastic price changes and shortages have negated earlier efforts to forecast the cost trends. These are being looked at again with some new variables added to perhaps make forecasting more reliable. The capital cost structure is currently being reexamined.

The capital costs and the costs of nuclear fission fuels and plants are currently being reevaluated along with some of the components, such as interest rates and labor costs. A cooperative effort with Oak Ridge National Laboratory in this effort is expected to yield relatively stable forecasts.

Some of these results, illustrating the variability of prices and financial conditions, are shown in Figures 1 through 6. It is apparent that some changes have occurred since these figures were drawn, and they are currently being reevaluated.

FIGURE 1. Historical and Projected Natural Gas Prices

FIGURE 2. Construction Cost Indices

FIGURE 3. Historical and Projected Coal Prices

FIGURE 4. Historical and Projected Crude Oil Prices

FIGURE 5. Bond Yields and Federal Reserve Discount Rate

FIGURE 6. Bond Prices and Yields 4% - 20 Yr, Aaa

Review of Fusion-Fission Hybrid Concepts

B. R. Leonard, Jr.

A review was made of all of the known studies in which the 14 MeV neutrons from a D-T fusion plasma were conceptually used to ultimately produce heavy-element nuclear fission. The results of this review have recently been published in Nuclear Technology.⁽¹⁾ In Table 1 the results of the neutronics analyses of hybrid concepts which have been carried out with modern computational methods are summarized and compared with a pure fusion D-T lithium blanket.

TABLE 1. Summary of Blanket and Plasma Characteristics of Neutronic Analyses of Hybrid Concepts

Characteristic	Pure Fusion	Lontai ⁽²⁾	Lidsky ⁽³⁾	Lee ⁽⁴⁾	PNL ⁽⁵⁾
Tritium production (per source neutron)	~1.4	~1.2	1.005	1.18	1.06
Fissile production (per source neutron)	---	~0.1	0.4	2.49	2.7
Blanket fissions (per source neutron)	---	~0.2	0	1.57	2.6
Blanket energy (per source neutron)	18.3	~40	22.4	306	500
Plasma characteristics					
n_i (cm ⁻³)	---	---	10^{14}	2.7×10^{14}	3.5×10^{13}
τ (sec)	---	---	0.625		1
Fraction of Lawson	≥ 1.0	≥ 1.0	0.5 - 0.8	~1/4	~1/6
T_i (keV)	---	---	20	60	10
14 MeV wall load (MW/m ²)	0.85 - 10	5	1	12.8	0.05

In addition, the review identifies several deficiencies of existing hybrid studies:

- Limited investigation of types of fusion devices
- Limited studies of possible blanket concepts
- Limited studies of specific intents of design
- Blanket temperature effects in operation
- Blanket neutronics with exposure
- Fuel cycle analysis, fissile recycle
- Engineering studies

A number of potential uses of hybrid devices, both before and after the attainment of pure fusion capability, have been projected. These are summarized in Table 2.

TABLE 2. Projected Potential Uses of Hybrids

Assumption	Uses
I. <u>Near Term</u> Lawson condition not achieved or achieved but technological problems	I. 1. Energy breakeven device to study problems of fusion reactor 2. A breeder of fissile material 3. A subcritical power plant
II. <u>Long Term</u> Pure fusion capability	II. 1. A breeder of fissile material for existing fission plants 2. A breeder of tritium for nontritium-producing fusion plants 3. A burner of plutonium and actinide waste of fission power economy

A Preliminary Evaluation of Major Hybrid Technology Problems

W. C. Wolkenhauer

Previous analyses of the fusion-fission hybrid at PNL have concentrated on the neutron physics properties of these reactors. The objective of the present work⁽⁶⁾ is the identification of major technological problems which must be solved in order to develop this potential source of power. In order to identify major needed development efforts, surveys have been performed in areas of thermal hydraulics, materials damage, moderator characteristics and tritium extraction.

While there are several different proposed hybrid configurations, the machine under study here consists of a D-T thermonuclear plasma surrounded by a fissile blanket. The fissile blanket is subcritical and the plasma is sub-Lawson. The lattice of the blanket chosen for study was a graphite-moderated helium-cooled lattice, fueled with natural or very slightly enriched uranium. The results of the neutronics study indicate that this particular hybrid exhibits a high potential as a viable power plant.

Results

The following conclusions have been reached relative to needed technological development of a fusion-fission hybrid reactor:^(a)

- In order to achieve a thermodynamic efficiency in the candidate design comparable to present fission power reactors, either a non-metal fuel must be used or the diameter of the fuel must be reduced from the initial design.
- It is attractive to consider employing the convertor region as a pre-heater (in the thermodynamic sense) for the lattice region.
- For pulsed operation of this design, the total cycle time should be less than 1 second for a pulse time of ~ 0.1 second, or less than 2 seconds for a pulse time of ~ 0.5 seconds.
- 316 stainless steel could be used as a first wall and structural material in the proposed hybrid blanket design. Its known

a. This work was conducted early in CY 1973 and describes a different design than that covered on the following page.

radiation damage limits are such that a 20 year design life seems reasonable. Wall erosion by sputtering or blistering does not appear to present a serious limitation. At the design temperature, helium embrittlement does not appear to be a problem.

- The radiation effects in graphite for the proposed hybrid design are relatively mild compared both to the pure fusion systems which are being conceived and to current graphite-moderated reactors.
- Corrosion between various components of the system (lithium, helium, stainless steel, etc.) does not appear to be a limiting phenomenon although a provision for removal of corrosion products to prevent fouling may be required.
- Tritium containment at acceptable levels appears possible with the use of stainless steel under the reference design conditions with the use of frequent extraction. Infrequent extractions can be employed if tungsten is used as the barrier.
- Considerable development work is required in the area of tritium extraction. In this instance, there appears to be no clear differentiation between the hybrid and the fusion reactor.

In summary, it would appear that many of the design constraints which limit fusion reactors can be somewhat relaxed in the case of the hybrid design. In addition, the problems of hybrid reactor design are, in many cases, similar to those of fusion reactor design in kind, if not in degree. Thus, the hybrid reactor benefits from progress in pure fusion design and, conversely, the fusion reactor benefits from hybrid development.

Study of a Fission Blanket for a Mirror Confinement CTR

W. C. Wolkenhauer, B. R. Leonard, Jr., D. F. Newman and U. P. Jenquin

A cooperative study with Lawrence Livermore Laboratory (LLL) has been started which has the goal of defining the neutronic characteristics of a fission blanket constrained to the characteristics of a mirror confinement CTR. The basic design characteristics of the Ying-Yang geometry mirror concept are extensions of a preliminary study done at LLL.

Results

The PNL fission blanket concept which is being studied uses currently available gas-cooled reactor technology and a plant power output within the range of present turbogenerator designs. The plasma, injector and magnet designs embrace technology presently used in the laboratory or which is expected from the next generation of devices. This assumes that the D-T plasma confinement remains within a small factor of the classical value. With this assumption, the design is conservative and could, in principle, be developed in a time span comparable to that needed to develop fission breeder reactors.

Initial calculations indicate the following characteristics:

- The device will employ a sub-Lawson ($n\tau \approx 3 \times 10^{13} \text{ cm}^{-3} \text{ sec}$) plasma.
- The fissile blanket will be subcritical by a fraction which is large compared to the delayed neutron fraction.
- The device will have an overall output-to-input energy ratio substantially greater than unity.
- The device will breed both fusion fuel (tritium) and fissile fuel (plutonium).

Neutron Transmutation of Radioactive Waste in the Blanket of a CTR^(a)

W. C. Wolkenhauer, B. F. Gore and B. R. Leonard, Jr.

The radioactive waste generated in fission power reactors must be stored, safeguarded or otherwise disposed of or protected from the biosphere for geologic time periods. The large and intense neutron sources which are projected for CTRs open up the possibility that the storage time, radioactive hazard, or amount of material might be substantially reduced by nuclear transmutations induced by neutrons in the blanket of a CTR. The feasibility and viability of this process depend upon the transmutation process leading to a reduction of radioactive waste, including progeny nuclei, in a time interval which is short compared to the natural radioactive half-life.

Progress in Cy 1973

Initial calculations demonstrated that the maximum nuclear reaction rates were obtainable by maximizing the thermal neutron flux, even for fission product nuclei with the smallest thermal neutron absorption cross-section values. Thus, calculations were made in which the thermal neutron flux was maximized by using a CTR blanket of beryllium. Calculations were performed for two different plasma neutron source strengths for both D-D and D-T plasmas for several fission product isotopes and elements and for actinide samples.

The results of the analysis are shown in Table 3 in terms of the hazard half-life. The hazard half-life is defined here as the time required for the hazard from a particular isotope and its daughters to decay to half its original value.

The following conclusions have been reached as a result of this analysis:

1. Fission product transmutation exhibits hazard index reduction rates which are attractive at neutron wall loading judged achievable in a CTR (i.e., $P_W = 10.0 \text{ MW/m}^2$).

a. Results of work funded by USAEC Division of Waste Management and Transportation

TABLE 3. Hazard Half-Life (years) as a Function of Neutron Wall Loading (P_w)

Target	Plasma	$P_w = 0.0$	$P_w = 1.0 \text{ MW/m}^2$	$P_w = 10.0 \text{ MW/m}^2$
^{90}Sr	D-T	32.0	7.6	2.4
$\text{Sr}^{(a)}$	D-T	32.0	8.0	7.4
^{90}Sr	D-D	32.0	4.7	0.48
^{137}Cs	D-T	30.0	19.0	5.6
$\text{Cs}^{(a)}$	D-T	14.0	12.0	4.0
^{137}Cs	D-D	30.0	16.0	4.4
^{85}Kr	D-T	11.0	8.6	6.1
^{129}I	D-T	1.6×10^7	0.35	0.027
Actinides	D-T	460 ^(b)	0.60	0.056
Actinides	D-D	460 ^(b)	0.23	0.037

a. This is isotopic mixture of Sr or Cs from LWR fuel at 33,000 MWd/MT and 10 years cooling.

b. ^{241}Am only.

2. Strontium-90 is the most attractive fission product transmutation candidate with ^{137}Cs being the next. Iodine-129 is a possible candidate. Krypton-85 should probably be eliminated as a candidate.

3. Isotopic separation is not attractive for Cs. It will be attractive for Sr only if it is quite cheap.

4. The D-D plasma is more effective for fission product transmutation than the D-T plasma for the same values of neutron energy wall loading. This is particularly true for ^{90}Sr transmutation.

5. Actinide transmutation in a CTR is much more attractive than fission product transmutation, notwithstanding the previous conclusions. Actinide transmutation appears to remove the problem in an extremely small fraction of the required storage time.

6. As in the case of fission products, the D-D plasma is a more attractive transmutation source for actinides than the D-T plasma for similar wall loadings.

7. In a D-T CTR, the plutonium transmutation hazard index for LWR plutonium ultimately decreases. This phenomenon offers a potential solution to the residual plutonium waste generated by operation of a fission economy. It therefore appears that a CTR can potentially transmute most of the high-level wastes from a fission economy into stable or short half-lived ash.

In summary, within the limits of the data and techniques used in this analysis, it appears that CTR transmutation is feasible and attractive, although it is more attractive for the actinides than the fission products. This is fortunate in that the actinides constitute the greater long-term hazard.

The detailed results of this study are contained in a BNWL report⁽⁷⁾ and have established some bounds on the maximum specific nuclear reaction rates which might be achieved. A further study was made to establish that the quantity of radioactive waste which might be reduced by transmutation would have a substantial impact on the total inventory of the waste. The fission product isotope ^{137}Cs was chosen for the transmutation target for this study. In this study, the inner 20 cm of blanket beryllium was progressively replaced by ^{137}Cs . It was found that the calculated specific transmutation rate of the ^{137}Cs was not very sensitive to ^{137}Cs concentration. For an 80% volume fraction of pure ^{137}Cs with a 1 cm thick Nb inner wall, the calculated transmutation rate was 290 kg/m of CTR length with an effective ^{137}Cs half-life of 9.9 year. This rate is about 1% of the length of the CTR blanket would balance the ^{137}Cs production rate in a fission reactor of the same power. Thus, it was concluded that satisfactory quantities of waste could be transmuted. [A summary of the results of this study was presented in an American Nuclear Society Meeting paper⁽⁸⁾ and has appeared as a Technical Note in Nuclear Science and Engineering⁽⁹⁾].

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MATERIALS RESEARCH AND RADIATION ENVIRONMENT SIMULATION

Knowledge of high energy neutron interaction with materials is a high priority need in the development of a controlled fusion power plant. Many of the CTR concepts currently being developed contain graphite in the blanket and/or shield regions for the purpose of moderating and reflecting the high energy neutrons. As the graphite interacts with the high energy neutrons, it experiences considerable radiation damage which results in large property changes. As part of our emphasis on the engineering aspects of CTR designs, we have projected graphite behavior for four design concepts. It was learned that reasonable graphite lifetimes can be expected for all of the designs except for the Theta Pinch reactor which operates at a first wall neutron loading of 3.5 MW/m^2 . A reduction of the wall loading to 1 MW/m^2 is expected to result in a more reasonable graphite lifetime for that reactor.

Consideration of the removal of atoms from the surface of the metal by sputtering is important at high power loadings on the vacuum walls. In particular, any reduction of thickness may produce unacceptable stress levels in the walls. The effect of impurities and radiation damage on the sputtering coefficient is not well known at this time. Thus, an experiment has been designed to experimentally determine the fast neutron-induced sputtering yields for materials which are likely candidates for use in thermonuclear reactors. Plans for the initial experiments have been completed, and much of the necessary preparatory work was carried out.

The radiation damage to the vacuum wall from high energy neutrons will be quite severe with each atom being theoretically displaced from its lattice position several thousand times during a reasonable reactor lifetime. This disruption of the metal lattice will be accompanied by the generation of large amounts of helium and hydrogen atoms which will fully perturb the mechanical and dimensional stability of the wall materials. Unfortunately, there is no way that such a damage state can be duplicated in a reasonable time using present day research reactors. Therefore,

scientists at PNL have been turning to high energy, heavy ion bombardment studies in order to achieve high damage rates in a shorter time. Using this technique, the swelling produced by heavy ion bombardment is currently being studied in high purity molybdenum. The results obtained so far show that the swelling is not strongly dependent upon the dose and that the maximum swelling is only about 3% after a dose of 150 displacements per atom. In addition, comparisons of ion bombardment data with neutron irradiation data indicate that heavy ion bombardment does give a reasonable simulation of neutron irradiation damage.

The materials properties criteria are particularly stringent in the case of the Theta Pinch reactor since the insulator must have a high electrical resistivity, a high dielectric strength, good thermal conductivity, and must retain these properties at temperatures of 800°C within a thermal gradient and after prolonged pulse irradiation. Electrical insulators are also used in various critical locations in other designs such as those employed in Mirror and Tokamak devices. Therefore, an electrical insulator program is directed toward providing the critical property data necessary for choice of optimum insulator materials. As an initial phase the various insulator requirements and the specific insulator environments have been defined for each of the CTR designs. Potential insulator materials which have the appropriate properties have also been reviewed for potential use in the CTR.

Sputter deposition may be the best method for producing inorganic insulator coatings. Thus, a program was initiated to establish whether or not high quality electrical insulating coatings can be fabricated by high rate RF sputter deposition. Several insulator deposits of Y_2O_3 and Al_2O_3 were made by that technique. The results so far are sufficiently encouraging to indicate eventual success at producing high-quality deposits for properties studies of phenomena such as crystal structure, grain size, texture, density, and oxygen-metal chemical ratio.

Gas cooling offers an important alternate and backup to liquid metal cooling of a CTR. The most likely coolant gas is helium, which is inert to

metals. However, experience has shown that impurities in helium at concentrations below present detection limits cause considerable reaction with active metals, such as niobium. A study has been initiated to provide a systematic assessment of data on helium coolant technology and refractory metal compatibility so that the role of helium cooling in fusion reactors can be put into perspective.

The decay of tritium to helium offers an important approach to study the embrittlement of candidate CTR first wall materials caused by helium. The tritium trick method is based on this principle and is used to inject helium, generated from the tritium decay, into alloys having substantial hydrogen solubilities. Research is being conducted to develop and verify the tritium-trick method using candidate CTR structural materials of niobium and vanadium alloys. The advantage of the method is that it can be used to prepare mechanical property specimens to assess damage effects in metals caused by helium apart from other radiation damage effects.

The power output for fusion reactors is expected to increase with the fourth power of the confined magnetic field. Thus, development of methods to fabricate magnet linings from the very brittle niobium-aluminum-germanium or niobium-aluminum superconductors may be essential to economical fusion reactor operation since they have the highest known critical fields. The cost of high field superconductors made by sputter depositions is expected to be quite reasonable and methods used for making superconductors using this technique are being developed and evaluated. A 4-ft length of sputtered niobium-aluminum-germanium on stainless steel was fabricated to demonstrate feasibility, and research is also proceeding to make filamentary conductors by coating extremely fine wires with sputter-deposited materials.

Graphite for Controlled Thermonuclear Reactors

W. J. Gray and W. C. Morgan

Many of the CTR concepts currently being developed contain graphite in the blanket and/or shield regions for the purpose of (1) moderating the high energy neutrons and reflecting them back into the tritium breeding portion of the blanket, and/or (2) moderating the high energy neutrons so that they can be more readily absorbed before penetrating the superconducting magnet regions. As the graphite moderates the high energy neutrons, it experiences considerable radiation damage which results in large property changes. Dimensional changes are potentially the most troublesome, and these changes are shown for a typical nuclear-grade graphite in Figure 1. For a thorough discussion of these and other property changes, the reader is referred to the literature⁽¹⁻⁵⁾.

As part of the increasing emphasis on the engineering aspects of CTR designs, we have projected graphite behavior in four design concepts: the Oak Ridge National Laboratory (ORNL) Conceptual Fusion Reactor Version III;⁽⁶⁾ the University of Wisconsin Tokamak (Austin Design);⁽⁷⁾ the Los Alamos Scientific Laboratory (LASL) Reference Theta Pinch Reactor;⁽⁸⁾ and the PNL Fusion-Fission Hybrid Reactor.⁽⁹⁾ These projections were made by calculating the relative numbers of atomic displacements in the fission and fusion reactor spectra and relating these to property changes observed for graphite samples irradiated in fission reactors.

Basis of Projections

Correlations between the relative numbers of atomic displacements in the fusion and fission reactor spectra were made using the Lindhard theoretical treatment of atomic stopping cross sections and neutron scattering cross section data from ENDF/B-III (Evaluated Nuclear Data File). Although the Lindhard treatment has not been experimentally verified at the higher neutron energies, even an error of a factor of two in this region would result in no more than $\pm 35\%$ error in our projections since only about 25 to 35% of the displaced atoms in the graphite of the present

CTR designs are a result of neutrons with energies greater than those of importance in fission reactor irradiations.

We have defined the lifetime of the graphite in CTR applications as that fluence at which the graphite has contracted and then expanded back to its initial dimension in any direction (see Figure 1). While this fluence is not an absolute upper limit to the graphite's lifetime, graphite begins to lose strength during the expansion phase because of the fine-scale cracking between and through the individual graphite particles. Thus, if graphite plays no structural role, an increase in lifetime of perhaps 20 to 40% beyond what we have projected could be realized.

Results

Our projections of graphite temperatures and lifetimes as well as other pertinent reactor parameters are given in Table 1. It can be seen that reasonable graphite lifetimes are expected for all but the LASL Theta Pinch Reactor. In that reactor, the first wall neutron loading is 3.5 MW/m^2 and it might be expected that most reactor concepts with first wall loadings above about 1 MW/m^2 would have short graphite lifetimes.

TABLE 1. Pertinent Reactor Parameters^(a)

	<u>ORNL</u>	<u>University of Wisconsin</u>	<u>LASL</u>	<u>PNL</u>
Type	Tokamak	Tokamak	Theta Pinch	Fusion- Fission Hybrid
Power (MWt)	1000	1140	3600	Not specified
Minor Radius (m)	3.5	2.5	0.3	2.0
Major Radius (m)	10.5	12.5	56	Not specified
Wall Loading (MW/m ²) (14.1 MeV Neutrons)	0.43	0.53	3.5	0.0476
Distance First Wall to Graphite (cm)	30	50	10	8
Graphite Thickness (cm)	40	20	22	150
Graphite Lifetime (Full Power Days)	~2500	~7500	150-250	>10,000
Graphite Temperature (°C)	1000-1100	400-500	800-1350	400-500

a. These parameters are given by References 6 through 9 respectively.
More recent versions of these CTR concepts may be different.

FIGURE 1. Length Changes of Typical Nuclear-Grade Graphite

Fast Neutron Sputtering Rates

O. K. Harling and M. T. Thomas

The objective of this work has been to experimentally determine the fast neutron-induced sputtering yields for materials which are likely candidates for use in thermonuclear reactors. Emphasis was placed on materials for structural components which will be exposed to the highest fast neutron fluxes, e.g., the inner plasma containment vessel.

The primary containment vessel for the plasma in a controlled thermonuclear reactor (CTR) will be exposed to high fluxes of energetic neutrons, ions, neutrals, as well as photons of a wide range of wavelengths. These radiations will produce damage both at the surface of the containment vessel or vacuum wall and inside the blanket which surrounds the fusing plasma. One important damage mechanism is likely to be neutron-induced sputtering. Neutron sputtering will erode the containment vessel wall and other surfaces such as the walls of the pipes and ducts inside the blanket. Several undesirable effects can result from this type of erosion including:

1. Thinning of the vacuum wall to a point where it has inadequate mechanical strength
2. Ejection of high atomic number material into the plasma with resultant losses of plasma energy
3. Removal of radioactive species from surfaces and the possibility that these will be transported to other locations where they will cause a radiation hazard
4. The possibility of increased corrosion from coolants if the sputtering acts to expose a continually freshened surface to the corrosive action of the coolant.

Great uncertainty exists concerning the actual magnitude of sputtering from candidate CTR materials. Relatively few measurements have previously been made of fast neutron sputtering yields, and those which are available disagree by several orders of magnitude. For gold, where there are several measurements, the experimental sputtering ratios⁽¹⁰⁻¹⁵⁾ range from 0.2×10^{-5} to 500×10^{-5} . For iron, a sputtering ratio of 600×10^{-5} has been

reported⁽¹⁶⁾ while a preliminary result for niobium of (0.25 ± 0.1) has recently been obtained.⁽¹⁷⁾ The results in Reference 17 indicate that most of the sputtered material is emitted in micron-sized "chunks" rather than atomically as in the case for ion-beam sputtering. Significant disagreement in experimental sputtering coefficients, a general inadequacy of theoretical understanding, and the likelihood that neutron sputtering yields are a strong function of microstructure, make it imperative to carry out a wide variety of carefully designed experiments.

Progress During CY 1973

The neutron sputtering program was initiated at PNL in October 1973. Program work activities have been divided into a number of major tasks. Those tasks which were completed or begun in CY 1973 are listed below along with major subtasks and their status.

Task No. 1 - Design Considerations for S_n Experiment

Status - completed for initial experiments

Subtasks

- a - Overall program objectives and plan
- b - Choice of candidate sample materials for initial experiments
- c - Determination of desired microstructural sample properties
- d - Survey of available fast neutron irradiation facilities,
Choice of source for initial sputtering experiments
- e - Environmental conditions and surface condition of samples
- f - Choice of sample analysis techniques and methods of sample
characterization and preparation

Task No. 2 - Design of Irradiation Apparatus

Status - completed

Subtasks

- a - Design of sample processing system
- b - Design of irradiation chamber
- c - Design of sample transfer system

Task No. 3 - Construction of Apparatus for S_n Experiment

Status - in progress

Task No. 4 - Development of Techniques for Neutron Activation Analysis of Sputtered Material

Status - in progress

Subtasks

- a - Selection of detection reactions
- b - Determination of sensitivities
- c - Background determinations for candidate collector materials
- d - Development of measurement techniques

Task No. 5 - Characterization and Preparation of Samples and Collectors

Status - in progress

Subtasks

- a - Auger electron spectroscopy to determine surface condition
- b - Scanning electron microscopy to characterize surface topography
- c - Contamination free handling and preparation

Progress in CY 1973

The neutron sputtering program was initiated during the final three months of CY 1973. Detailed design of initial experiments was completed and much of the necessary preparatory work was carried out.

Work currently scheduled consists of:

- Design of an experiment for the measurement of absolute yields from fast neutron sputtering
- Construction and testing of an irradiation chamber needed for the neutron sputtering measurements
- Development of the irradiation and the analytical techniques needed to evaluate the sputtering yields from several candidate CTR wall materials
- Preliminary measurements of sputtering ratios using a 14 MeV neutron source

Irradiation-Induced Swelling in Materials^(a)

J. L. Brimhall, E. P. Simonen and H. E. Kissinger

Work is continuing on the use of high-energy, heavy ions to simulate high neutron dose conditions.⁽¹⁸⁾ The first vacuum wall of a CTR will experience very high neutron exposures. The microstructural damage produced by the high energy ions is being studied as a function of irradiation dose and temperature. Of particular importance to materials' use in a fusion device is the amount of material swelling produced by irradiation. This swelling is a result of the formation of microscopic voids which can be analyzed by electron microscopy.

Progress in CY 1973

The swelling produced by heavy ion bombardment is currently being studied in high purity molybdenum. Molybdenum or a molybdenum alloy is one of the refractory metals under consideration as a vacuum wall material. The swelling in molybdenum at $950 \pm 25^\circ\text{C}$ is shown as a function of dose in Figure 1. Two important conclusions can be made from the figure. First, the results show that the swelling is not strongly dependent on the dose. In the expression $\% \Delta V/V = K (\text{dose})^m$, the exponent m is of the order $2/3$. This is a weaker dose dependence than found in stainless steel or in some other pure metals. Secondly, the maximum swelling is only about 3% after a dose of 150 dpa (displacements per atom). For the University of Wisconsin CTR design, this would correspond to the damage produced after approximately 8 years of operation. British workers at Harwell, doing similar work, found swelling values on molybdenum an order of magnitude lower than these.⁽¹⁹⁾ The reason for the differences in the results remains to be explored. However, impurity effects provide a possible explanation.

It must be remembered that in comparing ion bombardment data with neutron irradiation data, a temperature shift must be applied in order to account for the large differences in dose rate. An ion bombardment temperature of 950°C corresponds to about 700°C in neutron irradiation. The

a. Some results reported here have been made possible under a USAEC Division of Physical Research program.

highest doses from neutron irradiation data correspond only to the lower end of the curve in Figure 2. However, the magnitude of the swelling at these dose levels (0.1 to 0.5%) is comparable when the temperature shift is applied (see Reference 20 for neutron data). Therefore, data accumulated so far indicate that heavy ion bombardment does give a reasonable simulation of neutron irradiation damage.

FIGURE 2. Calculated Swelling as a Function of Dose for High Purity Molybdenum, Bombarded with 7.5 MeV Ta^{+++} Ions at a Temperature of $950 \pm 25^\circ C$.

An Evaluation of Electrical Insulators for
Controlled Thermonuclear Reactors^(a)

J. L. Bates and R. L. Gordon

The purposes of this joint Battelle-Columbus and Battelle-Northwest Laboratory study are (1) to define the electrical insulator requirements associated with controlled thermonuclear reactors (excluding the laser/relativistic beam), (2) to review the available literature on electrical, thermal, and irradiation properties and fabrication data for electrical insulators, (3) to determine the extent that these data can be applied to CTR requirements, in particular to the Theta-Pinch reactor, and (4) to recommend areas of needed research.

Progress in Cy 1973

The various insulator requirements for the Theta-Pinch, Mirror, and Tokamak designs and the specific insulator environments for each have been defined. Some of these requirements were difficult to assess because of the lack of specific design data.

Potential insulators which might be used have been screened and the list of materials to be considered reduced to a workable number. These include Al_2O_3 , MgO , BeO , Y_2O_3 , HfO_2 , some silicates, and aluminates (e.g., $3\text{Al}_2\text{O}_3 \cdot \text{SiO}_2$, $\text{MgO} \cdot \text{SiO}_2$, $\text{ZrO}_2 \cdot \text{SiO}_2$, $\text{MgO} \cdot \text{Al}_2\text{O}_3$, $3\text{Y}_2\text{O}_3 \cdot 5\text{Al}_2\text{O}_3$) and three nitrides - AlN , Si_3N_4 , and BN .

The literature survey has been completed and a review is now being made of the data.

a. Funded under a joint Battelle-Columbus/Battelle Northwest Program with the USAEC Division of Controlled Thermonuclear Research.

Sputter-Deposited Insulators

S. D. Dahlgren, N. Laegreid and R. W. Moss

The objective of the work on sputter-deposited insulators is to establish whether or not high-quality electrically insulating coatings can be fabricated by high-rate rf sputter deposition. The first task is to sputter deposit 13-mil-thick Al_2O_3 or Y_2O_3 coatings by the supported plasma rf-sputtering methods recently developed at PNL. Such insulators or coatings are essential to successful operation of the Theta Pinch Reactors.

Progress in CY 1973

The following paragraphs describe results of (1) preliminary work with low-rate rf sputtering that encouraged further work at high rates, and (2) the progress to date on high-rate rf sputtering using plasma-sprayed and sintered Al_2O_3 sputtering targets.

Several insulator deposits were made using low-rate diode rf sputter deposition. Coatings of Y_2O_3 were simultaneously deposited on niobium and molybdenum substrates maintained at about 400°C. The deposit on Nb adhered well to the substrate, was damage resistant, and had an excellent appearance, but the deposit on the Mo substrate did not adhere. Additional deposits (0.2-mil thick) of Y_2O_3 were made on Nb substrates. Unless the test probes penetrated the coating, no conductivity could be observed using elementary two-probe techniques. Again the deposits had excellent appearance and were determined to be transparent and crystalline.

Two Al_2O_3 deposits were made at high rates (about 0.5 mils/hr) from a low-purity plasma-sprayed target by the supported plasma rf-sputtering method. One deposit was made in a Cu substrate held at room temperature, and in the second deposition four Nb substrates were coated simultaneously. Oxygen-gas was added to the Kr sputtering gas during sputtering to maintain stoichiometry in the Al_2O_3 deposits. The deposit on copper was 1-mil thick. It was optically transparent and not crystalline. Heat treatment at 900°C caused crystallization into the cubic, gamma-alumina crystal

structure, but without any preferred deposit texture appearing. The deposit on four Nb substrates was 6-mils thick and was transparent. By varying heat transfer to the four substrates, the substrate temperature ranged from 25°C to 800°C. Unfortunately the coatings were damaged by an additional coating of iron from the target backing plate because the thickness of the flame-sprayed Al_2O_3 target received from the vendor was considerably less than specified.

High-rate deposition experiments with sintered high-purity Al_2O_3 and Y_2O_3 targets revealed target mounting problems which are currently under investigation. High purity is needed for optimum insulating properties. The targets were suspended in a holder from the top of the sputtering system so that substrates were located below the targets. High-rate sputtering caused considerable target heating, resulting in fractured targets before reasonable deposit thicknesses could be obtained. Equipment is presently being built in which the target-substrate relationship is inverted, thereby avoiding catastrophic failure due to target fracture.

The results so far are sufficiently encouraging to indicate eventual success in producing high-quality deposits for properties studies. In the future, appropriately thick, high-purity, inorganic insulator deposits will be made on Mo and Nb substrates at several temperatures. The deposits will be analyzed for crystal structure, grain size, texture, density, and oxygen-metal chemical ratio. The adherence of the deposits is expected to be determined in thermal cycling tests.

Compatibility of Refractory Metals with Helium in CTR Applications

L. A. Charlot and R. E. Westerman

The use of a gas coolant in fusion reactors has some attraction as an alternative to liquid metal coolants in CTR design. Helium, which is inert to metals, is the most likely choice as the gas coolant. However, reaction of impurities in the helium coolant with candidate structural metals, e.g., niobium, niobium alloys or vanadium alloys, may significantly affect the choice of coolant. A systematic assessment of data on helium coolant technology and refractory metal compatibility is essential so that the role of helium cooling in fusion reactors can be put into perspective. The objective of this study is to provide such an assessment.

This study was initiated late in the calendar year 1973 and will emphasize (1) data taken from the literature dealing with the thermodynamics of reaction of carbon, oxygen and nitrogen with refractory metals, principally niobium and niobium alloys, (2) the kinetics of reaction, (3) the effect of C, O, N contaminants on the mechanical properties of niobium alloys and (4) the current state of helium-coolant technology related to contaminant or impurity levels, detection and control. This literature data will then be used to assess contamination of a niobium conduit-helium circuit used to cool first wall and blanket areas of a CTR design.

Helium Embrittlement in CTR Materials
Using the Tritium Trick Method

A. B. Johnson, Jr., R. E. Westerman and L. A. Charlott

Helium embrittlement is one of the radiation-induced damage mechanisms likely to limit the life of CTR structural materials. The so called tritium trick is one method that can be used to inject ^3He , generated from the tritium decay, into alloys having substantial hydrogen solubilities. The advantage of the method is that it can be used to prepare mechanical property (tensile) specimens to assess helium damage effects in metals, apart from other radiation damage effects. The present study has, as one major goal, the development and verification of the tritium trick method on candidate CTR structural materials, niobium and vanadium alloys, that have a moderate to high affinity for tritium.

Progress in CY 1973

Data in the literature and preliminary experimental data on both niobium and vanadium have been used to establish the pressure, temperature and composition conditions for the tritium charging of buttonhead tensile specimens. The literature has also been reviewed to select strain rate and test temperature for the tensile testing of ^3He -doped niobium specimens.

Commercial purity niobium and vanadium rod stock was obtained for this study. Twenty-five miniature buttonhead tensile specimens have been machined from a part of the niobium rod stock. Tritium charging of these specimens is scheduled to begin February 1974 when a 2000 Ci inventory of tritium gas will be received.

The experimental procedure will be to stabilize the metal structure by a 1050°C vacuum anneal, tritiate the niobium specimens at 500°C, allow the tritium to decay to ^3He concentrations ranging from 20 to 400 appm, remove the excess tritium by a high temperature vacuum offgas, and then determine the high temperature tensile properties of the ^3He -doped

specimens. The charging/discharging of tritium has been performed on one vanadium specimen.

During May 1973 a vanadium buttonhead tensile specimen was tritiated at 500°C under an initial T_2 pressure of 10 torr. Total tritium pickup by the vanadium amounted to 4.6×10^3 appm. At this time two vanadium tensile specimens were deuterided under the same temperature and pressure conditions to serve as test controls. These specimens have recently been offgassed to remove the excess tritium or deuterium. The tritiated specimen has a calculated ^3He concentration of 166 appm. The high temperature tensile properties of these specimens are expected to be determined next.

Sputter Deposition of High Field Superconductors
For Fusion Reactor Magnets^(a)

S. D. Dahlgren

The sputter-deposition process permits Nb-Al-Ge to be considered for high field applications in the confinement magnets of controlled thermonuclear reactors. In the past, high critical temperatures and high critical fields have been reported for Nb-Al-Ge, but critical current densities were low. High critical current densities, however, have been found for Nb-Al-Ge superconductors made by sputter deposition.

The high current densities in sputter-deposited Nb-Al-Ge are attributed to small grain size. Current densities up to 4.4×10^5 A/cm² at 100 kOe and 4.2°K (Figure 3) were found for samples having grains of about 350Å diameter. Furthermore, the current density was not sensitive to magnetic field orientation, as is often observed; insensitivity to orientation is important for magnet applications where the magnetic field lines cross the magnet conductor at odd angles.

Progress in CY 73

Methods have been developed over the past year to make a sputter-deposited ribbon superconductor, and a 4-ft length of sputtered Nb-Al-Ge on stainless steel was fabricated to demonstrate feasibility. Longer lengths are expected to be made soon using the same equipment. Research is also proceeding to make filamentary conductors by coating extremely fine wires with sputter-deposited Nb-Al-Ge or Nb-Al. Filamentary conductors are needed where high rates of magnetic field change are desired.

Sputter-deposition results for Ni-Al-Ge are expected to be duplicated for other high-field superconductor compounds having the same crystal structure, e.g., Nb₃Al. The cost of high-field superconductors made by sputter-deposition is expected to be quite reasonable.

a. Result of work funded by Division of Physical Research USAEC and PNL internal funds.

FIGURE 3. Critical Current Density versus Transverse Magnetic Field for Specimens from a 66- μ -Thick Sputter-Deposit.

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BLANKET DESIGN

Efforts to assist in the design of fusion blankets are being carried out at PNL to achieve two major goals. One goal is the general development of experimental data, evaluated high energy neutron cross section sets, and evaluated analytical models for CTR blanket design. The other goal is the application of the developed data and analytical models to the design of tritium breeding graphite blankets and subcritical fission blankets. During the year the data requirements for the analysis of the blanket hybrid reactors were reviewed and identified in a paper which was presented at the International Atomic Energy Agency Symposium on Nuclear Data in Science and Technology. One conclusion of the review was that while data on reactions involving ^{238}U with 14 MeV neutrons are currently available, credible optimization of hybrid concepts to determine the potential advantages cannot be made without significant reevaluation of the data.

An effort to define and evaluate the neutron-nucleus interaction data which are important in the prediction of the neutronics of conceptual CTR blankets continued during the year. In addition to the evaluation of the needs of the hybrid concept which involve ^{238}U , ^{232}Th , and ^{237}Np , the data needs for the use of CTR neutrons to transmute fission products were also evaluated. It was determined that the uncertainties in the thermal capture cross section values are such that additional measurements are required for ^{90}Sr and ^{137}Cs before accurate estimates of the advantages of using a CTR for transmuting fission products can be made.

The ability to calculate with confidence the space energy distribution of initially 14 MeV neutrons slowing down in graphite is fundamental to the technology of CTR blankets. Two integral experiments have been performed at PNL in which the space and energy distributions were characterized by a series of activation detectors. In order to assess the validity of the measured activity values a review has been made of the experimental data. Several errors were located in the data and a publication containing the final results of the experiments is being prepared.

A study has been made to define the requirements of a valid experiment to measure the age of neutrons with initial energy of 14 MeV in graphite as a calculational bench-mark experiment for CTR blanket neutronics. In the course of the study consideration was given to the effects of the loss of neutrons due to leakage from a finite block of moderator. It appeared desirable to reevaluate a correction formula which appears in the published literature. Examination of the convergence properties of the series appearing in the formula raised doubts about the validity of both the generalization and special cases which have appeared in the literature. A paper in which the validity of this correction is discussed has now been published, and an abstract of the paper appears in this report.

Nuclear Data Requirements for Fusion-Fission (Hybrid) Reactors

W. C. Wolkenhauer and B. R. Leonard, Jr.

Neutron cross sections used in the analysis of blankets must be known much more precisely for studies of hybrid reactors than for pure fusion CTRs. These data requirements have been reviewed and identified in a paper presented at the International Atomic Energy Agency Symposium on Nuclear Data in Science and Technology, and the proceedings of the conference have now been published.⁽¹⁾ The abstract of this paper follows:

ABSTRACT

NUCLEAR DATA REQUIREMENTS FOR FUSION-FISSION (HYBRID) REACTORS.

The hybrid (fusion-fission) reactor is a class of power-producing reactors which has fusion and fission reactors as its extreme members. There are two distinctly different hybrid systems: Hybrid systems which optimize on the breeding of fissile fuel for fission reactors or tritium for D-T fusion reactors, and hybrid systems which optimize on power production and utilization of energy resources. Studies have been made on the hybrid since 1953 with much of the recent work being carried out by the authors at Battelle-Northwest.

The justification for pursuing the study of hybrid systems is the possibility of achieving a power plant which couples the short doubling times associated with the fusion reactor to the high power densities associated with the fission reactor while at the same time being able to extract power from the available uranium reserves from an essentially safe (subcritical) fission lattice.

The design and optimization of hybrid plants places new restraints on the required nuclear data needed for the analysis. All of the usual reactions considered in fission reactor design must be covered by the data but now the requirement is for good data from thermal to 14 MeV. In addition, $n,2n$ and $n,3n$ reactions which are not normally significant in fission reactor design become important. Finally, of course, plasma physics data and reaction cross-sections of structural materials for 14 MeV neutrons are required.

Most hybrid concepts are based upon the nuclear reactions of ^{238}U with 14 MeV neutrons. There are several neutron-producing reactions which take place. They include $n,2n$, $n,3n$, and fission reactions. While data on these reactions are currently available in the Evaluated Nuclear Data File (ENDF/B-III) of the USAEC, credible optimization of hybrid concepts to determine the potential advantages cannot be made without significant reevaluation of the ^{238}U data file. Hybrid systems can also be designed based on the thorium fuel cycle. In this regard, the same type of data is required here but the available data are inferior to those of ^{238}U .

Nuclear Data for CTR Neutronics

B. R. Leonard, Jr.

The work in this area is a continuing effort to define and evaluate the neutron-nucleus interaction data which are important in the prediction of the neutronics of conceptual CTR blankets. A study of particular nuclear data requirements for fusion-fission hybrid blankets was reported separately in this report. In addition, a detailed review of the nuclear data needs for ^{238}U , ^{232}Th and ^{237}Np was provided⁽²⁾ for a meeting of the United States National Defense Council (USNDC) CTR Subcommittee. The recommended nuclear data needs for these isotopes are summarized below.

ResultsTABLE 1. Nuclear Data Needs

^{237}Np			
<u>Reaction</u>	<u>Energy Range</u>	<u>Desired Accuracy of Capture Cross Sections</u>	<u>Comments</u>
n,2n	$E_T - 15 \text{ MeV}$	$\pm 10\%$ or $\pm 0.05 \text{ b}$	Only one $\pm 15\%$ measurement at 14.5 MeV. ^{236}Pu production.
n,3n	$E_T - 15 \text{ MeV}$	$\pm 20\%$ or $\pm 0.05 \text{ b}$ minimum	No measurements.
^{232}Th			
n,n'	4 - 15 MeV	Evaluation	Needs coupled channel calculation to discrete levels.
n,2n	10 - 15 MeV	Evaluation	Fit data with model calculation. Better description of energy spectra for two neutrons.
n,3n	$E_T - 15 \text{ MeV}$	$\pm 10\%$ or $\pm 0.05 \text{ b}$ minimum	Only one eta measurement.
n,3n	$E_T - 15 \text{ MeV}$	Evaluation	Nuclear model calculation needed. Improved description of secondary neutron energy spectra.

TABLE 1. (contd)

^{238}U			
Reaction	Energy Range	Desired Accuracy of Capture Cross Sections	Comments
n,2n	$E_T - 15$ MeV	Evaluation	Evaluation of cross section and energy distribution of secondaries.
n,2n	10 - 13 MeV	$\pm 10\%$	No data
n,2n	$E_T - 10$ MeV	$\pm 10\%$ or ± 0.05 b	Need confirmation of existing measurement.
n,3n	$E_T - 15$ MeV	Evaluation	Evaluation of cross section and energy distribution of secondaries.
n,3n	$E_T - 15$ MeV	$\pm 10\%$ or ± 0.05 b minimum	Limited data of poor accuracy.
n,n'	2.5 - 15 MeV	Evaluation	Evaluation. Discrete level scattering needed.
eta	14 MeV	$\pm 5\%$	Four experimental values range from 2.7 to 3.3.

In other sections of this report calculations have been reported on the possible use of CTR neutrons to transmute fission products which have small values of thermal capture cross sections to shorter-lived or stable isotopes. The fission product nuclei which were studied most extensively were ^{90}Sr and ^{137}Cs . These nuclei have very small and uncertain thermal capture cross sections. Even so, the projected transmutation rates are primarily due to thermal neutron capture in the systems which were studied. The result of these studies has been that ^{90}Sr appears to be a promising candidate for the projected neutron flux levels of a CTR. The prospects for ^{137}Cs are only of marginal interest. However, the uncertainties in the thermal capture values are such that, at the 90% confidence level, neither ^{90}Sr or ^{137}Cs would be of much interest or, conversely, the transmutation rates for ^{137}Cs could be twice those calculated.

The thermal capture cross section of ^{137}Cs has apparently been measured only one time. D. C. Stupugia⁽³⁾ obtained a value of 110 mb for the

activation of the ^{138g}Cs ground state. Although the error on this value is usually quoted (e.g., BNL-325, vol. II B, May 1966, and Third Edition, vol. 1, June 1973) as ± 33 mb, this was only the random error of the experiment. The total error assignment of Stuepaga was ± 50 mb. In addition, it is now known that there is a 2.9 min isomeric state of ^{138}Cs which exhibits some beta decay. Thus, there may be an additional systematic error in the value of the thermal capture cross section.

A single activation measurement of 0.8 ± 0.5 b exists⁽⁴⁾ for ^{90}Sr .

To the extent that the possible transmutation of fission products in CTRs is of interest, additional measurements of the thermal capture cross sections of ^{90}Sr and ^{137}Cs are of the highest priority. Accuracies of $\pm 10\%$ are desirable and $\pm 20\%$ would be useful. No other nuclear data needs of target fission product nuclei or their progeny have been developed.

Measurement of 14 MeV Neutrons Slowing Down in Graphite

R. J. Hall

The ability to calculate with confidence the space and energy distribution of initially 14 MeV neutrons slowing down in graphite is fundamental to the technology of CTR blankets. Consequently, two integral experiments have been performed at PNL in which the space and energy distributions were characterized by a series of activation detectors. In the first experiment,⁽⁵⁾ a rather small graphite stack was used in an open room. In a second experiment,⁽⁶⁾ the graphite moderator was contained in the High Temperature Lattice Test Reactor (HTLTR). Both experiments were performed at room temperature. The objective was to develop experimental techniques that would provide valid data which could serve as calculational benchmarks and which could be extended to other high temperature systems.

Progress in CY 1973

In order to assess the validity of the measured activity values a review has been made of the experimental data. This review has indicated that errors existed in several instances in the data. These include:

- An apparent error in the normalization of the manganese activity in the first experiment causing significant overestimation of the activity
- A procedural shortcoming in the foil-counting techniques used in the HTLTR experiment which caused errors in magnitude and position dependence of the several activation foils
- Several problems in normalization of foil activity in the HTLTR experiment.

The reanalysis of these experiments is nearly complete and a publication containing the results is being prepared.

Neutron Age in Graphite

B. F. Gore^(a)

A study undertaken to define the requirements for a proposed fusion-related experiment has led to the identification of a fundamental flaw in the logic supporting an accepted theoretical relation.

The experiment of interest was the measurement of the age of 14 MeV neutrons in graphite, to serve as a benchmark for CTR neutronics calculations. In the absence of such a benchmark experiment, calculational tests of graphite multigroup cross section sets had been made against the measured age to indium resonance of fission neutrons in graphite. These tests yielded smaller calculated values than the measured values.⁽⁷⁾

The theoretical relation in question is the formula for calculation of the neutron age as a series in the even flux moments measured in a rectangular prism of moderator material which abuts a planar source of fission neutrons. It is discussed in Beckurts and Wirtz⁽⁸⁾ and comes from a 1951 report by Weinberg and Noderer⁽⁹⁾ summarizing work of Weinberg and Rose⁽¹⁰⁾ which was not declassified until 1955. The formula is a special case of a general development in which a distributed source in the $Z = 0$ plane is expanded as

$$S(x,y) = \sum_{m,n=0}^{\infty} f_{mn} \cos(B_m x) \cos(B_n y) \quad (1)$$

where

$$B_m = \frac{(2m+1)\pi}{a}, \quad B_n = \frac{(2n+1)\pi}{b} \quad (2)$$

and it holds only when

$$f_{mn} = \delta_{m0} \delta_{n0} \quad (3)$$

a. This work was performed while the author was a USAEC-Sponsored NORCUS Faculty Research participant with Battelle-Northwest. The author is now a Battelle-Northwest Staff Member.

although this is not mentioned in Reference 1. For a rectangular prism of sides a and b the neutron age τ is then given by

$$\tau = \frac{1}{2} \overline{z^2} = \frac{\sum_{v=0}^{\infty} \frac{v+1}{(2v+2)!} \left(\frac{\pi^2}{a^2} + \frac{\pi^2}{b^2} \right)^v \overline{\xi^{2v+2}}}{\sum_{v=0}^{\infty} \frac{1}{(2v)!} \left(\frac{\pi^2}{a^2} + \frac{\pi^2}{b^2} \right)^v \overline{\xi^{2v}}} \quad (4)$$

where the $\overline{\xi^{2v}}$ are the even flux moments measured along the axis of the prism.

A challenge to the validity of this correction has been published in Nuclear Science and Engineering⁽¹¹⁾ under the title "Is the Calculation of Neutron Age from Experiments Using Extended Planar Sources Valid?" The abstract of this paper follows.

ABSTRACT

In a class of experiments using extended planar sources, the age of fission neutrons is calculated by "correcting" the measured second moment of the flux through the use of a series in the higher flux moments. In this paper the "correction" is generalized to include terms in addition to the leading term of an eigenfunction expansion of the neutron source distribution. In the generalized "correction" series, expansion coefficients are shown to be series themselves, which cannot be shown to converge in general. Examination of physically reasonable examples, one of which included only the effect of the energy-dependent extrapolation length of a published experiment, reveals divergences in the series for all expansion coefficients but that of the leading term in the correction series. Since the assumption of an energy-independent extrapolation length was central to the derivation of the correction series in question, this indictment is quite general.

The experimental values available for comparison with calculations are tabulated on the following page.

<u>Experiment</u>	<u>Age to Indium Resonance</u>
Fermi ⁽¹²⁾	317.0
Hill ⁽¹³⁾	310.6 $\pm$ 3.0
Davey ⁽¹⁴⁾	337.9
Hendrie ⁽¹⁵⁾	312.6 $\pm$ 0.5
Campbell ⁽¹⁶⁾	307.8 $\pm$ 2.0

Of these experiments, those by Hill, Davey and Hendrie calculated the reported values of age using Equation (4) which has been shown to be invalid. Further, the Fermi experiment is known to contain errors due to the use of a large (293 gm) "point" source and to excessively thick (0.012 in.) cadmium foil boxes.⁽¹⁰⁾ Thus this work⁽¹¹⁾ puts the historical fluctuations of the measured age to indium resonance of fission neutrons in perspective (they were not relevant). By the process of elimination, then, it emphasizes that the latest (and smallest) value is the one against which calculations testing multi-group cross sections should be compared until a 14 MeV age measurement is performed.

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PLASMA PHYSICS

During the year significant advances have been made in investigations of nonlinear aspects of the cyclotron-resonance interaction between electromagnetic waves and charged particles in plasma media. Although the work was not accomplished with fusion research needs in mind, the results should be of importance to the controlled thermonuclear research effort. As a result of the PNL study, identification of all six constants of the particle motion have been identified so that the nonlinear interaction effects can be solved exactly. The new results should allow for deeper analytical penetration into applications of plasma physics results. Hopefully, some processes will yield their detailed behavior to this nonlinear method and thereby clarify the physics and allow a better estimate of their feasibility.

Research on the Nonlinear Cyclotron-Resonance Interaction

H. B. Liemohn and B. H. Duane

During CY 1973, BNW staff investigated nonlinear aspects of the cyclotron-resonance interaction between electromagnetic waves and charged particles in plasma media. The intended application of this work is stimulation of ULF-VLF wave amplification by localized injection of plasma in the terrestrial magnetosphere. However, the interaction is an important energy exchange process in laboratory plasmas as well, and our new results should be applicable to the controlled thermonuclear research effort. The ULF-VLF wave amplification study was funded by the Advanced Research Projects Agency through a contract from the Office of Naval Research.

The cyclotron-resonance interaction has received extensive attention from the international scientific community due to its significant role in magnetoplasmas. However, its nonlinear aspects have been particularly elusive because its exact solution is described by elliptic functions in the complex plane. This fact has prevented useful application of the results because many side constraints and most constants of the motion remained unknown. The existing literature on the nonlinear interaction had simply not carried the exact problem far enough before introducing crucial assumptions. Understanding of the theory of doubly periodic (elliptic) functions which describe the nonlinear ion orbits and the formidable mathematics connected with it has been an insurmountable obstacle in the past. These problems, however, have been overcome and the PNL study has resulted in the identification of all six constants of the particle motion. Heretofore, only two were known.

With all of the constants now available, the coupled Vlasov-Maxwell equations became integrable analytically. The formidable system of coupled nonlinear differential equations was reduced to a nontrivial eigenvalue problem. The discrete set of eigenvalues span a complex six-dimensional hyperspace which makes their evaluation difficult

numerically. Nevertheless, an extensive computer program has been compiled to probe all aspects of the interaction: particle trajectories, arbitrary phase space distributions, nonlinear wave amplitudes, and systematic convergence in the eigenvalue hyperspace.

The new results allow for deeper analytic penetration into other applications of plasma physics as well. The constants of motion allow measured distributions at boundaries to be analytically continued into the interior region. Closed-form integration can be carried several more steps in many cases. Finally, the technique is certainly appropriate for other nonlinear problems.

Fusion research may be served in several ways. First, the existing nonlinear formalism can be used to optimize conditions for electromagnetic pumping of plasma energy density. Second, radiation loss rates for specific modes of emission are readily identified and may be assessed quantitatively. The theory is applicable to a wide range of ion species, plasma densities (including solids), and magnetic fields. Third, after reviewing the current status of problem areas, one or more new avenues of investigation could be initiated with this nonlinear method of analysis. Fourth, while many studies of fusion plasmas have resorted to numerical integration at an early stage, the physics of certain processes remains obscure. Hopefully, some processes will yield their detailed behavior to this nonlinear method, and thereby clarify the physics and allow a better estimate of their feasibility.

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