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AEC Research and Development Report

THE PACIFIC NORTHWEST LABORATORY
ANNUAL REPORT ON
CONTROLLED THERMONUCLEAR REACTOR
TECHNOLOGY - 1972

NOVEMBER 1972

BNWL-1685

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THE PACIFIC NORTHWEST LABORATORY
ANNUAL REPORT
ON
CONTROLLED THERMONUCLEAR REACTOR TECHNOLOGY - 1972

by

THE PACIFIC NORTHWEST LABORATORY
CONTROLLED THERMONUCLEAR REACTOR
TECHNOLOGY COMMITTEE

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November 1972

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PACIFIC NORTHWEST LABORATORIES
Richland, Washington 99352

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THE PACIFIC NORTHWEST LABORATORY
ANNUAL REPORT
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I . INTRODUCTION

.This report summarizes the research tasks carried out by the Pacific Northwest Laboratory (PNL) in FY 1971 which are directed toward finding solutions to some of the engineering problems of Controlled Thermonuclear Reactor (CTR) power plants.

While there has been a significant amount of worldwide progress in the development of thermonuclear plasmas in recent years, the situation with regard to the engineering problems associated with CTR power plants can be described as dismal. However, certain specific high priority engineering problems have been identified and are being studied. At PNL, those problems which are considered to be of a high priority and in which PNL has special expertise, have been studied in the past fiscal year. Some of these problems include neutronics, hybrid (fission-fusion) systems, fusion cross section theory, economics, graphite materials problems, hydrogen isotope effects, thermal hydraulic analysis, sputtering effects and ir-radiation induced swelling. These specific areas were subjected to various degrees of study which was mostly supported by internal funds.

The next section of this report reviews the conclusions reached in the research. The final section of this report gives a detailed description of the research effort.

II SUMMARY

This report, which is prepared in a similar fashion to last year's annual CTR engineering report, summarizes several research and development

efforts undertaken at PNL during Fy 1972 in CTR engineering problems. The work was supported, in general, by internal funds, except that some specific pieces of work (such as the hybrid studies and the waste management studies) were supported by direct USAEC funding of these specific projects. In addition, some of the other projects reported on here were supported in part by related research projects.

A quick glance at the table of contents of this report. shows that research projects were carried on in neutronics, cross section theory, economics, materials projects of several kinds, and thermal hydraulics. The latter project is an example of an area of work that is just being initiated. In general, the above projects represent a marriage of the critical needs of the eTR engineering program with the specific abilities of the Laboratory. This is in line with the BNW eTR project philosophy of only carrying out work which at this time falls within the dual area of critical program need and specific BNW competence.

The hybrid (fusion-fission) project was carried out in two parts. The initial phase consisted of developing the position of hybrid systems relative to other sources of power on a sort of overview basis according to economic, resource, and environmental criteria. The second phase of the work consisted of a thorough time dependent neutronics study of one specific hybrid lattice.

The overview study was accomplished by characterizing the hybrid reactor as a class of reactors which has fission and fusion reactors as the extreme members of this class. If this viewpoint is taken and the hybrid reactor is then studied according to economics, resource, and

The use of a CTR device as a means of transmuting fission products was studied and, on the basis of the work to date, looks reasonably promising. The radiological burden is decreased and stable daughters are produced. There appears to be the physical space in a CTR for this process and the blanket heat load is not excessive. A few CTRs could, in principle, handle the fission products from many fission reactors. Finally, as fission products were found to be net neutron producers in this environment, the process could be a net producer of energy.

The participation of BNW in the standard blanket calculations proposed at the ORNL working sessions are reported here. While comparison with the results of other laboratories cannot be made based upon the data reported here, the first phase results have been reported in other sources to compare favorably with the total benchmark effort. One conclusion that has been reached as a result of extension of the standard blanket calculations at BNW is that blankets constituted in a fashion similar to the one considered in this study do not contain nearly enough lithium inventory for long periods of operation. Therefore, constant blanket refueling, at least for this class of blankets, will be a necessity.

Fusion cross section measurements were subjected to a least squares fitting routine based upon theoretical models of the fusion process. The $T(d,n)He^4$ fit is acceptable except for a residual region problem and a discrepancy at the high energy end. The $He^3(d,p)He^4$ and $D(d,p)T$ fits are truly acceptable. The $D(d,n)He^3$ fit is not nearly as acceptable and points out some areas for needed further work. The $T(He^3, x)y$ fit is very good,

but the $T(t,2N)He^4$ fit points out a need for further measurements at certain energies. In addition to the above studies, an abstract of the computer code developed under this program which analyzes symmetric fusion ignition is included in a separate section.

.The economic characteristics of fusion reactors was explored on a preliminary basis and is reported here. The main conclusion that can be drawn from the work to date is that the mathematical models developed for fission reactor economics can be applied to fusion reactors with little modification. True economic comparisons will have to wait for further study.

A major conclusion of the studies to date which have been performed on graphite for eTR power plants is that on the basis of current data, there seems to be little hope that conventional graphites can sustain the projected eTR fluences. However, new types of graphite may possibly demonstrate the required dimensional stability.

In the hydrogen isotope effects work reported in this document, two preliminary measurements have been made to date. Hydrogen solubility curves for niobium have been generated and found to be in reasonable agreement with the literature. In addition, solubility determinations have been made on a niobium-1 zirconium alloy.

Sputter-deposition appears to be a very attractive means for developing new and better superconducting compounds. It appears to be peculiarly useful as a research tool.

eTR vacuum wall sputtering research is reported in this document. The primary effort to date has been in the development of correlations which can be used in preliminary design work. Physical sputtering may be the most important process in determining the eTR vacuum wall lifetime. Sputtering rates are shown to be such that certain pulsed machines would indeed have wall erosion problems. The proposed steady state machines, at least at proposed particle energies, appear to have fewer problems.

Transmission electron microscopy has been used to study swelling in niobium and Nb-1% Zr alloy which has been bombarded with 7.5 MeV Ta+++ ions at elevated temperatures. At 800°e, the swelling saturated to a value of 4.5% in pure niobium. At 900°C, the swelling would appear to saturate at a value between 7 and 10%. The saturation is associated with the formation of a void lattice. Voids are also observed in Nb-1% Zr alloy at 900°C, but the swelling was less than in pure niobium. The void distribution was very inhomogeneous in Nb-1% Zr. These results give a preview of some of the material damage to be expected in long time CTR operation.

As one can see, the major emphasis of this year's report includes studies on hybrid systems and material damage research. These areas are distinctly separate in that one task advocates the potential development of a new power source whereas the other task develops solutions to problem areas which tend to plague the whole eTR engineering effort. Obviously, a strong effort will exist in the latter area for some years. On the other hand, unless the advantages of high power density short doubling time systems are recognized and, unless the fact is recognized that all cheap uranium and thorium areas will be exploited by some means in the future, development of the hybrid system will not take place at a rate commensurate with its known advantages.

III. DISCUSSION

This section consists of a series of individual papers which report, in detail, the methods used and data developed in the research tasks. The papers are prepared in such a way that they can be abstracted from this report. The one exception to this general format is that the references are grouped together in the last section to form a limited bibliography for the areas of investigation.

A. Fusion-Fission (Hybrid) Systems - B. R. Leonard, Jr. and W. C. Wolkenhauer

The fusion-fission or hybrid reactor can be considered as a class of reactors which has fusion and fission reactors as its extreme members. The concept first appeared in the United States literature in 1953 and has been sporadically studied since that date. There are many different possible combinations which the concept embraces. The possible spectrum of combinations can range from coupled systems which envision separate fission and fusion plants whereby the fusion plant breeds fissile material which can be burned in the fission plant and single systems in which a fusion device breeds fissile material in its blanket and initiates the fission reaction in the blanket material.

The concepts can involve subcritical or critical fission assemblies and fusion devices with energy balances below or above a breakeven point. There are several different fissile fuel cycles which can be employed including ^{238}U - ^{239}Pu cycles, ^{232}Th - ^{233}U cycles, and mixtures of the two. The fissile cycle can be optimized to breed fissile material or it can be optimized to provide maximum energy production. Various reactor technologies such as molten-salt, liquid-metal, or gas-cooled techniques may be employed.

Finally, the concept can be applied to either linear or toroidal magnetically confined fusion devices, or alternately, it can be applied to inertial confinement systems. A large portion of the possible hybrid plants have the interesting combination of benefits of high power density from fission and short doubling times from fusion. Hybrid reactor systems offer opportunities for arriving at economic, resource utilization, or

ecological optimums which may not exist in the extreme members of this group. In determining the perturbations that the development of a hybrid reactor would have on the fusion program, the discussion will be limited to a comparison with deuterium-tritium (O-T) plasmas which are confined magnetically.

Economics

Detailed economic analysis of hybrid systems is not possible at this time due mostly to the lack of design definition. However, some observations can be made about the probable economics of hybrid systems relative to the pure fission and pure fusion systems. For a CTR, a major capital cost item is the base cost of initiating a fusion reaction. The economic optimization which must then be performed is to determine the maximum return on this investment.

A thermonuclear event (O-T) produces 17.6 MeV of energy in the form of an energetic neutron and a helium nucleus. In terms of energy resource economy there are three possible uses for the fusion neutron after kinetic energy has been extracted. It can be used to breed a tritium atom from lithium, it can be used to breed a fissile atom from a fertile atom or it can be used to fission a fissile atom. Possible alternate uses, such as transmutation of fission products or other isotopes will not be considered in this analysis.

It would seem probable that the mission selected for the neutron would be that which provides the greatest economic return if other considerations are approximately equal. The second and third alternatives (breeding or fission) are really variations of the same alternative in

that the bred fissile atom will ultimately be used for fission. The tritium bred in the first alternative will be used to create a fusion event with consequent release of approximately 18 MeV of energy which can be marketed on the energy market at the current price. Alternately, if the neutron is used in a fission event, approximately 200 MeV of energy is released for ultimate sale.

Thus, it would appear that neutrons available from a fusion plant would yield a greater economic return when used to create a fission event than when used to ultimately create a fusion event. Furthermore, this situation would hold true as long as suppliers of fertile material existed which can be recovered at present costs and only when the fertile fuel reserves have been depleted, would it be economically reasonable to use the neutron for tritium production.

Obviously, tritium production has to take place in a hybrid plant to replenish that consumed. The above discussion points out, however, that there are strong economic arguments for making the ratio of the output energy to the input energy for a fusion reactor used in a hybrid system as low as possible because of the high relative cost of tritium production.

It would appear then, that some hybrid systems would approach the power costs expected for fission breeder plants if the plants had similar capital and operating costs. There might, however, be secondary economic savings which could accrue to the hybrid system by virtue of its anticipated low inventory of fissile fuel and increased safety (subcritical fission blanket and possible lower fission product content). Indeed, it

The production of fission products per unit of output power in a hybrid reactor would be similar to that of a pure fission device. Because of the projected short doubling times for hybrid plants, which would require nearly constant fuel reprocessing, the fission product inventory of a hybrid reactor may be somewhat less than that of a fission reactor. On the other hand, a fission device would contain a significant actinide inventory.

Because the fusion device employed by a hybrid reactor does not require a breakeven energy balance, the tritium inventory could be considerable less than that of a pure fusion device. However, the tritium inventory problem is not nearly as severe a hazard as the fission product problem. For

equivalent plant sizes, the relative biological hazard from ^{131}I alone has been estimated to be a factor of at least 10^6 times that from ^3H .

Economic fusion plants employing magnetic confinement systems have been shown to be several thousand megawatts in size. For a toroidal tokamak reactor, the smallest size of an economic system has been estimated to be 5000 MWt. Economic hybrid reactors should be of the size of current fission reactors (≤ 3000 MWt). If thermal conversion power systems are used on all systems with approximately the same efficiencies, this would imply that a pure fusion plant would be releasing much more energy to the local environment as compared to either a hybrid or fission reactor if each is built at the minimum power level required to produce economical power. At the present time, there are large areas in the United States that cannot use the large powers projected for the pure fusion plants. Therefore, because thermal pollution is partly a local environmental problem, the hybrid system would appear to have an advantage relative to a pure fission system in this respect. However, a fusion plant based on inertial confinement has been projected to be economically possible in sizes of the order of present day fission systems. In this case, the thermal pollution problem appears to be equivalent to other thermal power plants.

Resource Preservation

The resources required to fuel a hybrid system include fissionable or fissile breeding material (U or Th), deuterium and lithium for the tritium breeding for the O-T cycle. To produce an equivalent amount of power, a hybrid system which achieves most of its power from a fission

reaction would fission approximately the same amount of material as a fission system. It would utilize a much smaller amount of lithium than a comparable fusion system.

A hybrid system could potentially utilize depleted uranium more efficiently than a fission breeder and the fissile breeding rate for ^{233}U and ^{239}Pu , even in subcritical systems, could exceed that of a fission breeding system. A hybrid plant offers the possibility of utilizing energy reserves from both lithium and fissile fuels to their fullest extent. The total energy reserves for lithium and of fertile-fissile material have been estimated to be approximately the same. The development of a deuterium-deuterium (D-D) fusion reactor would eliminate resource preservation as a primary goal in energy research and development.

In considering the resource preservation characteristics of fusion, fission and hybrid systems, the alternate markets for fuel must be considered. For example, there are apparently no other significant markets for uranium and thorium other than use as a fissile fuel. There is, however, the possibility of the existence of significant alternate uses for lithium. There is an extensive commercial market for lithium at the present time. In the future, if liquid metal-water power sources are developed into commercial use, the potential alternate market for lithium could greatly expand. The question of alternate uses of candidate fuel materials is one which has not perhaps as yet received the attention it should in the overall analysis.

Strategy for R&D

From the previous discussion, when the three factors of economics, environmental quality, and resource development, are considered in

relation to the hybrid plant, one finds that various minima appear to exist for each factor somewhere in the spectrum of possible hybrid power plants. Further, it appears that these minima could even include the extremum members of the hybrid spectrum. That is, the economic minimum could be the fission reactor and the environmental minimum could be the fusion reactor. However, if all three of these factors are considered, the overall optimum point appears likely to lie somewhere in the spectrum of the hybrid reactor. The specific location would be dependent upon the weights assigned to each factor. If further investigation establishes the location of this optimum in the hybrid spectrum, this could have a significant impact on the relative priorities for eTR research and development.

The major perturbations of the development of hybrid systems on the overall fusion development programs are in the areas of blanket development, neutronics, economics, and fissile fuel processing. It would appear that a thorough investigation of the economic potential and the environmental impact of a hybrid system is required in order to establish its relative importance to the future fusion program. Additionally, an increasing amount of blanket development effort and neutronics measurements are needed for the hybrid system relative to the pure fusion device. Finally, if the economic analysis demonstrates a cost advantage for the hybrid system, and if the neutronics measurements demonstrate the viability of hybrid systems, the result could be an increase in support for engineering technology studies in the overall fusion program.

Hybrid systems have a potential for causing a perturbation in the overall development of fusion devices in another way. Early design,

construction, and operation of hybrid systems could provide design and operational experience which would be of direct benefit to the fusion reactor development program. In addition, the pure fusion prototype must breed tritium at a breakeven rate for a (O-T) reaction eTR if any appreciable quantity is to be consumed. Early operation of a hybrid plant can provide information as the viability of such an approach because break-even breeding of tritium appears to be assured in hybrid design.

Major technological problem areas which need to be studied and which are unique to the hybrid plant are shown in Table A.1. These areas are reviewed on the basis of perturbations to costs for pure fusion R&D.

The potential advantages of the fusion-fission hybrid reactor suggest that a detailed consideration be given to the concept as an economical source of power. In the near term the neutronics characteristics of these power plants must be understood to the extent that a judgment about the viability of the concept can be made. In addition, the probable economics of such a plant must be studied to the extent necessary to make confident economic decisions. As a result of such studies a better determination can be made of a longer term realistic research and development program for the hybrid concepts.

TABLE A.1

MAJOR REQUIRED R&D TASKS
FOR HYBRID REACTOR DEVELOPMENT

ECONOMIC ANALYSIS

Economic Feasibility

Comparative Analysis; Develop Relative Worths of Fuels

Detailed Design Analysis

Cost/Benefit Study

Detailed Operations Analysis

SYSTEMS ANALYSIS

Determination of Optimum Fusion Reaction

Selection of Preferred Fusion Device

Determination of Preferred Blanket Technology

Selection of Optimum Hybrid (Fusion-Fission Mixture)

NEUTRONICS

Determine Neutron Balance

Criticality Analysis

Fuel Management Cycle Analysis

Optimum Power Density Analysis

Fission Product Containment

Collection and Analysis of Nuclear Data

Development of Design Models

SYSTEM CHEMISTRY

Fission Product Processing

Corrosion Problems in Fissile Fuels

Plutonium Processing

TABLE A.1 (Continued)

ENVIRONMENTAL QUALITY

Thermal Pollution Analysis

Optimum Energy Conversion

.Radioactive Materials Analysis and Management

MATERIALS

Neutron and Charged Particle Damage

Material Swelling Due to Increased Neutron Flux

Nil Ductility Effects from Neutron Bombardment

Determination of Mechanical Properties in the Presence of
Fast Fission Events

PROTOTYPE DESIGN

HTGR/MSBR Technology Application

Fuel Clad Analysis

Use of Liquid Fuels

Magnetic Properties of Fissile Fuels

Shape of Magnetic Fields

Optimum energy Conversion Scheme

Heat Transfer and Fluid Dynamics Analysis

Refueling and Maintenance

Lattice Design

B. A Hybrid Neutronics Analysis - B. R. Leonard, Jr.

The justification for pursuing the study of fusion-fission hybrid systems is based on three observations:

- That an efficient pure-fusion CTR is not yet attainable and could be economically prohibitive.
- Plasma characteristics already attained show promise of achieving those which would allow reaching the objectives of fusion-fission hybrid systems.
- That fusion-fission hybrids afford excellent utilization of the energy reserves of fissile and fertile material and achieve fuel doubling times well matched to demands.

There appear to be only two distinctly different objectives for fusion-fission hybrid systems.

- Hybrid systems which optimize on the breeding of fissile fuel for fission reactors.
- Hybrid systems which optimize on power production and utilization of energy resources and diminish the need for critical fission reactors.

The sometimes stated objective of utilizing the hybrid to multiply the power of a sub-Lawson CTR in order to study or prove its feasibility does not seem to be a reasonable one. It would seem more logical to study the sub-Lawson CTR in a simple CTR plant and provide the negative power needs with a fission power reactor. The dual system would most likely be less expensive, certainly contain less risk, and provide a simple clean CTR device for further development. The only requirement of such a CTR would

be to breed tritium and without having to generate power this requirement should be easily met. In addition, the power requirements of the sub-Lawson CTR need not be excessive. A plant which is optimized for the production of fissile material, however, would still generate several hundred megawatts of thermal energy. Thus, a plant, even though optimized for fission breeding, might well be a net producer of electricity.

Discussion

The BNW studies of hybrid systems which have been carried out in the last few months have been designed to explore the objective of the optimization of power multiplication with a sub-critical fission lattice. These studies are an outgrowth of parametric studies of fissile lattices for CTRs which were reported in, "The Pacific Northwest Laboratory Annual Controlled Thermonuclear Reactor Technology Report - 1971," p. 39, BNWL-1604 (June 1971) and presented orally at the International Working Sessions on Fusion Reactor Technology, Oak Ridge National Laboratory, June 28 - July 2, 1971, CONF-710624. The earlier studies served as a guide to the initial design configuration of the present work. The present work was designed to study the neutronic problems attendant in attaining the objective of optimizing the power produced in the blanket while making maximum utilization of fissile and fertile material in a sub-critical fission lattice.

The lattice chosen in this study is a niobium clad natural uranium, graphite moderated, helium cooled lattice with a 7 inch pitch as shown approximately to scale on Figure B.1. The basic details of the fission lattice were not perturbed in this study. An isotropic 14 MeV neutron source was defined to be uniformly distributed in a cylindrical thermonuclear plasma of 160 cm radius. No further assumptions were made of the plasma characteristics other than it produced N neutrons/sec on a

FUSION-FISSION BLANKET

FIGURE B. 1

time average. The inner radius of a 0.5 cm niobium vacuum wall was chosen to be 200 cm. The next region was designated as a neutron convertor region. It consisted of 0.5 in. diameter pins of depleted uranium-carbide and a quantity of natural lithium with an atom density half that of the uranium. The depleted uranium pins were assumed clad in 20 mil Nb and an equal amount of niobium was included for either lithium clad or channels. The fission lattice was terminated with a 20 cm thickness of graphite to act as a reflector of neutrons for the fission lattice and to thermalize neutrons for absorption of neutrons in lithium in the final 30 cm thickness of the lattice.

The neutron convertor region conceptually serves several purposes. It provides about unit multiplication of the source neutrons and produces energy through fission. It is an important source of breeding plutonium by ^{238}U neutron capture. It also decouples neutronically one side of the fission lattice from the other side of the cylinder. This helps to prevent an excessive power peaking near the interior edge of the lattice. The lithium in this region aids in this purpose. It also shields the plutonium produced in this region from low-energy neutrons which could otherwise make this region an intense fission source. Most of the tritium breeding from neutron absorption in ^6Li is also performed in this region.

Neutronic Analysis

The neutronics problem is the calculation of the reaction rate per 14 MeV source neutron of all the neutron induced reactions throughout the blanket. The problem is calculated by standard fission reactor techniques, the initial point being a room temperature clean system since this is

amenable to experimental verification. The first step is the calculation of the effective neutron cross sections in the repeating cells of the convertor and fission lattice. The cell calculations are handled in two energy groups. The region above thermal is calculated by the neutron slowing-down code HRG (Hanford-Revised GAM). The thermal cross sections for the cells are calculated by the neutron transport code BRT (Battelle-Revised THERMOS). All microscopic cross sections used in these calculations were obtained from the current version 3 of ENDF/B. The effective cross sections resulting from the cell calculations are those appropriate to treating each region as a homogeneous mixture of materials for the calculation of the distribution in space and energy of the neutron flux. This calculation was made with the neutron transport code ANISN in cylindrical geometry in the P3-S8 approximation. The results of this calculation also give the reaction rate per source neutron of all of the reactions of interest so that the neutron balance characteristics of the lattice can be determined. The calculation of the effect of changing isotopic composition of the materials in the blanket with continued irradiation is performed with the transmutation code ALCHEMY. The reaction rates calculated by ANISN and the radioactive decay rates are input to an ALCHEMY calculation for each region. ALCHEMY then calculates the isotopic content at selected time intervals. To determine the neutronic characteristics of the lattice at some chosen time interval the isotopic contents calculated by ALCHEMY are used to recalculate the cell cross sections and these in turn are used to recalculate the reaction rates with ANISN.

Results of the Calculations

The base calculation was taken to be a blanket which had a homogeneous convertor, 8 cm thick, and 160 cm thick fission lattice. The homogeneous convertor region is one in which the lithium is distributed uniformly throughout the zone. Several perturbations were made on the base case to get an indication of the effects involved. These perturbations included increasing the size of the fission lattice, doubling the size of the convertor region, or rearranging the lithium within the convertor region. For example, the homogeneous convertor was replaced with a heterogeneous convertor in which the lithium has been placed in a separate layer between the depleted uranium and the fission lattice. For the homogeneous case most of the tritium production is taking place by ^6Li absorptions in the convertor. The production of tritium by neutron inelastic scattering on ^7Li is much less than one normally expects for conventional blankets. This is a natural result of having really very little lithium in the fast neutron region. Most of the ^6Li absorptions occurring in the convertor regions are by thermal neutrons leaking from the fission lattice and the lattice multiplication is not high enough to produce sufficient tritium for breeding purposes. This problem may not be real if either data and/or calculational methods are predicting too low a multiplication. This has been observed in thermal fission-reactor calculations, in part at least, where a systematic underprediction of k_{eff} has resulted as a consequence of some refined ^{238}U resonance cross sections. This systematic effect is more important to the fusion blanket calculation since our multiplication is proportional to $1/(1-k_{\text{eff}})$ and therefore the underprediction is magnified by an order of magnitude. In addition, we conclude that

a fuel of natural uranium and recycled plutonium or enriched uranium is of interest and we can get any multiplication we require. Finally, all CTRs may not be required to breed tritium. An ultimate balance is necessary and the optimum situation might well be one in which only some plants are optimized for tritium production.

The change to the heterogeneous convertor slightly increases tritium production but has a much larger effect in lowering the multiplication of the fission lattice. In both the homogeneous and heterogeneous arrangements, the overall breeding of ^{239}Pu relative to ^{235}U burned is accomplished.

The Effect of Enriched Fuel in the Fission Lattice

The neutron balance of the natural uranium lattice was calculated for increasing ^{235}U enrichment of the fuel. Even the highest enrichment used (0.94 at.%) is insignificant compared with the enrichments used in commercial power reactors. The notable trends with ^{235}U enrichments are that all of the neutron characteristics of the lattice, tritium breeding and all reaction rates show a steady increase due to the increased multiplication of the fission lattice. The fissile breeding decreases, however, which is characteristic of enriched thermal fission reactors.

The thermal energies produced in the lattice per source neutron (Q) were also compared. This parameter includes the prompt energy production of all neutron events in the blanket that produce or absorb energy. It does not include radioactive decay energy. As a comparison, the same parameter for the standard calculational benchmark lattice was 18.3 MeV (Section D). The Q values obtained correspond to energy multiplications

of a factor of 25 for the natural uranium lattice to a factor of 45 for the highest enrichment used.

Since the results indicate that tritium breeding increases and fissile breeding decreases it is of interest to see where the breeding values are equal. The conversion ratios plotted against the ^{235}U enrichment and also against the estimated k_{eff} of the blanket are shown in Figure B.2. Both plots show that these conversion ratios should be equal and slightly positive at an enrichment very close to an enrichment of 0.763 atom percent.

Fission Lattice Characteristics

The values of thermal energy produced per source neutron provide the specifications to be applied to both the fission lattice and the plasma. The first determination is that of the blanket. The limitations are those of power density in the fission blanket and plant dimensions. Table B.1 shows these calculated parameters for a power density which is characteristic of the operation of existing gas cooled reactors with this type of fuel. The three Q values shown are for the natural uranium and two lowest enrichments. The resultant equivalent length of the plant for 1000 MW(t) is seen to be a reasonable one. The fuel exposures are a direct result of the power densities chosen.

Plasma Characteristics

The power densities of the fission lattice of Table B.1 were based on the DT reaction density of $3.3 \times 10^{-10} \text{ cm}^{-3} \text{ sec}^{-1}$ shown at the top of Table B.2. This reaction density corresponds to the ion densities shown

TRITIUM AND FISSILE CONVERSION RATIOS VS ²³⁵U ENRICHMENT

FIGURE B.2

TABLE B.1

FISSION BLANKET CHARACTERISTICS

	<u>q</u>		
	465 MeV	545 MeV	715 MeV
POWER DENSITY OF FUEL	0.75	0.87	1.15 watts/cm ³
LENGTH OF 1000 MW(t) PLANT	50	43	33 meters
ANNUAL EXPOSURE OF FUEL	800	935	1225 MWD/te

TABLE B.2

PLASMA CHARACTERISTICS

FUSION SOURCE DENSITY	$3.3 \times 10^{-10} \text{ em}^{-3} \text{ sec}^{-1}$
AVERAGE ION DENSITY	
T = 10 keV	$3.5 \times 10^{13} \text{ em}^{-3}$
T = 20 keV	$1.8 \times 10^{13} \text{ em}^{-3}$
NEUTRON ENERGY TRANSPORT THROUGH VACUUM WALL	0.05 MW/m^2
TRITIUM CONSUMPTION FOR 1000 MW(t) PLANT	$\sim 2 \text{ kg/year}$

EQUIVALENT TOKAMAK ASPECT RATIO

<u>Q</u>	
465 MeV	5.0
545 MeV	4.3
715 MeV	3.3

for two projected ion temperatures; 10 keV and 20 keV. They are seen to be substantially sub-Lawson even for a steady-state plasma. The power transported through the vacuum wall, in the form of 14 MeV neutrons, is noted to be some two orders of magnitude less than usually projected for non-fissile blankets. Also, it is noted that the required tritium inventory is quite low since only about 2 kg would be consumed in a year's operation. The total tritium in the plasma is only 70 mg on a time-averaged basis. Although the plasma confinement system is not defined nor projected in this study, the 1000 MW(t) plant corresponds to the Tokamak aspect ratios as shown.

Transmutation Effects

The transmutation effects on the neutron balance have been calculated for an irradiation of the natural uranium lattice to a time of 60 weeks. This is typical of a partial refueling of this type of fission lattice. This calculation included the effect of the fission-product inventory. The most noticeable effect is a decrease of over 10% in the Q value.

Specific Problems in Present Lattice Studies

The present fission lattice is observed to be a rather satisfactory one. The power multiplication is quite high and rather sensitive to the fissile content of the fuel. The transmutation calculation for the natural uranium fuel, however, indicates a rather stable power over the one year period which was calculated. The sensitivity of the power multiplication to k_{eff} indicates that the natural uranium lattice should be placed on a sounder basis. One thing we should do is study the replacement of the

proposed niobium clad of the fission lattice fuel with a magnesium-oxide ore which is ordinarily used in this type of gas cooled fission reactor. This replacement is expected to result in a small increase in k_{eff} due to decreased neutron absorption. This modification will probably require an adjustment in fission lattice pitch dimensions because thicker clads are required.

A second neutronics problem with the uranium-graphite lattice is that the k_{eff} values calculated are probably low because of the usage of the current ENDF/B-3 description of the ^{238}U cross section. The problem, which has recently arisen, is the description of a large number of p-wave (angular momentum = one) capture resonances, the real existence of which is questionable. This has resulted in a calculational bias to produce calculated values of k_{eff} which are the order of 3% low for clean critical experiments. If the ^{238}U cross section description is correct, then the calculational methods have been adjusted over the years to produce agreement with experiment. What we need to do is to check the present data and calculational methods against uranium-graphite critical and exponential experiments in order to adjust the data or methods to give better estimations of k_{eff} and ^{238}U capture.

Optimization of Blanket Designs for Breeding Tritium

The suggestion that not all CTR reactors must be required to breed tritium would seem to be a valid one. This concept seems to offer the potential benefits that tritium handling problems in perhaps a majority of power plants could be avoided. This situation would probably enhance the viability of these plants, both in construction and operation. There

are necessary steps that must be taken to study the potential of this system. Whether or not the concept is really attractive will depend upon the gross net rate at which tritium can be produced in a plant designed specifically for that purpose. The tritium breeder is almost compelled to utilize fissile and/or fertile materials in the blanket because the basis for the concept almost certainly depends upon achieving the maximum multiplication of the DT source neutrons. The achievement of the optimized scope design will probably be strongly dependent on the maximum utilization of neutrons in the system. Such studies are a natural follow on to the present fusion-fission hybrid studies. An optimum plant may be a combination of tritium and fissile breeding but systems with distinctly separate objectives should be defined initially.

Optimization of Blanket Design for Breeding Fissile Material

It can be demonstrated that existing studies have not satisfied the objective of the conceptual design of optimized blankets which breed fissile material subject to the required restraints. From a neutronics standpoint the potential of achieving such an optimum system has been clearly recognized for two decades. It is imperative that optimized conceptual designs of fissile breeding blankets be satisfactorily explored and documented. Only from such studies will the potential benefits and possible deficiencies of the concept be clearly defined so that logical judgments can be made. At the present time it appears that the plasma parameters which would allow the utilization of this concept may be achieved in the next few years. The achievement of plasma parameters necessary for a viable and economic power plant which does not utilize fission in some manner, however, may

not be achieved in the next few decades. Fast breeder reactors as power plants with fuel doubling times sufficiently short to meet projected demands may not be available. Even aside from the necessarily argumentative projection of the fast breeder, there is a need to study all potential systems of energy resource utilization.

C. The Controlled Thermonuclear Reactor as a Fission Product Burner - W. C. Wolkenhauer

The USAEC and other agencies are vitally interested in finding acceptable means of storage or elimination of the large quantities of fission products that are expected to exist in the latter part of the twentieth century. Announced plans call for the permanent storage of these radioactive wastes in deep salt mines. Several alternate storage concepts are being explored. Some of these include storage in bedrock or basalt formations under federal reservations. Other long-range storage proposals which are being considered include burial in a nuclear chimney and burial in polar ice caps. Possible long-range alternatives for elimination of the fission products from the biosphere include extra-terrestrial disposal and nuclear transmutation. This section summarizes some aspects of the latter option which have resulted from studies conducted at PNL.

Introduction

At least three different proposals have been made for transmutation of radioactive wastes into stable daughters. They include transmutation in a fission reactor, transmutation by means of an Intense Neutron Generator (ING) device, and transmutation with a Controlled Thermonuclear Reactor (CTR) if one were available. This paper will be concerned with the CTR transmutation approach although some of the conclusions arrived at in this analysis might be quite applicable to other transmutation concepts as well.

Why is a eTR of particular interest in the transmutation of radioactive waste? First of all, one should recognize that the fission products which are discharged from a fission reactor can be characterized as having relatively small neutron reaction cross sections. If this were not the case, they would not exist in any quantity in the reactor at discharge. As will be shown later, some of the most noxious isotopes have a half-life of the order of thirty years. If an assumption is used that, as a minimum, one wishes to transmute at about at least the rate of the nuclear half-life in order to shorten the effective half-life, one can perform

the following simple calculation. Many of the interesting fission products have neutron reaction cross sections of the order of a barn. By equating the decay rate to the transmutation rate;

$$\lambda \cong a \phi$$

or:

$$\frac{0.693}{t_{1/2}} \cong \sigma \phi$$

One can arrive at an estimate of the minimum required flux for transmutation, at this level.

$$\phi = \frac{0.693}{a t_{1/2}}$$

For a thirty year half-life and a reaction cross section of one barn, the minimum required flux level can be shown to be of the order of 10^{15} n/cm²-sec.

The initial justification, then, for analyzing a eTR as a fission product burner is that it is one of a very few devices that one can conceive of that provides this level of neutron flux in appreciable quantities.

Selection of Transmutation Concepts

Transmutation, particularly if it is a neutron absorbing process, could prove to be relatively expensive. It is, therefore, perhaps worthwhile to try and identify those fission products which could be most usefully transmuted. This identification process could be carried on either by selecting transmutation targets on the basis of the highest cross section for transmutation or by identifying the most noxious fission products. In this study, the latter approach was used.

To initiate the selection process, a computer calculation was performed on a typical Pressurized Water Reactor (PWR) with the ISOGEN code. In the calculation, the fuel which was 3.4% enriched in ²³⁵U was exposed to 30,000 MWd/tonne at 20 MW/tonne. The fuel was then allowed to cool for 10 years and the inventory of fission products at that point in time was

calculated. In addition, the activity level in curies for each fission product, the beta emission in watts for each fission product and the gamma emission in watts for each fission product were determined.

In order to determine the relative hazard associated with the fission products, those isotopes with appreciable curie activity were arranged in descending order of activity in curies and also arranged by beta-watts and gamma-watts, respectively. In all cases the percent of the total activity corresponding to each isotope was compared along with the percent accumulation as the less hazardous isotopes are added.

These comparisons are particularly useful in determining candidate materials for inclusion in transmutation schemes. For example, ^{90}Sr and ^{137}Cs account for almost 85% of the activity in curies with the next isotope, ^{147}Pm , only providing some 5% more in activity; ^{90}Sr and ^{137}Cs account for 95% of the beta activity with the next isotope, ^{85}Kr , providing about 1% additional activity; ^{137}Cs provides 75% of the activity from gamma decay; and ^{134}Cs provides 17% of the gamma activity and the next isotope, ^{154}Eu provides 7% of the gamma activity. Therefore, as a working hypothesis, ^{90}Sr and ^{137}Cs were identified as prime targets for transmutation.

In order to check these selections, maximum permissible concentrations (MPCs) for the fission product isotopes were obtained from 10 CFR 20, Appendix B, Table 1. These MPC values were divided into the activities of each isotope and the isotopes were then reordered by (curies/MPC) from the highest value on down. It is hoped in this way to account for the relative biological hazard associated with the form and specific energy of each radioactive emission. The results of such an analysis is that the first three isotopes in descending order are ^{90}Sr , ^{137}Cs , and ^{154}Eu . Other potential candidates are ^{134}Cs and ^{127}Sb . On the basis of the above approach, ^{137}Cs and ^{90}Sr were selected as the prime targets for subsequent analyses.

Transmutation will probably be based upon chemical separation of preselected fission products for target material. If this is the case, it is interesting to notice that by selecting ^{90}Sr and ^{137}Cs for

transmutation, one would also carry ^{134}Cs , another of the isotopes of interest, along in the process.

One should notice that other analyses, based upon different fuel, different cooling times, and other selection criteria, might well result in the selection of different transmutation targets.

The Transmutation Chain

Having selected ^{90}Sr and ^{137}Cs as potential transmutation targets, a transmutation chain for each isotope must be selected in order to proceed with the analysis. Along with β^- decay, at least two neutron reactions must be considered. They are the n,γ and $n,2n$ reactions. The former reaction usually has an appreciable cross section for most isotopes and is continuous over the energy range. The latter reaction is a threshold reaction and typically achieves a maximum in the neighborhood of 15 MeV and has a threshold in the neighborhood of 8 MeV. Other potential reactions, such as n,α or n,p , will be ignored in this analysis as being generally of secondary importance.

In the selection of a transmutation chain for neutron transmutation, a certain amount of arbitrariness is in order because, in principle, the chains can be open-ended. However, by comparison of the probable relative transfer rates one can arrive at a reasonable schematic of the chains. The chains selected for this analysis are shown in Figure C.1.

By observing the transmutation chains shown in Figure C.1 on a nuclide chart, one can see that they end in stable isotopes which are typically above and to the left of the target isotope on a chart of the nuclides. Therefore, one can perhaps say that an isotope which is bordered above and on its left in the nuclide chart is a prime candidate for neutron transmutation. Many of the fission products, being neutron rich, fit this description.

The Nuclear Data

In order to accurately analyze the above transmutation chains, nuclear cross section data is required on all of the isotopes included in the chain. As a minimum, the reaction cross sections are required

TRANSMUTATION CHAINS FOR Sr-90 AND Cs-137
IN A CONTROLLED THERMONUCLEAR REACTOR

FIGURE C.1

although a more thorough analysis will require the use of scattering data as well. Unfortunately, none of the above isotopes are represented on the ENDF/B-III library. Therefore, other sources need to be consulted. To perform the desired analysis, 100 group reaction cross sections in the GAM-II format were required. These 100 group cross section sets were obtained in various ways for each of the isotopes.

For the ^{137}Cs transmutation chain, all of the isotopic 100 group cross sections were obtained in the following manner. Data on the n,y reaction cross was obtained and plotted on log-log paper. For the $n,2n$ reaction cross section, measured values were obtained from various sources at about 15 MeV and then, the cross section shape was calculated by means of the formalism of Pearlstein.^t The shape was normalized to the measured value and the results were also plotted on log-log paper. From these graphs, 100 group reaction cross sections for both the n,y and $n,2n$ reaction cross sections were obtained. The sources of data for the data graphs are summarized by isotope in Table C.1.

With regard to the ^{90}Sr transmutation chain, the sources for the data are summarized in Table C.2. ^{90}Sr and ^{89}y n,y cross sections were obtained in a manner similar to those in the ^{137}Cs chains. The n,y cross sections for the other isotopes were obtained by taking Cook's data and processing it directly into a GAM-II format with ETOG. All of the $n,2n$ reaction cross sections in the ^{90}Sr burnup chain were obtained by the method described for the ^{137}Cs burnup chain.

By and large, the isotopes of interest to this study can be characterized as ones about which one can find relatively little nuclear data. There are exceptions, however. For instance, there is a wealth of data on ^{90}Zr . Therefore, the reader may find it curious that Cook's theoretical evaluation was used. In the final analysis, this data was used in order to be consistent with the other lesser known isotopes. It is felt that, particularly with a relative analysis such as this one, the overall consistency of the data is of primary importance.

^tNuc. Sci. and Eng. 23,238 (1965).

TABLE C.1

NUCLEAR DATA FOR ^{137}Cs TRANSMUTATION CHAIN

Isotope	Thermal n,y Cross Section (0.025 eV)	Comment	Epithermal n,y Cross Section	Comment	n,2n Measured Cross Section
^{137}Cs	Item 1, Vol. IIB, p. 55 - 137-1	$\frac{1}{V}$ up to 40 eV	Item 2	From 1000 eV to 10 MeV then $\frac{1}{V}$ back to 40 eV	Item 2, at 15 MeV
^{136}Cs	Item 3	$\frac{1}{V}$ to 50 eV	Item 3		
^{135}Ba	Item 2	$\frac{1}{V}$ to 25 eV	Item 4	Theoretical evaluation from 25 eV	
^{136}Ba	Item 1, Vol. IIB, p. 56 - 136-1	$\frac{1}{V}$ to 100 eV	Item 4	Theoretical evaluation from 100 eV	Item 1, Vol. IIB, p. 56 - 136-2
^{137}Ba	Item 2	$\frac{1}{V}$ to 400 eV	Item 4	Theoretical evaluation from 400 eV	
^{138}Ba	Item 1, Vol. IIB, p. 56 - 138-1	$\frac{1}{V}$ to 1 KeV	Item 1, Vol. IIB, p. 56 - 138-4	Resonance Shapes Calculated	Item 1, Vol. IIB, p. 56 - 138-5

Item 1 Neutron Cross Sections BNL-325, Supplement No.2, May 1966

Item 2 B. R. Leonard, Private Communication

Item 3 J. L. Cook, Fission Product Cross Sections, AAEC TM-549, June 1970

Item 4 Newsletter Bulletin 10, CCND-NW/10, December 1969

TABLE C.2

NUCLEAR DATA FOR ^{90}Sr TRANSMUTATION CHAIN

Isotope	Thermal n,y Cross Section (0.025 eV)	Comment	Epithermal n,y Cross Section	Comment	n,2n Measured Cross Section
^{90}Sr	Item 2	$\frac{1}{V}$ to 40 eV	Item 2	-	Item 2, at 15 MeV
^{89}Sr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	
^{89}Y	Item 1, Vol. IIA, p. 39-0-1	$\frac{1}{V}$ to 1 KeV	Item 4		
^{90}Y	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	
^{91}Y	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	-
^{90}Zr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	
^{91}Zr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	Item 2, at 15 MeV
^{92}Zr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	Item 2, at 15 MeV

Item 1 Neutron Cross Sections BNL-325, Supplement No.2, May 1966

Item 2 B. R. Leonard Private Communication

Item 3 J. L. Cook, Fission Product Cross Sections, AAEC TM-549, June 1970

Item 4 Newsletter Bulletin 10, CCND-NW/10, December 1969

The 100-group cross-section sets developed have to be collapsed into effective one-energy group cross sections in order to perform the transmutation analysis. This collapsing was performed in the following manner.

A CTR blanket, labeled the standard blanket for identification purposes, has been the subject of intense neutronic analysis by several groups both in the United States and abroad. This blanket, while not necessarily optimum, was developed in 1971 at the ORNL International Working Session on the Engineering Problems of CTR Power Plants and by an ad-hoc committee representing the CTR neutronics community. It has since been used for purposes of comparison of the different analytical techniques of various laboratories.

It was decided to use this standard blanket as the vehicle for transmutation of the fission products. Therefore, using the ANISN code, the hundred group cross-section sets for the isotopes of interest were collapsed over region 6 of this blanket. The blanket is shown in Figure C.2.

The collapsing calculation was performed for a small isotopic content and, therefore the isotopes did not have a material effect on the spectrum of the blanket. The effect of this assumption will be tested in later analysis. As we shall see, the decision to load the fission products in region 6 was not particularly astute and later analysis will be performed with fission products in different blanket locations. The effective one-energy group constants obtained in this manner were then used to solve the differential equations of transmutation for ^{90}Sr and ^{137}Cs . The values obtained from this analysis are reported in the next section.

STANDARD BLANKET CONFIGURATION

DISTANCES IN em	0	150	200	200.5	203.5	204			264	294	300
ORIGIN	PLASMA	VAC.	Nb	Li 94	Nb		Li 94		C	Li 94	
				6 Nb			6 Nb			6 Nb	
							20 em	20 em	20 em		
ZONE NUMBER	1	2	3	4	5		6		7	8	
REGION NUMBER	1	2	3	4	5	6	7	18	9	10	
MATERIAL	A	B	C	D	C		D		E	D	
NUMBER OF INTERVALS PER ZONE	1	1	3	6	3		30		15	3	
THICKNESS (em)	150	50	0.5	3	0.5		60		30	6	

COMMENT: THE INTERVALS IN EACH ZONE ARE OF EQUAL STEP LENGTH. THERE ARE A TOTAL OF 62 INTERVALS IN THE PROBLEM.

FIGURE C.2

THE TRANSMUTATION CALCULATION

The transmutation chains displayed in Figure C.1 can be described by sets of coupled linear differential equations. Once described in equation form, they can then be solved for isotopic densities as a function of fluence, (i.e., flux x time).

The differential equations, when written in matrix notation, are solved by the ALCHEMY program. The basic input required is the neutron flux, the time in time steps, and the values for the coefficients of the transposed differential equation matrix. The values of the coefficients used in the analysis are summarized in Table C.3 for ^{137}Cs and ^{90}Sr .

In this analysis, one is primarily concerned with the parent/daughter balance. That is, one is trying to calculate the relative decrease in radioactivity as a function of reaction rate. Absolute destruction rates will be more accurately calculated in further analyses.

The calculation was performed using a nominal one-energy group neutron flux of $5 \times 10^{15} \text{ n/cm}^2 \text{ sec}$ and time steps out to three years. The flux level is somewhat higher than that usually associated with CTR power plants. This value was selected based upon the hope that by the time transmutation is applied in a CTR, that the technology will have advanced far enough to allow for the implied vacuum wall loading. If this high a value proves to be unrealistic, longer irradiation times will be required. In fact, one should keep in mind that what is being calculated is a reaction rate (i.e. $\sigma_i^j \phi t$). It will be shown later that a more astute choice of target location can greatly increase σ_i^j and therefore the reaction rate term. This would, in turn, either allow one to operate with smaller flux values or shorter irradiation times.

TABLE C.3

COEFFICIENT VALUES USED IN THE TRANSMUTATION ANALYSIS

The σ_i^1 and σ_i^2 represent the cross sections for the (n,y) and (n,2n) reactions with isotope i, respectively.

	^{137}Cs			^{90}Sr		
	σ_i^1 (barns)	σ_i^2 (barns)	λ_i (sec ⁻¹)	σ_i^1 (barns)	σ_i^2 (barns)	λ_i (sec ⁻¹)
N_1	0.0444	0.147	0.8261×10^{-9}	0.0188	0.148	0.76318×10^{-9}
N_2	0.1168		0.617×10^{-6}	0.010616		0.15855×10^{-6}
N_3			0.3587×10^{-3}			0.19850×10^{-4}
N_4	0.06711	0.06352		0.010714		
N_5	0.4565			0.032527		0.29991×10^{-5}
N_6	0.8652			0.041618		0.13598×10^{-6}
N_7	0.005912	0.1174				0.54544×10^{-4}
N_8			0.1359×10^{-3}	0.0051359		
N_9				0.073411	0.14192	
N_{10}					0.12315	

where: $N_1 = ^{137}\text{Cs}$ concentration
 $N_2 = ^{136}\text{Cs}$ concentration
 $N_3 = ^{138}\text{Cs}$ concentration
 $N_4 = ^{136}\text{Ba}$ concentration
 $N_5 = ^{135}\text{Ba}$ concentration
 $N_6 = ^{137}\text{Ba}$ concentration
 $N_7 = ^{138}\text{Ba}$ concentration
 $N_8 = ^{139}\text{Ba}$ concentration
 $N_9 = ^{139}\text{La}$ concentration

or: $N_1 = ^{90}\text{Sr}$ concentration
 $N_2 = ^{89}\text{Sr}$ concentration
 $N_3 = ^{91}\text{Sr}$ concentration
 $N_4 = ^{89}\text{Y}$ concentration
 $N_5 = ^{90}\text{Y}$ concentration
 $N_6 = ^{91}\text{Y}$ concentration
 $N_7 = ^{92}\text{Y}$ concentration
 $N_8 = ^{90}\text{Zr}$ concentration
 $N_9 = ^{91}\text{Zr}$ concentration
 $N_{10} = ^{92}\text{Zr}$ concentration

THE NEUTRON BALANCE MODEL

One of the potential problems associated with neutron transmutation of fission products is the propensity of daughter products to compete with their parents for neutrons. If this effect becomes profound, one may have to routinely separate out the stable daughter products from the target material. On the other hand, because of the possibility of achieving significant (n,2n) reactions in the high energy neutron flux of a CTR, the scheme can be a net producer of neutrons.

In order to have an analysis tool to monitor the above phenomena, a neutron balance model for each transmutation chain was developed. Quite simply, this is a method of calculating the ratio of neutron gain through (n,2n) reactions to neutron loss through (n,y) etc., reactions.

For both ^{137}Cs and ^{90}Sr the neutron balance can be shown to be

$$\text{Neutron Balance} = \frac{2}{1 + \frac{\sum_i N_i \sigma_i^1}{\sum_i N_i \sigma_i^2}}$$

where the superscripts 1 and 2 represent the (n,y) and (n,2n) reactions, respectively.

This equation can be used to calculate the neutron balances as a function of exposure. At time equal to zero the neutron balance for both chains become

$$\text{Neutron Balance} = \frac{2}{\frac{\sigma_i^1}{\sigma_i^2}}$$

Thus using the values in Table C.3 the neutron balance for ^{137}Cs and ^{90}Sr is 1.536 and 1.775, respectively. Therefore, at time zero, the transmutation chains for both ^{137}Cs and ^{90}Sr are net producers of neutrons.

The results of the ALCHEMY analysis will allow calculation of the neutron balance as a function of time.

$^{90}\text{Sr} + ^{137}\text{Cs}$
NEUTRON BALANCE

FIGURE C.3

THE DAUGHTER PRODUCTS

The large majority of the daughter products which are created are stable and after about an equivalent reaction rate of one year at 5×10^{15} flux level, are well out of a transient accumulation stage. Therefore, this analysis demonstrates that, for these particular isotopes in a neutron flux of this spectrum, the reduction in radioactivity of the fission products is paramount. That is, in large measure, transmutation of ^{90}Sr and ^{137}Cs , result in stable daughter products in a eTR flux. Therefore, this concept passes its first test for viability. Isotopes which on the periodic chart are bordered on the left and above by stable isotopes have an excellent chance of success with regard to neutron transmutation concepts. ^{90}Sr and ^{137}Cs meet these criteria and indeed transmute to stable daughter products. The probable rate at which this transmutation will occur is the subject of a later discussion.

THE NEUTRON BALANCE AND CONSEQUENT ECONOMIC IMPLICATIONS

The neutron balance for the ^{137}Cs and ^{90}Sr transmutation chains as a function of time are displayed in Figure C.3. These results are based upon the ALCHEMY transmutation calculation results reported earlier in this report. The results show that the ^{90}Sr and ^{137}Cs transmutation chains are net exporters of neutrons. In addition, these results are excellent indicators of the effect of the presence of daughter products in the loading. Notice that the neutron balance tends to decrease with time. This is due primarily to parasitic neutron absorption by the daughter products. Overall, it appears that for extended exposure times (up to three years at high flux levels) the neutron balance for both chains is only slightly depressed by this parasitic absorption. It would appear, therefore, that from a neutronics standpoint, early separation of the daughter products from the mixture is not of vital importance to the success of this concept.

The fact that these transmutation chains appear to be net exporters of fission products is also of major importance to the probable economics of the process.

If a transmutation process is a net absorber of neutrons, the process could prove to be quite expensive relative to other means of waste disposal. An initial evaluation of the minimum cost of such a process can be made by examining the worth in dollars of the neutrons consumed in the process. It is felt that the neutron worth can be the largest single cost factor in such a process. The best present and most probable future use for neutrons is in the generation of plutonium for fission power plants.

The present value of plutonium is of the order of \$6/gm. The value is expected to peak with the large-scale introduction of commercial Liquid Metal Fast Breeder Reactor (LMFBR) power plants. At that time, the value of plutonium is estimated to be \$15/gm. Because the LMFBRs are predicted to produce more plutonium than they require, the excess plutonium will probably be used in thermal neutron power plants. Large-scale production of plutonium from LMFBRs will tend to depress the plutonium value in the early 2000's and by the year 2020 plutonium should asymptotically approach a value of about \$2/gm. It is in this time frame that one could expect that a eTR fission product transmutation scheme might be employed. A neutron value of $(N.V.)_{Pu} = 7.94 \times 10^{-22}$ \$/neutron can be obtained based on a value of \$2.00/gm of plutonium and assuming plutonium production without neutron leakage or parasitic losses.

The current best alternative solution to the fission product problem, storage in deep salt caverns, has been estimated to cost of the order of 45 mills per megawatt hour electric ($MWhr_e$). If so, the cost of storage for fission products per megawatt hour thermal ($MWhr_{th}$) is then 15 mills, presuming an average of 33% efficiency in energy conversion.

One fission produces on the average two fission products and 3.2×10^{-11} W-sec or 1.152×10^{-21} $MWhr_{th}$. Only 5% of the fission products need be considered for transmutation purposes and, therefore, one can show that 8.681×10^{19} neutrons are required to transmute the fission products from one megawatt hour thermal. Thus this amount of neutrons would have a value of 15 mills when compared to long-term storage. The transmutation process has been estimated to be relatively inefficient ($\sim 50\%$) because of the low absorption cross sections which would double the number

of required neutrons to 1.736×10^{20} . Then, the Neutron Value when used for transmutation of fission products, (NV)F.P., can be estimated as 8.6×10^{-23} \$/neutron.

From these calculations, one finds that in 2020 a neutron would have ten times as much value when used for plutonium production as compared to being used for fission product transmutation. From this kind of preliminary argument, one might conclude that fission product transmutation will be of the order of ten times as expensive as the best current alternate solution to the problem. If the other isotopes of cesium are considered, the process becomes twenty times as expensive.

The conclusion that one draws from this discussion is that, if a transmutation process is a net neutron absorber, as would be likely with neutrons of energies below 8 MeV (the typical $n,2n$ threshold), the process is relatively expensive compared to storage. On the other hand, if transmutation is a net neutron producer, as seems likely in a CTR flux, then the neutrons produced can generate significant revenue. This opens up the possibility that fission product transmutation in a CTR could prove to be a revenue producing process and could perhaps be justified on economic grounds quite separate from its waste management function.

OTHER CONSIDERATIONS

Both the fusion-fission transmutation balance and possible loading patterns for fission products were studied on a preliminary basis. One question which was discussed earlier was the question of the minimum required flux level for adequate transmutation of fission products. This was established in a preliminary manner by reaction rate considerations. Another question of a scoping nature is to try to determine at what rate one might burn fission products in a CTR relative to their production in fission reactors. That is, one can try to find what fraction of the electric economy would have to be fusion power, considering the rest to be fission, in order that the particularly noxious fission products be transmuted at the rate that they are created.

A third question which must be explored is to determine, at least in a preliminary manner, just what fraction of the available neutrons can be reasonably absorbed in a fission product region of a CTR blanket.

Based on the preliminary analyses it is concluded that if one could load noxious fission products into the blanket such that the available neutrons were used with 50% efficiency, then 8% of the total power would have to be CTR power in order to transmute on a breakeven basis. Since the analysis was done on the basis of isotopically pure ^{137}Cs and ^{90}Sr , other isotopes in the mixture will tend to increase the CTR power fraction required.

Another conclusion is that more fission products need to be loaded into the blanket to achieve 50% utilization of neutrons that are available from an on-balance situation. This indicates that a few CTRs would be used to burn fission products rather than scattering the load out among many. Also, fission products can perhaps be loaded into CTR blankets without space limitations being a problem. However, further iterative calculations of a much more detailed nature are required to determine the proper loading scheme. It appears, however, that a loading of the order of 12 cm thickness in a CTR achieves about 50% available neutron utilization.

CONCLUSIONS AND SOME SPECULATIONS

The major conclusion which can be drawn about the work which has been done to date is that the process appears to be viable at this point. This conclusion, however, will not be firm until the previously noted unfinished portions of the analysis have been completed.

First of all, transmutation of the specific isotopes analyzed indicates that the radiological hazard is decreased. That is, the ultimate daughter products created are stable and for the most part are non-radioactive. This is absolutely necessary for the feasibility of any transmutation scheme. It is a function of the neutron flux spectrum and this conclusion does not necessarily apply to all neutron transmutation schemes.

Secondly, while a particularly attractive transmutation rate has not been demonstrated by the analysis to this point, it appears that if detailed parametric studies are performed on specific target loading techniques, that a transmutation rate several times that of the natural decay of these isotopes can be achieved.

It appears that there will be adequate physical space in a eTR blanket for fission product loadings and that the thermal burden of these fission products on the blanket will not be too great.

It has been shown that a few eTR power plants can, in principle, transmute all of the ^{90}Sr and ^{137}Cs created by the rest of the electrical economy, if the economy were all nuclear power.

The daughter products produced appear to be approaching well defined equilibriums after a year or so operation. They should, therefore, cause few perturbations on the characteristics of operating eTR power plants.

Finally, the question of the economics of eTR power plants as fission product burners was found to be most interesting. First of all, a transmutation scheme which is a net absorber of neutron would be relatively expensive. However, a eTR transmutation scheme, because of its unique neutronic characteristics, is a net producer of neutrons for long irradiation periods. This finding has a number of implications. It indicates that the process could be very cheap and indeed, could be a net producer of income. Secondly, it shows that the process does not require early separation of the daughter products. Finally, because certain fission products may prove to be of considerable value to eTR power plants, this finding has some implications as to what strategy should be used in Waste Storage.

In addition to the above conclusions, there are some inferences which can be drawn from the analysis which, while they border on being speculations, can perhaps be presented here as discussion points.

First of all, one should notice that transmutation is not an absolute method of removing all fission products from the environment. It probably can be used only on certain isotopes and therefore should be considered as one of an arsenal of tools to be used, in conjunction with other approaches, to solve the total waste management problem.

Secondly, because the process is by its nature exponential in character, it depends upon having a certain fission product inventory in order to be successful. As such, it will never completely eliminate all of the fission products which it is brought to bear on. That is, the

transmutation rate drops as the inventory goes down. Therefore, a transmutation scheme determines the absolute upper limit of the inventory of a particular isotope in the fuel cycle rather than bringing about its total removal from the environment.

A final speculation has to do with the application of CTR power plants to the transmutation of transuranium wastes (^{239}Pu , etc.). As a rule of thumb, the longer the natural half life of a particular isotope and the larger the cross section of that isotope, the more attractive that isotope is for transmutation. On this basis, while ^{90}Sr and ^{137}Cs have been found to be perhaps adequate transmutation targets, at least as far as this analysis has been carried, the CTR should be able to do a much better job on the transmutation of the transuranium wastes because, in general, their half lives are orders of magnitude higher, and their neutron cross sections are also significantly higher.. In addition, the CTR is likely to be able to outperform most other approaches if, in the final analysis, the CTR is found to have the high neutron flux and excess neutron capacity currently predicted. Another reason why CTR transmutation of transuranium isotopes appears particularly attractive in a CTR has to do with the specific neutronic characteristics of a CTR. Recalling that this analysis demonstrated the importance of high energy neutron threshold reactions (i.e., $n,2n$, etc.) in CTR transmutation, and looking at the cross sections for these isotopes for the $n,2n$, $n,3n$, and fission cross sections as a function of energy, one can tentatively conclude that these reactions would be of major importance in the transmutation of transuranium isotopes in a CTR. The net effect could be to transmute these isotopes back up the chain (i.e., $\text{Pu}^{240} \rightarrow \text{Pu}^{239}$) and ultimately, to fission them. The net effect could be a rapid decrease of the radiological burden on the environment. Of course, if the plutonium were originally bred in a CTR blanket, as has been proposed, it is likely to have a higher Pu^{239} content than present transuranium isotopic mixtures and therefore would be more attractive material both economically and environmentally.

D. Calculation of the Controlled Thermonuclear Standard Blanket
As Established by the ORNL Working Session Neutronics Committee:
Phases I and II - D. L. Prezbindowski

With the increasing success observed in controlled thermonuclear experiments, scientists and engineers are beginning to turn their attention toward the engineering problems of designing and building controlled thermonuclear power plants. The most likely first generation Controlled Thermonuclear Reactor (CTR) will operate on the deuterium-tritium fuel cycle in which most of the useful energy from the reaction is given off in an energetic (14.1 MeV) neutron. Moreover, this neutron must be used to breed tritium from lithium to continue the reaction as tritium is not available in nature. Therefore, it is most important that scientists and engineers interested in CTR technology have a good grasp on the physics properties of 14.1 MeV neutrons.

Introduction

At the Oak Ridge National Laboratory (ORNL) working session on Fusion Reactor Technology, held in June 1971, several neutronics calculations were reported describing the effect of 14.1 MeV neutrons slowing down in representative CTR blanket design. As has been noted previously, serious discrepancies have been observed between the work of several investigators. Therefore, at the neutronics section of the ORNL Working Session, it was suggested that a standard blanket model be analyzed by different groups so that the only variable is the method of calculation. An Ad-Hoc committee was formed and a standard blanket configuration was agreed upon. As part of its CTR technology effort, BNW has participated in this exercise.

The exercise was performed in two phases, both of which are reported here. In Phase I, ENDF/B Version II data was used along with a 100 group

gam - type energy structure. In Phase II, ENDF/B Version II data was used in both the 100 group structure and, in our case, a 50 equal lethargy group structure above 2.7 MeV. The group structure in this calculation below 2.7 MeV was not changed from Phase I.

Discussion

A description of the standard blanket is given below. This is followed by a discussion of the assumptions employed in my calculation. The items to be calculated are then described along with the particular options used in this exercise.

Standard Blanket Configuration

The agreed upon standard blanket model geometry is shown in Figure C.2.

The atomic densities used by all participants in this comparison are shown in Table D.1.

Assumptions of the Calculation

The calculation reported here generally follows the calculational assumptions suggested for all participants in the exercise. These are:

- One dimensional geometry was assumed.
- An isotopic source of 14 MeV neutrons was assumed to be distributed throughout the central 150 cm source region shown in Figure C.2.
- The source was initially normalized to an energy transport through the first wall of 10 MW/M^2 which is equivalent to $4.43 \times 10^{14} \text{ n/cm}^2 \text{ sec}$ of 14.06 MeV neutrons.
- The calculation was performed using a P-3, S-4 scattering description.
- The neutron energy group structure used was the 100 group GAM-II group structure and a 50 equal lethargy group structure in the energy range 14 MeV-2.7 MeV.

TABLE 0.1

STANDARD ATOMIC DENSITIES

Material Code* Letter	Constituent	Number Density
A	Isotropic flux source of neutrons	
B	Vacuum	
C	Niobium	0.05556 x 10 ²⁴ /c.c

	Niobium	0.003334 "
0	Lithium-6	0.003234 "
	Lithium-7	0.04038 "

E	Carbon	0.0804 "

*Refers to the material numbers in Figure 0.1

TABLE 0.2

NUCLEAR DATA USED FOR STANDARD BLANKET

Calculational Model

The SN transport theory as embodied in the ANISN code was used to calculate the multigroup fluxes and activations in one dimension. The cross section data used were from the ENOF/B-II and ENOF/B-III data files for Phase I and II, respectively. The data used is summarized in Table 0.2 by ENOF/B material number. The data used in the ANISN flux calculation for the first phase was processed into a 100 energy group structure with energy group boundaries identical to those in the GAM-II library. The processed data file was made available through RSIC under the label DLC-2C. Cross sections used to calculate the specific activations data for the reaction

$(n,2n)$ in Nb, (n,T) in Li-6 and $(n,n'T)$ in Li-7

were processed through ETOG at BNW. The data used in the ANISN flux calculation for the second phase was processed into a 100 energy group structure with energy group boundaries identical to those in the GAM-II library. The data was also processed into a 50 equal lethargy group library in the energy range 14.9 MeV to 2.7 MeV. The data was processed at Battelle-Northwest using ETOG-BNW. ETOG-BNW was derived from ETOG-I (WCAP-3845-1) as it was distributed by Westinghouse. The code was upgraded to be able to process ENDF/B-II form data and is currently being modified to handle the data on ENDF/B-III that specify options that have not been used before, such as the interference term in resonance scattering and the $LF = 1$ option for secondary neutron energy distributions. It does not calculate a scattering matrix for any reaction except (n,n') and $(n, 2n)$; however, the group averaged values for the remaining reactions are calculated for insertion by the user.

The calculations were made by the person solely responsible for maintaining the code on site. He has used ETOG since the first preliminary ENDF/B-I version was distributed by Dannels et al to SCWEG members.

Calculations - Phase I

All participants in this exercise were requested to perform the following calculations and report the results.

1. Calculate Nb(n,2n) reaction rates (normalized to one 14 MeV neutron) in each of the appropriate 45 intervals. Present graphs (with error bars for Monte Carlo results).
2. Calculate ${}^7\text{Li}(n,Tn)$ reaction rates in each of 39 intervals in the same manner.
3. Present tables for total Nb(n,2n) reaction rates and rates in regions 3, 4, 5, 6, 7, 8, and 10.
4. Present tables for total ${}^7\text{Li}(n,Tn)$ reaction rates and rates in regions 4, 6, 7, 8, and 10.

Calculations Performed - Phase II

All participants in this exercise were requested to perform the following calculations and report the results.

1. For each region calculate and report the total (n,2n) reaction rate, the total parasitic absorption rate, and also T_6 , T_7 and T as appropriate. Report the final neutron balance between summed (n,2n) rates, and summed parasitic absorption, T_6 , and boundary leakage.
2. Calculate and report the spatial distribution (by interval) of the flux associated with the source group, i.e., the top energy group.

3. Calculate and report the influence of using 50 energy groups between 14.9 MeV and 2.7 MeV instead of the 17 groups given by the GAM-II structure.
4. State the history and origin of the data files you use for Nb, ^6Li , ^7Li , and include names of the evaluations, and relationship to other well-known files.
5. Specify the nuclear data processing code used, its origin, parentage, how up-to-date a version it is, and how familiar with it the calculator is.
6. State the neutronics code used, its origin, parentage, etc.
7. Answer why you think that your results from the first round of action differed from other *O.D./M.C.* calculations.
8. Make any suggestions as to why there are differences of 15% or more between flux values for the source energy group in the first wall region seen between M.C. and D.O. calculations, (note in particular the ORNL contribution which used the same nuclear data, and the same processing code, for the *M.C./D.O.* comparison).

The results of the above calculations are shown in the next section.

Results of the Calculations

Results for Phase I are presented in Figures 0.1 and D.2 and Tables 0.3, 0.4, D.5 and 0.6 for the calculations requested. The apparently low reaction rates in region number 4 on Figures 0.1 and 0.2 are due to the relatively smaller step lengths in this region. The calculated breeding ratio for the blanket is 1.24786. The (n,2n) reaction cross section of the specified Nb ENDF/B data is too low by about a factor of 3. There is a corresponding positive error in the inelastic cross sections which tends to over-soften the neutron spectrum. These effects reinforce each other to produce a Nb (n,2n) reaction rate and breeding ratio which appear too low.

The results of the calculations for Phase II are presented in Tables 0.7, 0.8 and 0.9 for the calculations requested. Table 0.7 contains the total n,2n reaction rates, total parasitic absorption rate, T_6 , T_7 , and T and final neutron balance between summed (n,2n) rates, and summed parasitic absorption, T_6 , and boundary leakage. Table 0.8 contains the spatial distribution by interval of the flux associated with the source group, i.e., the top energy group. Table D.8 includes fluxes for the 100 group calculation between 14.9 and 2.7 MeV using the GAM-II group structure and for the 50 group calculation between 14.9 and 2.7 MeV using equal lethargy intervals.

The influence of using 50 groups in the energy range 14.9 MeV and 2.7 MeV was tested by generating a 50 group ANISN cross section library from ENDF/B-III data with a lower cutoff at 2.7 MeV. Comparisons with the 100 group calculation which covers the range 0-14.9 MeV are made in terms of the source group spatial flux distribution and the n,2n and ${}^7\text{Li}$ n,n' α zone reaction rates. These reactions were chosen because their

FIGURE D.1

FIGURE 0.2

lower thresholds are above the 2.7 MeV cutoff, and, therefore, insensitive to the difference in the lower energy cutoff between the two problems. The top three groups in the 50 group structure span approximately the same energy range as the top group in the GAM-II group structure. (The 50 group structure used is included in Table 0.10.)

In the 50 group calculation the neutron source was placed in the second group which spans the interval 14.4 to 13.9 MeV. This is consistent with a nominal 14.1 MeV energy of neutrons produced in the O-T reaction. This placement of the source, of course, resulted in a zero flux in group 1. The effect on the spatial flux distribution of the source group in going to the finer group structure can be seen in Table 0.8. In interval 62, the flux from the 50 group calculation is more than a factor of 2 lower than the flux from the 10a group calculation. This reflects the fact that the source flux in the 50 group calculation sees a larger average total cross section than the source flux in the 10a group calculation. The GAM-II group structure should be refined in the MeV region if one wants calculations with better than a factor of 2 accuracy in the source group flux.

The effect of the group structure refinement on the $n,2n$ reaction rate and T_7 can be seen by comparing Tables 0.7 and 0.9. It can be seen that the $n,2n$ reaction rate is generally lower in the Li-Nb mixture and higher in the pure Nb regions in the 50 group calculation. The total $n,2n$ reaction rate is 17% lower than that of the 10a group calculation. The T_7 reaction rate is consistently larger in all zones in the 50 group calculation. The total T_7 in the 50 group calculation is 11% larger than the T_7 in the 100 group calculation.

TABLE 0.3

INTERVAL	RADIUS	Nb (n, 2n) ACTIVATION RATE	
		By Interval* - Phase I	
		Nb	Act Rate
		(Events/Sec)	MeV n
1	0.000		
2	150.000		
3	200.000	8.19563	- 3
4	200.167	7.82592	- 3
5	200.333	7.47779	- 3
6	200.500	1.29005	- 3
7	201.000	1.22999	- 3
8	201.500	1.17570	- 3
9	202.000	1.12483	- 3
10	202.500	1.07752	- 3
11	203.000	1.03154	- 3
12	203.500	5.48094	- 3
13	203.667	5.25254	- 3
14	203.833	5.03588	- 3
15	204.000	3.29018	- 3
16	206.000	2.81191	- 3
17	208.000	2.41998	- 3
18	210.000	2.09632	- 3
19	212.000	1.82724	- 3
20	214.000	1.60121	- 3
21	216.000	1.41116	- 3
22	218.000	1.24948	- 3
23	220.000	1.11088	- 3
24	222.000	9.91199	- 4
25	224.000	8.87154	- 4
26	226.000	7.96438	- 4
27	228.000	7.16398	- 4
28	230.000	6.45621	- 4
29	232.000	5.82778	- 4
30	234.000	5.26753	- 4
31	236.000	4.76633	- 4
32	238.000	4.31667	- 4
33	240.000	3.91237	- 4
34	242.000	3.54807	- 4
35	244.000	3.21937	- 4
36	246.000	2.92241	- 4
37	248.000	2.65393	- 4
38	250.000	2.41113	- 4
39	252.000	2.19156	- 4
40	254.000	1.99314	- 4
41	256.000	1.81411	- 4
42	258.000	1.65305	- 4
43	260.000	1.50883	- 4
44	262.000	1.38075	- 4
45	264.000		
46	266.000		
47	268.000		
48	270.000		
49	272.000		

* Intervals are defined in Figure 0.1

$$\sigma_{\Sigma} = \mathbf{0.073024}$$

TABLE 0.4

Li-7 (n,Tn) ACTIVATION RATE

BY INTERVAL - PHASE I

INTERVAL	RADIUS	Li-7 Act Rate (Events/sec/14 MeV n)
1	000.000	
2	150.000	
3	200.000	
4	200.167	
5	200.333	
6	200.500	1.38516 - 2
7	201.000	1.33306 - 2
8	201.500	1.28573 - 2
9	202.000	1.24110 - 2
10	202.500	1.19895 - 2
11	203.000	1.15680 - 2
12	203.500	
13	203.367	
14	203.833	
15	204.000	3.84654 - 2
16	206.000	3.40362 - 2
17	208.000	3.02056 - 2
18	210.000	2.68838 - 2
19	212.000	2.40028 - 2
20	214.000	2.14855 - 2
21	216.000	1.93035 - 2
22	218.000	1.73916 - 2
23	220.000	1.57092 - 2
24	222.000	1.42219 - 2
25	224.000	1.29013 - 2
26	226.000	1.17268 - 2
27	228.000	1.06733 - 2
28	230.000	9.72721 - 3
29	232.000	8.87503 - 3
30	234.000	8.10531 - 3
31	236.000	7.40832 - 3
32	238.000	6.77600 - 3
33	240.000	6.20152 - 3
34	242.000	5.67890 - 3
35	244.000	5.20317 - 3
36	246.000	4.76997 - 3
37	248.000	4.37556 - 3
38	250.000	4.01679 - 3
39	252.000	3.69096 - 3
40	254.000	3.39586 - 3
41	256.000	3.12976 - 3
42	258.000	2.89158 - 3
43	260.000	2.68079 - 3
44	262.000	2.49781 - 3
45	264.000	
46	266.000	
47	268.000	
48	270.000	
49	272.000	

TABLE 0.5

Nb (n,2n) ACTIVATION RATE
BY REGION* - PHASE I

<u>REGION</u> $\rightarrow$	<u>RADIUS</u>	<u>Nb</u> <u>(Reactions/sec/</u> <u>14 MeV n)</u>
1.	000.000	
2	150.000	
3	200.000	
4	200.167	2.34003 - 2
5	200.333	6.92963 - 3
6	200.500	1.57693 - 2
7	201.000	1.88096 - 2
8	201.500	5.80946 - 3
9	202.000	2.17482 - 3
10	202.500	3.14924 - 5
		<hr/>
		$\Sigma = 7.30237 - 2$

* Regions are defined in Figure 0.1

TABLE 0.6

Li-7 (n, Tn) ACTIVATION nATE
BY REGION - PHASE I

REGION	RADIUS	Li-7 <u>{Reactions/sec/14 Mev</u>
1	000.000	
2	150.000	
3	200.000	
4	200.167	7.60077 - 2
5	200.333	
6	200.500	2.41706 - 1
7	201.000	8.80736 - 2
8	201.500	3.66523 - 2
9	202.000	
10	202.500	6.85408 - 4
		<hr/>
		$\Sigma = 4.43124 - 1$

TABLE 0.7.

CALCULATED REACTION RATES IN EACH ZONE OF THE STANDARD BLANKET
FROM THE 100 GROUP CALCULATION GAM-II GROUP STRUCTURE - PHASE II

Region	(n,2n)*	ABS**	T6***	T7****	T*****
3	5.36772, -02	2.24044 -02	0.0	0.0	0.0
4	2.65150 -02	5.15957 -02	4.16127 -02	7.74893'-02	1.19102- -01
5	3.62694 -02	2.22044 -02	0.0	0.0	0.0
6	7.22511 -02	3.09755"-01	2.51739: -01	2.69716 -01	5.21454 -01
7	2.20367 -02	2.55532 -02	2.04924, -01	1.06667' -01	3.11592 -01
8	8.02672 -03	3.09743 -01	2.56817' -01	4.52222' -02	3.02039 -01
10	1.06496' -04	5.62025' -02	5'.47634 --02	8.24783 ' -04	5.55881 -02
Total	2.18883 -01	1.02743' +00	8.09852" -01	4.99918' -01	1.30976 +00

Total Leakage from System: 4.060681 -02

- * The units of the table entries are (events/sec/14 MeV neutron).
 ** σ_a does not include reactions that produce secondary neutrons.
 *** T6 = Tritium production via $^6\text{Li} (n, \alpha) \text{T}$ Reaction.
 **** T7 = Tritium production via $^7\text{Li} (nf, n' \alpha) \text{T}$ Reaction.
 ***** T = T6 + T7.

TABLE 0.8

SPATIAL DISTRIBUTION OF THE SOURCE GROUP NEUTRON FLUX
FOR THE 100 GROUP AND 50 GROUP CALCULATIONS - PHASE II

SPACE POINTS	RADIUS	100 GROUP CALCULATIONS	SPACE POINTS	RADIUS	100 GROUP CALCULATIONS
1	0	2.53978 - 03	36	2.46000 + 02	2.71256 - 05
2	1.50000 + 02	1.56227	37	2.48000	2.39316
3	2.00000	1.35414	38	2.50000	2.11211
4	2.00167	1.28769	39	2.52000	1.86456
5	2.00333	1.22488	40	2.54000	1.64633
6	2.00500	1.15880	41	2.56000	1.45380
7	2.01000	1.09019	42	2.58000	1.28377
8	2.01500	1.02639	43	2.60000	1.13347
9	2.02000	9.67059 - 04	44	2.62000	1.00043
10	2.02500	9.11843	45	2.64000	8.61142 - 06
11	2.03000	8.60441	46	2.66000	7.20612
12	2.03500	8.16911	47	2.68000	6.02440
13	2.03667	7.80354	48	2.70000	5.03315
14	2.03833	7.45669	49	2.72000	4.20311
15	2.04000	6.56840	50	2.74000	3.50888
16	2.06000	5.29840	51	2.76000	2.92875
17	2.08000	4.32182	52	2.78000	2.4424
18	2.10000	3.56248	53	2.80000	2.03976
19	2.12000	2.96521	54	2.82000	1.70220
20	2.14000	2.48993	55	2.84000	1.42053
21	2.16000	2.10731	56	2.86000	1.18560
22	2.18000	1.79582	57	2.88000	9.89672 - 07
23	2.20000	1.53954	58	2.90000	8.26381
24	2.22000	1.32658	59	2.92000	6.90410
25	2.24000	1.14800	60	2.94000	5.91784
26	2.26000	9.97691 - 05	61	2.96000	5.21718
27	2.28000	8.69328	62	2.98000	4.59948
28	2.30000	7.59418			
29	2.32000	6.64807			
30	2.34000	5.83002			
31	2.36000	5.11999			
32	2.38000	4.50177			
33	2.40000	3.96206			
34	2.42000	3.48984			
35	2.44000	3.07593			

TABLE 0.8 (cont1d)

SPACE POINTS	50 GROUP CALCULATIONS	SPACE POINTS	50 GROUP CALCULATIONS
1	2.57326 -03	36	1.67883 - 05
2	1.56813	37	1.45505
3	1.35319	38	1.26148
4	1.28484	39	1.09390
5	1.22026	40	9.48766 - 06
6	1.14847	41	8.22986
7	1.07087	42	7.13917
8	9.99435 - 04	43	6.19278
9	9.33660	44	5.37087
10	8.73047	45	4.54580
11	8.17174	46	3.73478
12	7.72328	47	3.06824
13	7.37217	48	2.51962
14	7.03910	49	2.06854
15	6.11274	50	1.69791
16	4.78363	51	1.39356
17	3.79584	52	1.14370
18	3.05140	53	9.38620 - 07
19	2.48221	54	7.70326
20	2.04064	55	6.32234
21	1.69316	56	5.18936
22	1.41675	57	4.25995
23	1.19302	58	3.49775
24	1.01030	59	2.87302
25	8.59559 - 05	60	2.42007
26	7.34110	61	2.09811
27	6.28925	62	1.81904
28	5.40174		
29	4.64896		
30	4.00768		
31	3.45824		
32	2.98878		
33	2.58516		
34	2.23754		
35	1.93773		

TABLE 0.9

CALCULATED REACTION RATES IN EACH ZONE OF THE STANDARD BLANKET
FROM THE 50 GROUP CALCULATION REFINED GROUP STRUCTURE - PHASE- II

<u>REGION</u>	<u>n,2n*</u>	<u>T₇**</u>
3	5.57391' -02	0.0
4	1.76280- -02	8.5321 -02
5	3.78974' -02	0.0
6	4.92133' -02	3.0065 -01
7	1.54843- -02	1.20897-01
8	5.76866- -03	5.21797-01
10	7.9032- -04	1.2146- -03
TOTAL	1.81809 -01	5.6026' -01

* The units of the table entries are (event/sec/ 14 MeV neutron)

** T₇ = Tritium Production Via ⁷Li (nf,n'α)T Reaction

TABLED.10
50 GROUP MULTIGROUP STRUCTURE

GROUP	ENERGY RANGE	LETHARGY RANGE	GROUP	ENERGY RANGE	LETHARGY RANGE
1	1.4420+07 - 1.4918+07	-.400 - -.366	26	6.1631+06 - 6.3763+06	.450 - .484
2	1.3938+07 - 1.4420+07	-.366 - -.332	27	5.9571+06 - 6.1631+06	.484 - .518
3	1.3472+07 - 1.3938+07	-.332 - -.298	28	5.7580+06 - 5.9571+06	.518 - .552
4	1.3021+07 - 1.3412+07	-.298 - -.264	29	5.5655+06 - 5.7580+06	.552 - .586
5	1.2586+07 - 1.3021+07	-.264 - -.230	30	5.3794+06 - 5.5655+06	.586 - .620
6	1.2165+07 - 1.2586+07	-.230 - -.196	31	5.1996+06 - 5.3794+06	.620 - .654
7	1.1759+07 - 1.2165+07	-.196 - -.162	32	5.0258+06 - 5.1996+06	.654 - .688
8	1.1366+07 - 1.1759+07	-.162 - -.128	33	4.8578+06 - 5.0258+06	.688 - .722
9	1.0986+07 - 1.1366+07	-.128 - -.094	34	4.6954+06 - 4.8578+06	.722 - .756
10	1.0618+07 - 1.0986+07	-.094 - -.060	35	4.5384+06 - 4.6954+06	.756 - .790
11	1.0263+07 - 1.0618+07	-.060 - -.026	36	4.3867+06 - 4.5384+06	.790 - .824
12	9.9203+06 - 1.0263+07	-.026 - .008	37	4.2401+06 - 4.3867+06	.824 - .858
13	9.5887+06 - 9.9203+06	.008 - .042	38	4.0984+06 - 4.2401+06	.858 - .892
14	9.2682+06 - 9.5887+06	.042 - .076	39	3.9614+06 - 4.0984+06	.892 - .926
15	8.9583+06 - 9.2682+06	.076 - .110	40	3.8289+06 - 3.9614+06	.926 - .960
16	8.6589+06 - 8.9583+06	.110 - .144	41	3.7009+06 - 3.8289+06	.960 - .994
17	8.3694+06 - 8.6589+06	.144 - .178	42	3.5772+06 - 3.7009+06	.994 - 1.028
18	8.0896+06 - 8.3694+06	.178 - .212	43	3.4576+06 - 3.5772+06	1.028 - 1.062
19	7.8192+06 - 8.0896+06	.212 - .246	44	3.3421+06 - 3.4576+06	1.062 - 1.096
20	7.5578+06 - 7.8192+06	.246 - .280	45	3.2303+06 - 3.3421+06	1.096 - 1.130
21	7.3052+06 - 7.5578+06	.280 - .314	46	3.1223+06 - 3.2303+06	1.130 - 1.164
22	7.0610+06 - 7.3052+06	.314 - .348	47	3.0180+06 - 3.1223+06	1.164 - 1.198
23	6.8250+06 - 7.0610+06	.348 - .382	48	2.9171+06 - 3.0180+06	1.198 - 1.232
24	6.5968+06 - 6.8250+06	.382 - .416	49	2.8196+06 - 2.9171+06	1.232 - 1.266
25	6.3763+06 - 6.5968+06	.416 - .450	50	2.7253+06 - 2.8196+06	1.266 - 1.300

Differences in calculational results can be caused by a wide variety of problems ranging from key punch errors to theoretical inadequacies. It is difficult, if not impossible, to give technically sound responses to requests 7 and 8 without at least a detailed study of the computer code outputs of the particular calculations involved.

Acknowledgments

The author would like to thank B. R. Leonard, Jr. for many helpful suggestions and reviews of the work. Also K. B. Stewart and M. G. Zimmerman for assistance in running the computer calculations and cross-section data generation.

E. FUSION CROSS SECTION THEORY - B. H. Duane

Previous work of this sort, as reviewed sequentially by Post,⁴³ by Glasstone,⁴⁴ by Rose,⁴⁵ and by Artsimovich,⁴⁶ appears to have been limited to use of Gamow's⁴⁷ charge barrier penetration probability

$$p = (q_1 q_2 / 2\epsilon_0 \hbar v) \exp(-q_1 q_2 / 2\epsilon_0 \hbar v), \quad (1)$$

which is merely the low energy limit of Mott's⁴⁸ more broadly relevant **charge barrier penetration probability**

$$p = (q_1 q_2 / 2\epsilon_0 \hbar v) / [\exp(q_1 q_2 / 2\epsilon_0 \hbar v) - 1]. \quad (2)$$

In both expressions, v is the incident approach velocity of charges q_1 and Q_2' , $\hbar$ is the angular momentum quantum, and ϵ_0 is the dielectric permittivity of space (in rationalized mks units). (In cgs units, ϵ_0 is replaced by $1/4\pi$.) Mott derives the charge barrier penetration probability (2) by use of Temple's⁴⁹ exact solution of the Schroedinger equation for two charged particles as a confluent hypergeometric wave function of parabolic coordinates.

Observing that the high energy limit ($v \rightarrow \infty$) of Gamow's penetration probability (1) is zero (as it manifestly should not be), and that the high energy limit of Mott's penetration probability (2) is unity (as it manifestly should be), may help to explain persistent previous failures of Gamow fits to follow fusion cross section measurements satisfactorily above 13-20 keV.

Multiplying Mott's charge barrier penetration probability (2) by a flexible truncated summation of Breit-Wigner⁵⁰ nuclear resonance cross sections leads to fitting fusion cross section measurements $\sigma(E)$, as a function of laboratory bombardment energy E , to the theoretical form

$$\sigma(E, \underline{A}) = [E \{ \exp(A_1 / \sqrt{E}) - 1 \}]^{-1} \sum_{n=1}^N A_{3n-1} / [1 + (A_{3n} E - A_{3n+1})^2]. \quad (3)$$

The parameters $\underline{A} = (A_1, A_2, A_3, A_4, A_5, \dots)$ appearing here are evaluated by nonlinear least-squares fitting to available measurements. As computer programmed in BNW Program Learn-III,⁵¹ the parameter search proceeds by iterative minimization of the inverse-variance-weighted sum-of-squares of discrepancies between measurements $\sigma(E_m)$ and theory $\sigma(E_m, \underline{A})$,

$$Q(\underline{A}) = \sum_{m=1}^M [\sigma(E_m) - \sigma(E_m, \underline{A})]^2 / [\delta\sigma(E_m)]^2, \quad (4)$$

where $\delta\sigma(E_m)$ is the root-mean-square measurement uncertainty for $\sigma(E_m)$, and M is the number of measurements.

Subsequent to achievement of solid (36-bit) convergence on the best fit, nonlinear statistical dispersion analysis⁵¹ yields the root-mean-square parameter uncertainties $\delta\underline{A}$ as the square roots of the diagonal elements of the mean square correlated error tensor,

$$\langle (\delta\underline{A})(\delta\underline{A}) \rangle = [Q(\underline{A}) / (M-J)] \cdot [(1/2) \partial^2 Q(\underline{A}) / \partial \underline{A} \partial \underline{A}]^{-1}, \quad (5)$$

where J is the number of fitted parameters. Automated nonlinear statistical regression analysis,⁵¹ finally, investigates exhaustively the possibility of eliminating parameters without significantly deteriorating the quality of the fit.

Measurements

The cross section measurements for the six most important fusion reactions $[T(d,n)He^4, He^3(d,p)He^4, D(d,p)T, D(d,n)He^3, T(t,2n)He^4, T(He^3,X)Y]$ shown in Tables E.1-E.6 reproduce in tabular form the comprehensive (1956)

graphical compilation edited by Jarmie and Seagrave.⁵² The $T(d,n)He^4$ measurements are those of Arnold,⁵³ Conner,⁵⁴ Argo,⁵⁵ Bame,⁵⁶ Hemmendinger,⁵⁷ Stratton,⁵⁸ and Brolley.⁵⁹ The $He^3(d,p)He^4$ measurements are those of Jarvis,⁶⁰ Arnold,⁶¹ Kunz,⁶² Frier,⁶³ Yarnell,⁶⁴ Bonner,⁶⁵ and Allred.⁶⁶ The $D(d,p)T$ measurements are those of Arnold,⁶⁷ Wenzel,⁶⁸ Blair,⁶⁹ Preston,⁷⁰ Davenport,⁷¹ and Eliot.⁷² The $D(d,n)He^3$ measurements are those of Arnold,⁷³ Smith,⁷⁴ Hunter,⁷⁵ Blair,⁷⁶ Preston,⁷⁷ and Erickson.⁷⁸ The $T(t,2n)He^4$ measurements are those of Agnew.⁷⁹ The $T(He^3,X)Y$ measurements are those of Moak.⁸⁰ These data are shown as machine-reproduced Fortran card images both to minimize transcription error and to provide detailed reproducibility for this entire analysis.

Best Theory

The best theoretical representations for these six fusion reactions [$T(d,n)He^4$, $He^3(d,p)He^4$, $D(d,p)T$, $D(d,n)He^3$, $T(t,2n)He^4$, $T(He^3,X)Y$] are presented in both graphic and analytic form in Figs. [E.1,E.2,E.3,E.4,E.5, E.6J respectively. The preceding measurements $a(E_m)$ and their root-mean-square uncertainties $\delta\sigma(E_m)$ are machine plotted as circles split by vertical bars. (The horizontal bar on the right of each circle is an irrelevant quirk of the plotting routine.) The best theory $\sigma(E,A)$ and its root-mean-square uncertainty $\delta\sigma(E,A)$ are machine plotted as three panned lines, which merge where the theory is firm and diverge where the theory is shaky. The best theoretical equation, its best parameters, and their root-mean-square uncertainties appear below each plot. Each theoretical representation is the best of three or more exploratory fits using both fewer and more parameters.

Table E.1: $T(d,n)He^4$ FUSION CROSS SECTION MEASUREMENTS

DEUTERON KINETIC ENERGY (LABORATORY ELECTRON VOLTS) E_m , m=1,77								
7.580+3,	8.100+3,	9.100+3,	10.40+3,	13.20+3,	15.70+3,	16.20+3,	16.30+3,	TD EVOI
19.00+3,	21.00+3,	22.50+3,	25.40+3,	26.00+3,	30.30+3,	31.30+3,	33.50+3,	TD EV02
35.20+3,	36.50+3,	40.20+3,	43.00+3,	46.00+3,	52.50+3,	53.00+3,	60.40+3,	TO EV03
62.50+3,	67.00+3,	73.50+3,	76.50+3,	83.00+3,	88.00+3,	93.50+3,	98.00+3,	TD EV04
105.0+3,	110.0+3,	116.0+3,	119.0+3,	127.0+3,	131.0+3,	138.0+3,	144.0+3,	TD EV05
150.0+3,	157.0+3,	160.0+3,	168.0+3,	175.0+3,	186.0+3,	188.0+3,	208.0+3,	TO EV06
217.0+3,	228.0+3,	252.0+3,	255.0+3,	268.0+3,	305.0+3,	328.0+3,	355.0+3,	TD EV07
385.0+3,	405.0+3,	500.0+3,	655.0+3,	757.0+3,	800.0+3,	1000.+3,	101Q.+3,	TD EV08
1270.+3,	1500.+3,	1750.+3,	2000.+3,	2225.+3,	250,0.+3,	3000.+3,	3500.+3,	TO EV09
4000.+3,	4600.+3,	5050.+3,	6000.+3,	1050.+4,				TD EVIO
$T(d,n)He^4$ FUSION CROSS SECTION (BARNS) $a(E_m)$, m=1,77								
27.30-5,	42.50-5,	92.50-5,	162.0-5,	780.0-5,	17.00-3,	2140.-5,	21.20-3,	TDBARN01
4460.-5,	61.80-3,	84.00-3,	143.0-3,	157.0-3,	273.0-3,	293.0-3,	394.0-3,	TDBARN02
440.0-3,	512.0-3,	725.0-3,	740.0-3,	1120.-3,	1480.-3,	1620.-3,	2160.-3,	TDBARN03
2300.-3,	2830.-3,	3150.-3,	3625.-3,	4150.-3,	4450.-3,	4650.-3,	4850.-3,	TDBARN04
5050.-3,	4900.-3,	5150.-3,	4750.-3,	4800.-3,	4600.-3,	4350.-3,	4200.-3,	TDBARN05
3950.-3,	3715.-3,	3520.-3,	3350.-3,	3050.-3,	3040.-3,	2750.-3,	2475.-3,	TDBARN06
2230.-3,	2040.-3,	1770.-3,	1650.-3,	1500.-3,	1240.-3,	1100.-3,	965.0-3,	TDBARN07
765.0-3,	825.0-3,	665.0-3,	390.0-3,	393.0-3,	296.0-3,	290.0-3,	268.0-3,	TDBARN08
207.5-3,	168.0-3,	145.0-3,	127.5-3,	117.5-3,	107.5-3,	100.0-3,	94.50-3,	TDBARN09
91.00-3,	90.00-3,	81.50-3,	86.00-3,	47.50-3,				TDBARNIO
FRACTIONAL PROBABLE ERROR $.674489768*oa(E_m)/a(E_m)$, m=1,77,								
10.00-2,	10.00-2,	9.000-2,	9.000-2,	8.000-2,	7.000-2,	6.000-2,	5.000-2,	TD FPEOI
5.000-2,	4.000-2,	4.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.00Q-2,	TD FPEQ2
3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.00Q-2,	TD FPEQ3
3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.00Q-2,	TD FPEQ4
3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	3.000-2,	TO FPEQ5
3.000-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	TO FPE06
15.00-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	15.00-2,	TD FPEQ7
15.00-2,	15.00-2,	7.000-2,	15.00-2,	7.000-2,	15.00-2,	3.000-2,	7.000-2,	TD FPE08
7.000-2,	1.000-2,	7.000-2,	7.000-2,	10.00-2,	7.000-2,	7.000-2,	7.000-2,	TD FPEQ9
7.000-2,	7.000-2,	1.000-2,	7.000-2,	12.00-2,				TD FPEIO

Table E.2: $\text{He}^3(\text{d,p})\text{He}^4$ FUSION CROSS SECTION MEASUREMENTS

DEUTERON KINETIC ENERGY (LABORATORY ELECTRON VOLTS) E_m , m=1,82							
35.40+3,	39.50+3,	42.50+3,	45.10+3,	49.00+3,	54.50+3,	58.00+3,	62.50+3,HE3 EV01
64.00+3,	66.00+3,	67.00+3,	74.00+3,	76.00+3,	80.00+3,	84.00+3,	81.00+3,HE3 EV02
92.00+3,	93.00+3,	107.5+3,	135.0+3,	160.0+3,	190.0+3,	193.0+3,	216.0+3,HE3 EV03
240.0+3,	245.0+3,	247.5+3,	270.0+3,	285.0+3,	300.0+3,	305.0+3,	320.0+3,HE3 EV04
325.0+3,	355.0+3,	368.0+3,	373.0+3,	380.0+3,	380.0+3,	400.0+3,	410.0+3,HE3 EV05
410.0+3,	420.0+3,	430.0+3,	430.0+3,	440.0+3,	450.0+3,	455.0+3,	460.0+3,HE3 EV06
470.0+3,	475.0+3,	480.0+3,	485.0+3,	487.0+3,	500.0+3,	500.0+3,	510.0+3,HE3 EV07
520.0+3,	520.0+3,	530.0+3,	535.0+3,	550.0+3,	550.0+3,	565.0+3,	510.0+3,HE3 EV08
600.0+3,	600.0+3,	650.0+3,	675.0+3,	710.0+3,	155.0+3,	180.0+3,	825.0+3,HE3 EV09
840.0+3,	885.0+3,	905.0+3,	970.0+3,	1100.+3,	1225.+3,	1375.+3,	1475.+3,HE3 EV10
1700.+3,	1000.+4,						HE3 EV11
$\text{He}^3(\text{d,p})\text{He}^4$ FUSION CROSS SECTION (BARNs) $a(E_m)$, m=1,82							
11.13-5,	25.50-5,	21.70-5,	58.90-5,	81.50-5,	155.1-5,	200.0-5,	280.0-5,HEBARN01
330.0-5,	295.0-5,	401.0-5,	610.0-5,	720.0-5,	650.0-5,	10.12-3,	11.75-3,HEBARN02
15.00-3,	10.75-3,	18.25-3,	42.00-3,	81.00-3,	133.0-3,	201.5-3,	192.5-3,HEBARN03
284.0-3,	272.5-3,	365.0-3,	365.0-3,	465.0-3,	455.0-3,	550.0-3,	680.0-3,HEBARN04
545.0-3,	620.0-3,	810.0-3,	927.0-3,	670.0-3,	870.0-3,	850.0-3,	710.0-3,HEBARN05
930.0-3,	920.0-3,	710.0-3,	890.0-3,	933.0-3,	890.0-3,	705.0-3,	910.0-3,HEBARN06
815.0-3,	640.0-3,	945.0-3,	670.0-3,	820.0-3,	180.0-3,	880.0-3,	110.0-3,HEBARN07
610.0-3,	750.0-3,	700.0-3,	560.0-3,	560.0-3,	700.0-3,	670.0-3,	640.0-3,HEBARN08
480.0-3,	600.0-3,	420.0-3,	490.0-3,	385.0-3,	410.0-3,	350.0-3,	342.0-3,HEBARN09
310.0-3,	277.0-3,	307.0-3,	243.0-3,	195.0-3,	180.0-3,	175.0-3,	147.0-3,HEBARN10
135.0-3,	44.00-3,						HEBARN11
FRACTIONAL PROBABLE ERROR $.674489768 * 8a(E_m)/a(E_m)$, m=1,82							
20.40-2,	16.18-2,	6.682-2,	14.43-2,	9.202-2,	9.677-2,	1.500-2,	5.357-2,HE3FPE01
4.545-2,	13.56-2,	4.364-2,	4.098-2,	4.167-2,	10.77-2,	3.155-2,	2.128-2,HE3FPE02
3.333-2,	9.385-2,	11.64-2,	11.61-2,	9.877-2,	11.28-2,	11.04-2,	11.04-2,HE3FPE03
25.89-2,	11.01-2,	10.96-2,	10.96-2,	20.97-2,	5.495-2,	5.505-2,	16.92-2,HE3FPE04
5.505-2,	5.645-2,	13.58-2,	1.551-2,	4.478-2,	10.92-2,	12.94-2,	5.634-2,HE3FPE05
8.065-2,	7.065-2,	5.634-2,	10.11-2,	6.967-2,	10.67-2,	5.674-2,	6.593-2,HE3FPE06
8.452-2,	4.851-2,	6.878-2,	4.851-2,	9.146-2,	10.26-2,	11.36-2,	7.143-2,HE3FPEQ7
5.327-2,	8.333-2,	5.714-2,	5.327-2,	5.357-2,	5.357-2,	8.955-2,	3.516-2,HE3FPE0a
4.082-2,	7.500-2,	4.082-2,	4.082-2,	4.878-2,	4.878-2,	5.117-2,	5.111-2,HE3FPE09
4.397-2,	4.397-2,	4.397-2,	4.397-2,	4.397-2,	4.397-2,	4.397-2,	4.391-2,HE3FPE10
4.397-2,	3.000-2,						HE3FPE11

Table E.3: D(d,p)T FUSION CROSS SECTION MEASUREMENTS

DEUTERON KINETIC ENERGY (LABORATORY ELECTRON VOLTS) E_m , m=1,55									
13.12+3,	15.20+3,	17.80+3,	20.20+3,	17.00+3,	19.50+3,	20.25+3,	22.30+3.00	EVe1	
25.00+3,	28.00+3,	29.00+3,	30.00+3,	33.00+3,	35.20+3,	35.50+3,	38.50+3,DD	EV02	
41.00+3,	49.80+3,	50.00+3,	51.00+3,	60.50+3,	69.00+3,	70.00+3,	19.00+3,DO	EV03	
81.00+3,	90.00+3,	100.0+3,	100.0+3,	111.0+3,	115.0+3,	134.0+3,	152.0+3,OD	EV04	
155.0+3,	163.0+3,	181.0+3,	200.0+3,	205.0+3,	250.0+3,	210.0+3,	-300.0+3,OD	EV05	
310.0+3,	345.0+3,	360.0+3,	400.0+3,	430.0+3,	455.0+3,	410.0+3,	505.0+3,DD	EV06	
550.0+3,	960.0+3,	1500.+3,	2000.+3,	2500.+3,	3000.+3,	3500.+3,	DO	EV07	
D(d,p)T FUSION CROSS SECTION (BARNS) $\sigma(E_m)$, m=1,55									
.3380-4,	.7100-4,	1.420-4,	2.630-4,	.1420-3,	.2480-3,	.2630-3,	.4210-3,DO8ARNO1		
.6200-3,	.8550-3,	1.125-3,	1.125-3,	1.450-3,	1.800-3,	1.900-3,	2.550-3,DDBARN02		
2.550-3,	4.800-3,	4.350-3,	6.400-3,	6.440-3,	9.400-3,	8.800-3,	11.20-3,D·08ARN03		
11.20-3,	13.25-3,	14.80-3,	15.50-3,	17.00-3,	17.70-3,	19.10-3,	23.70-3,DOBARN04		
26.00-3,	27.20-3,	28.90-3,	30.50-3,	34.70-3,	36.60-3,	45.00-3,	43.00-3,DDBARN05		
41.50-3,	47.80-3,	52.00-3,	52.50-3,	50.00-3,	55.60-3,	61.25-3,	59.30-3,DOBARN06		
61.80-3,	77.30-3,	85.00-3,	90.00-3,	89.50-3,	89.60-3,	89.60-3,	DD8ARN01		
FRACTIONAL PROBABLE ERROR $.674489768*\sigma(E_m)/\sigma(E_m)$, m=1,55									
4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	6.000-2,	4.000-2,	4.000-2,00	FPE01	
4.000-2,	4.000-2,	6.000-2,	4.000-2,	4.000-2,	4.000-2,	5.000-2,	6.000-2,00	fPE02	
4.000-2,	4.000-2,	5.000-2,	8.000-2,	4.000-2,	5.000-2,	4.000-2,	8.000-2,00	FPE03	
4.000-2,	4.000-2,	4.000-2,	5.000-2,	4.000-2,	4.000-2,	8.000-2,	5.000-2,DO	FPE04	
8.000-2,	8.000-2,	8.000-2,	5.000-2,	8.000-2,	5.000-2,	8.000-2,	5.000-2,00	FPE05	
8.000-2,	5.000-2,	8.000-2,	5.000-2,	8.000-2,	5.000-2,	8.000-2,	5.000-2,00	FPE06	
5.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	DO	FPE01	

Table IV: $D(d,n)He^3$ FUSION CROSS SECTION MEASUREMENTS

Table E.4:

DEUTERON KINETIC ENERGY (LABORATORY ELECTRON VOLTS) E_m , m=1,39								
13.10+3,	19.20+3,	22.50+3,	25.50+3,	31.00+3,	30.50+3,	36.50+3,	40.80+3,DO	EV01
46.80+3,	52.00+3,	60.00+3,	67.30+3,	14.00+3,	81.00+3,	92.00+3,	100.0+3,OD	EVQ2
115.0+3,	155.0+3,	210.0+3,	270.0+3,	370.0+3,	470.0+3,	970.0+3,	990.0+3,00	EVQ3
1350.+3,	1500.+3,	1150.+3,	1995.+3,	2200.+3,	2500.+3,	2500.+3,	2800.+3,00	EVQ4
2950.+3,	3100.+3,	3400.+3,	3450.+3,	3700.+3,	4970.+3,	1020.+4,	DO	EV05
$D(d,n)He^3$ FUSION CROSS SECTION (BARNS) $\sigma(E_m)$, m=1,39								
3.400-5,	20.60-5,	38.20-5,	61.00-5,	112.5-5,	1.150-3,	1.990-3,	2.57Q-3,DDBARN01	
3.660-3,	5.200-3,	6.920-3,	8.750-3,	10.25-3,	12.40-3,	15.30-3,	17.QQ-3,DDBARN02	
19.60-3,	28.80-3,	38.50-3,	51.00-3,	60.00-3,	73.00-3,	87.00-3,	96.0Q-3,DDBARN03	
105.0-3,	103.0-3,	105.0-3,	105.0-3,	105.0-3,	102.0-3,	105.0-3,	1Q3.Q-3,DDBARN04	
103.0-3,	103.0-3,	95.00-3,	102.0-3,	103.0-3,	93.00-3,	69.00-3,	DDBARN05	
FRACTIONAL PROBABLE ERROR $.674489768*8a(E_m)$, m=1,39								
4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,DD	FPE01
4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,	4.000-2,OD	FPEQ2
4.000-2,	8.000-2,	8.000-2,	8.000-2,	8.000-2,	8.000-2,	4.000-2,	15.00-2,DD	FPEQ3
15.00-2,	4.000-2,	15.00-2,	4.000-2,	15.00-2,	4.000-2,	15.00-2,	15.00-2,DO	FPE04
4.000-2,	15.00-2,	15.00-2,	4.000-2,	15.00-2,	6.000-2,	10.00-2,	DD	FPEQ5

Table E.5: $T(t,2n)He^4$ FUSION CROSS SECTION MEASUREMENTS

TRITON KINETIC ENERGY (LABORATORY ELECTRON VOLTS) E_m , $m=1,43$

37.50+3,	87.50+3,	113.0+3,	125.0+3,	133.0+3,	155.0+3,	175.0+3,	191.5+3,TT	EVO1
225.0+3,	232.5+3,	250.0+3,	315.0+3,	335.0+3,	420.0+3,	430.0+3,	520.0+3,TT	EV02
602.0+3,	620.0+3,	690.0+3,	105.0+3,	780.0+3,	810.0+3,	875.0+3,	900.0+3,TT	EV03
975.0+3,	1000.+3,	1060.+3,	1155.+3,	1160.+3,	1185.+3,	1205.+3,	1225.+3,TT	EV04
1275.+3,	1313.+3,	1320.+3,	1400.+3,	1465.+3,	1560.+3,	1650.+3,	1773.+3,TT	EV05
1908.+3,	2075.+3,	2220.+3,					TT	EV06

$T(t,2n)He^4$ FUSION CROSS SECTION (BARNs) $a(E_m)$, $m=1,43$

11.00-3,	17.00-3,	11.75-3,	22.25-3,	34.25-3,	25.50-3,	30.00-3,	30.15-3,TTBARN01
34.50-3,	43.00-3,	36.25-3,	42.25-3,	51.25-3,	54.25-3,	60.25-3,	60.75-3,TTBARN02
65.15-3,	64.75-3,	72.00-3,	75.15-3,	75.75-3,	79.00-3,	77.00-3,	86.50-3,TTBARN03
89.00-3,	100.0-3,	94.75-3,	104.0-3,	106.8-3,	124.0-3,	88.50-3,	108.3-3,TTBARN04
110.0-3,	114.8-3,	130.5-3,	125.0-3,	157.8-3,	127.5-3,	136.5-3,	142.3-3,TTBARN05
151.5-3,	172.5-3,	180.5-3,					TTBARN06

FRACTIONAL PROBABLE ERROR $.674489768 * \sigma a(E_m) / a(E_m)$, $m=1,43$

10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,TT	FPE01
10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,TT	FPE02
10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,TT	FPE03
10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,TT	FPE04
10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,	10.00-2,TT	FPE05
10.00-2,	10.00-2,	10.00-2,					TT	FPE06

Table E.6: T(He³,X)Y FUSION CROSS SECTION MEASUREMENTS

He ³ KINETIC ENERGY (LABORATORY ELECTRON VOLTS)	E _m , m=1,21							
100.0+3, 155.0+3, 200.0+3, 235.0+3, 275.0+3, 217.5+3, 295.0+3, 327.5+3,	THE3							EV1
355.0+3, 375.0+3, 395.0+3, 435.0+3, 482.5+3, 512.5+3, 565.0+3, 597.5+3,	THE3							EV2
631.5+3, 682.5+3, 712.5+3, 760.0+3, 800.0+3,	THE3							EV3
T(He ³ ,X)Y FUSION CROSS SECTION (BARNS)	a(E _m), m=1,21							
4.550-4, 3.725-3, 9.000-3, 1.550-2, 2.625-2, 3.000-2, 2.900-2, 4.10Q-2,	THE3							81
5.000-2, 5.425-2, 6.980-2, 7.850-2, 8.950-2, 9.900-2, 1.050-1, 1.124-1,	THE3							82
1.200-1, 1.415-1, 1.450-1, 1.700-1, 1.800-1.	THE3							83
FRACTIONAL PROBABLE ERROR	.674489768*oa(E _m)/a(E _m), m=1,21							
20.00-2, 20.00-2, 20.00-2, 20.00-2, 20.00-2, 20.00-2, 20.00-2, 2Q.OQ-2,	THE3FPE1							
20.00-2, 20.00-2, 20.00-2, 20.00-2, 20.00-2, 20.00-2, 20.00-2, 2Q.OQ-2,	THE3FPE2							
20.00-2, 20.00-2, 20.00-2, 20.00-2, 20.00-2,	THE3FPE3							

The $T(d,n)He^4$ fit in Fig. E.1 reveals a type of residual flaw painfully familiar to anyone who has done least-squares fitting to resonance cross sections. An otherwise beautiful fit is marred by one prominent discrepancy at the high energy end of the measurement information. Such an end-point anomaly cannot be fit firmly to another 3-parameter resonance without additional measurements delineating its mean energy and mean width." Fortunately, this lone discrepancy at 10.5 MeV is unimportant here, since it lies two decades above the 10-100 keV threshold rise important in fusion work.

Both the $He^3(d,p)He^4$ fit in Fig. E.2 and the $D(d,p)T$ fit in Fig. E.3 approach perfection, with the theory needle threading nicely through all available measurements within measurement scatter.

Careful scrutiny of the $D(d,n)He^3$ fit in Fig. E.4, in frustrating contrast, suggests the need for further work on the nuclear theory for this reaction. The measurements quite clearly show an unusual level-flat-top resonance. The best-fit theory shows a tilted-flat-top resonance persistently lower than the measurements. The threshold rise from 10-1000 keV is represented adequately for fusion work, but we cannot yet claim to fully understand the $D(d,n)He^3$ reaction near its 1-10 MeV peak.

The $T(t,2n)He^4$ fit in Fig. E.5 reflects a need for more measurements, both in the 10-100 keV range to define the threshold rise and in the 2.22-10 MeV range to define the peak and the high energy tail. The uncertainties in the fitted parameters here stem principally from paucity of relevant data.

The superb fit to the $T(He^3X)Y$ data shown in Fig. E.6 leads to the prediction that further measurements in the 1-10 MeV range will find that the reaction cross section peaks at $.48 \pm .02$ barns at $6.0 \pm .1$ MeV.

$T(d,n)He^4$ FUSION CROSS SECTION $\sigma(E)$ AS FUNCTION OF ENERGY E

FIGURE E.1

Measurements: ϕ : Charged Particle Cross Sections (LA-2014 March 1956)

Theory: $\equiv$ • Nonlinear least squares fit, BNW Prog. Learn (BNWL-390)

$$\sigma(E) = [E \{ \exp(A_1 \sqrt{E}) - 1 \}]^{-1} [A_2 / \{ 1 + (A_3 E - A_4)^2 \} + A_5]$$

A_1	$1453 \pm 14 \sqrt{eV}$
A_2	$(502 \pm 26) \times 10^5 \text{ eV-barn}$
A_3	$(1368 \pm 45) \times 10^{-8} \text{ eV}$
A_4	$1.076 \pm .055$
A_5	$(409 \pm 15) \times 10^3 \text{ eV-barn}$

$\text{He}^3(\text{d,p})\text{He}^4$ FUSION CROSS SECTION $\sigma(E)$ AS FUNCTION OF ENERGY E

FIGURE E.2

Measurements: ϕ : Charged Particle Cross Sections (LA-2014 March 1956)

Theory: $\equiv$. Nonlinear least squares fit, BNW Prog. Learn (BNWL-390)

$$\sigma(E) = [E \{ \exp(A_1/\sqrt{E}) - 1 \}]^{-1} [A_2 / \{ 1 + (A_3 E - A_4)^2 \} + A_5]$$

A_1	$2823 \pm 35 \sqrt{\text{eV}}$
A_2	$(259 \pm 21) \times 10^5 \text{ eV barn}$
A_3	$(398 \pm 13) \times 10^{-8} \text{ eV}$
A_4	$1.297 \pm .072$
A_5	$(647 \pm 41) \times 10^3 \text{ eV barn}$

D(d,p)T FUSION CROSS SECTION $\sigma(E)$ AS FUNCTION OF ENERGY E

FIGURE E.3

Measurements: Φ : Charged Particle Cross Sections (LA-2014 March 1956)

Theory: $\equiv \equiv \equiv$ Nonlinear least squares fit, BNW Prog. Learn (BNWL-390)

$$\sigma(E) = [E \{ \exp(A_1/E) - 1 \}] - I [A_2 / \{ 1 + (A_3 E - A_4)^2 \}]$$

$$A_1 = 1457.7 \pm 5.0 \sqrt{eV}$$

$$A_2 = (372 \pm 10) \times 10^3 \text{ eV-barn}$$

$$A_3 = (436 \pm 26) \times 10^{-9} \text{ eV}$$

$$A_4 = 1.220 \pm .038$$

D(d,n)He³ FUSION CROSS SECTION $\sigma(E)$ AS FUNCTION OF ENERGY E

FIGURE E.4

Measurements: Φ : Charged Particle Cross Sections (LA-2014 March 1956)

Theory: $\equiv \equiv \equiv$ Nonlinear least squares fit, BNW Prog. Learn (BNWL-390)

$$\sigma(E) = [E\{\exp(A_1/\sqrt{E})-1\}]^{-1} [A_2/\{1+(A_3E-A_4)^2\}]$$

$$\begin{aligned} A_1 &= 1514 \pm 13 \sqrt{\text{eV}} \\ A_2 &= (482 \pm 29) \times 10^3 \text{ eV barn} \\ A_3 &= (308 \pm 37) \times 10^{-9} \text{ eV} \\ A_4 &= 1.177 \pm .077 \end{aligned}$$

$T(t,2n)He^4$ FUSION CROSS SECTION $a(E)$ AS FUNCTION OF ENERGY E

FIGURE E.5

Measurements: ϕ : Charged Particle Cross Sections (LA-2014 March 1956)

Theory: $\equiv \equiv \equiv$ Nonlinear least squares fit, BNW Prog. Learn (BNWL-390)

$$a(E) = [E\{\exp(A_1/IE)-1\}]-1[A_2/\{1+(A_3E-A_4)^2\}]$$

$$\begin{aligned} A_1 & 1214 \pm 72 \sqrt{eV} \\ A_2 & (448 \pm 59) \times 10^3 \text{ eV-barn} \\ A_3 & (102 \pm 10) \times 10^{-8} \text{ eV} \\ A_4 & 2.09 \pm .14 \end{aligned}$$

T(He³,X)Y FUSION CROSS SECTION $\sigma(E)$ AS FUNCTION OF ENERGY E

FIGURE E-6

Measurements: ϕ : Charged Particle Cross Sections (LA-2014 March 1956)

Theory: $\equiv$: Nonlinear least squares fit, BNW Prog. Learn (BNWL-390)

$$\sigma(E) = [E\{\exp(A_1/\sqrt{E})-1\}]^{-1}[A_2]$$

A ₁	3893 $\pm$ 34 $\sqrt{eV}$	Probability			
A ₂	(1125 $\pm$ 68)x10 ⁴ eV·barn	(225-600 keV)		x	y
		.51 $\pm$.02	p+n		He ⁴
		.43 $\pm$.02	d		He ⁴
		.06 $\pm$.02	p		He ⁵

Thermal Averages

Thermally averaging the best theory $\sigma(E, \underline{A})$ for each fusion reaction over a Maxwellian distribution of reacting ions at temperature kT yields

$$\langle v\sigma(kT) \rangle = \int_0^\infty v\sigma(E, \underline{A}) [\exp(-mv^2/ZkT)] V^2 dv / \int_0^\infty [\exp(-mv^2/ZkT)] v^2 dv, \quad (6)$$

where $E = m_2 v^2/s$ is the laboratory kinetic energy with which the beam ion of mass m_2 strikes the target ion of mass m_1 , and $m = m_1 m_2 / (m_1 + m_2)$ is the reduced mass of the ion pair. In fusion design work, appropriate numeric integration automated to achieve four-figure integration accuracy by successive integration-mesh refinement dynamically generates thermal averages from Eq. (6), as illustrated in Table E.7. The temperature mesh in Table E.7 was chosen for preliminary design optimization studies, except for the last five entries which provide a final design fine mesh centered about the spontaneous ignition temperature of 19.33 keV for unconfined D-T.

Summary

Quantum theory functional forms describing charge barrier penetration followed by nuclear resonance interaction have been least-squares fitted to all available cross section measurements for the six most important fusion reactions $[T(d,n)He^4, He^3(d,p)He^4, D(d,p)T, D(d,n)He^3, T(t,2n)He^4, T(He^3, X)Y]$. The best fits are presented in both graphic and analytic form in Figs. E.1-E.6 and are reduced to thermal averages in Table E.7. Further work is needed on the nuclear theory of the $D(d,n)He^3$ reaction.

Acknowledgments

Most of the data processing was performed by Mrs. P. Z. Aldridge. The work was funded by the United States Atomic Energy Commission.

Table E.7: FUSION REACTIVITY $\langle\sigma v\rangle(kT)$ AS FUNCTION OF ION TEMPERATURE kT

Ion Temperature kT (keV)	$T(d,n)He^4$ $\langle\sigma v\rangle$ ($m^3 Is$)	$He^3(d,p)He^4$ $\langle\sigma v\rangle$ ($m^3 Is$)	$D(d,p)T$ $\langle\sigma v\rangle$ (m^3/s)	$D(d,n)He^3$ $\langle\sigma v\rangle$ ($m^3 Is$)	$T(t,2n)He^4$ $\langle\sigma v\rangle$ ($m^3 Is$)	$T(He^3,X)Y$ $\langle\sigma v\rangle$ ($m^3 Is$)
10.00	1.087×10^{-22}	2.233×10^{-25}	5.809×10^{-25}	6.259×10^{-25}	7.243×10^{-25}	1.156×10^{-26}
20.00	4.240×10^{-22}	3.785×10^{-24}	2.432×10^{-24}	2.725×10^{-24}	2.519×10^{-24}	2.623×10^{-25}
30.00	6.649×10^{-22}	1.451×10^{-23}	4.753×10^{-24}	5.410×10^{-24}	4.554×10^{-24}	1.134×10^{-24}
40.00	8.021×10^{-22}	3.224×10^{-23}	7.200×10^{-24}	8.259×10^{-24}	6.624×10^{-24}	2.805×10^{-24}
50.00	8.701×10^{-22}	5.435×10^{-23}	9.652×10^{-24}	1.111×10^{-23}	8.686×10^{-24}	5.287×10^{-24}
60.00	8.969×10^{-22}	7.813×10^{-23}	1.206×10^{-23}	1.391×10^{-23}	1.074×10^{-23}	8.515×10^{-24}
70.00	8.999×10^{-22}	1.016×10^{-22}	1.442×10^{-23}	1.663×10^{-23}	1.280×10^{-23}	1.240×10^{-23}
80.00	8.892×10^{-22}	1.236×10^{-22}	1.672×10^{-23}	1.927×10^{-23}	1.489×10^{-23}	1.685×10^{-23}
90.00	8.708×10^{-22}	1.435×10^{-22}	1.898×10^{-23}	2.182×10^{-23}	1.701×10^{-23}	2.178×10^{-23}
100.00	8.484×10^{-22}	1.612×10^{-22}	2.118×10^{-23}	2.431×10^{-23}	1.918×10^{-23}	2.712×10^{-23}
200.00	6.276×10^{-22}	2.432×10^{-22}	4.133×10^{-23}	4.624×10^{-23}	4.267×10^{-23}	9.177×10^{-23}
300.00	4.953×10^{-22}	2.494×10^{-22}	5.845×10^{-23}	6.500×10^{-23}	6.337×10^{-23}	1.606×10^{-22}
400.00	4.168×10^{-22}	2.386×10^{-22}	7.194×10^{-23}	8.117×10^{-23}	7.679×10^{-23}	2.256×10^{-22}
500.00	3.666×10^{-22}	2.253×10^{-22}	8.173×10^{-23}	9.465×10^{-23}	8.384×10^{-23}	2.852×10^{-22}
600.00	3.327×10^{-22}	2.130×10^{-22}	8.831×10^{-23}	1.054×10^{-22}	8.655×10^{-23}	3.397×10^{-22}
700.00	3.085×10^{-22}	2.025×10^{-22}	9.237×10^{-23}	1.136×10^{-22}	8.654×10^{-23}	3.895×10^{-22}
800.00	2.908×10^{-22}	1.936×10^{-22}	9.453×10^{-23}	1.196×10^{-22}	8.494×10^{-23}	4.352×10^{-22}
900.00	2.774×10^{-22}	1.863×10^{-22}	9.530×10^{-23}	1.239×10^{-22}	8.242×10^{-23}	4.772×10^{-22}
1000.00	2.670×10^{-22}	1.802×10^{-22}	9.508×10^{-23}	1.266×10^{-22}	7.943×10^{-23}	5.161×10^{-22}
19.00	3.937×10^{-22}	3.140×10^{-24}	2.216×10^{-24}	2.478×10^{-24}	2.322×10^{-24}	2.142×10^{-25}
19.25	4.013×10^{-22}	3.294×10^{-24}	2.270×10^{-24}	2.539×10^{-24}	2.371×10^{-24}	2.256×10^{-25}
19.33	4.039×10^{-22}	3.347×10^{-24}	2.288×10^{-24}	2.559×10^{-24}	2.387×10^{-24}	2.295×10^{-25}
19.50	4.089×10^{-22}	3.453×10^{-24}	2.323×10^{-24}	2.601×10^{-24}	2.420×10^{-24}	2.374×10^{-25}
19.75	4.165×10^{-22}	3.617×10^{-24}	2.377×10^{-24}	2.663×10^{-24}	2.469×10^{-24}	2.496×10^{-25}

F. ECONOMIC REGIME FOR FUSION REACTORS - D. E. Deonigi, R. L. Engel, and J. R. Young

An overall goal for the fusion reactor development efforts is construction of a competitive fusion reactor plant - a facility that the power industry will voluntarily build and operate because it produces electricity at a lower cost than the competitive power plant or because it has intangible advantages, such as a lower environmental impact, that offset a higher electricity cost. Consequently, there is a need for a definition of the competitive fusion plant as an aid in determination of (1) the goal for the development efforts and (2) the progress toward obtaining a competitive plant.

An exploratory study was made to determine the electricity cost for the barely competitive fusion reactor late in this century and early in the next century. A primary purpose of the effort was to explore the calculational methods to determine the feasibility of the selected approach.

Summary

The electricity cost for the barely competitive fusion power plant was determined by forecasting the electricity costs for the competitive power plants during the period of 1972 to 2020. The forecast was made by use of the Systems Analysis Task Force LP computer model of the U. S. Power Industry developed by PNL for cost-benefit analysis of the U. S. fast breeder development effort.

The forecast was based on the primary assumptions that the U. S. Power Industry would contain fossil plants, light water reactors (LWRs), high temperature gas reactors (HTGRs), and liquid metal fast breeder reactors

(LMFBRs), and that the cost of uranium will advance from the current price of about \$7/lb to as much as \$50/lb as the lower cost ores are consumed.

The relevant results of the forecast are presented in Table F.1 as (1) a schedule for construction of each type of power plant, (2) the electricity production cost for the least competitive type of base-loaded power plant being built during each biennium (i.e., having the highest average electricity production cost during its 30-year life), and (3) the unit capital cost for the barely competitive fusion reactor. These electricity and capital costs represent the maximum permissible ones for the barely competitive fusion reactor, assuming that only economics are being considered. Intangible aspects, such as environmental effects, also must be considered and could result in the competitive reactor having an electricity production cost significantly different than the costs in Table F.1.

The results obtained showed that the calculational methods are feasible. However, the results should not be interpreted as the economic characteristics of the fusion reactors that would be built. The barely competitive fusion power plant has a benefit-cost ratio of 0.0. Additional studies of various fusion plants with lower electricity production costs are necessary to determine the economic characteristics of the ones with benefit-cost ratios greater than one. These additional studies will require several additional forecasts of the power plant construction schedules - the several forecasts exploring the introduction of fusion reactors with various economic characteristics and introduction dates.

Characteristics of the Power Industry Forecast

The LP Computer Model of the U.S. Power Industry forecasts the number and types of power plants that will be built in the U.S. through the year

TABLE F.1

FORECAST U. S. POWER PLANT CONSTRUCTION SCHEDULE
1990 to 2019

Period	Number of 1000 MWe Power Plants Built					Unit Electricity Cost for Plants Started During Period (mill s/kWh)	Unit Capital Cost for Barely Competitive Fusion Reactor.* (\$/kW)
	Fossil	LWR	HTGR	LMFBR	Total		
1990-1991	14	58	39	32	143	8.7	440
1992-1993		58	35	57	150	6.6	330
1994-1995		58	30	88	176	5.7	280
1996-1997		51	18	118	186	5.6	280
1998-1999		35	18	158	211	5.7	280
2000-2001		11	18	195	225	5.8	290
2002-2003		7	18	229	254	6.2	310
2004-2005			2	260	263	6.4	320
2006-2007			9	288	298	7.0	350
2008-2009			9	303	312	6.3	320
2010-2011		37	9	304	350	6.5	330
2012-2013		37	9	320	367	5.4	270
2014-2015		49	9	353	411	5.0	250
2016-2017		49	9	374	432	4.8	240
2018-2019		65	7	412	484	4.7	230

*Assuming, 0.3 mills/kWh for fueling, operation, and maintenance costs, annual return on capital, and an 0.85 plant factor.

2020, and also calculates the total cost for producing electricity during that period. The benefits from introduction of a power plant of different design can be determined by comparing the electricity production costs for two cases - one with and one without that type power plant. The assumptions and input data used for making the forecast of this study were similar to those used in the updated Breeder Benefit-Cost Study,⁸¹ The important assumptions necessary for defining the general characteristics of this forecast are presented in Tables F.2 and F.3 below.

TABLE F.2

BASIC ASSUMPTIONS FOR U. S. POWER INDUSTRY FORECAST

1. Characteristics of Power Plants Included in Forecast:.

Type Power Plant	Introduction Date	Fuel Assumptions
Fossil	1972	Six price regions (25 to 41¢/10 ⁶ btu)
LWR	1972	Five potential fuel cycles
HTGR	1972	
Reference LMFBR	1986	
Advanced LMFBR	1994	

2. Separative work costs:

- a. \$32/SWU for 1977 to 1984
- b. \$35/SWU after 1984

3. Coal costs rise 1%/yr.

4. The uranium cost increases progressively from \$7 to \$50/lb as the low cost ores are consumed.

5. A cost discount rate of 7%.

TABLEF.3

LB CAPITAL COST DATA				
Capital Costs in Dollars per Kilowatt				
Decision Year	Plant Type			LMFBR
	Fossil	LWR	HTGR	
1970	188	222		
1980	183	"212	220	
(1984)				241
1990	179	204	208	242
2000	176	196	199	212
2010	174	192	192	197
2020	172	188	186	188
2030	170	186	181	183
2040	169	184	178	180
2070	166	183	175	177

G. GRAPHITE FOR CONTROLLED THERMONUCLEAR REACTORS - G. L. Tingey

The environment of Controlled Thermonuclear Reactors (CTR) places stringent temperature and radiation requirements on core materials. Most current CTR conceptual designs include graphite as one of the candidate materials. Although there are major problems in the use of graphite, its characteristic properties such as high temperature stability, low neutron capture cross-section, and moderating ability make it the foremost candidate for the moderator, the reflector, and shielding of CTR. In fact, some have suggested its use for other applications including the vacuum wall.

Introduction

Engineering studies, both in the U.S. and U.K., have favored the deuterium-tritium (D-T) fuel cycle for the first power producing CTR. In this system the high energy neutrons produced by the D-T reaction are either captured in the tritium generating blanket or are moderated in the graphite. These neutrons have the capacity to displace carbon atoms from their lattice positions and produce a volume change in the graphite as well as a change in most physical properties. These irradiation induced changes are strongly dependent upon temperature, neutron fluence, and type of graphite. Figure G.1 shows the volume change for two types of graphite as a function of irradiation fluence at various irradiation temperatures.

Expected Range of CTR Conditions

Although current CTR designs do not define in detail the materials or the environment of the reactor, the temperatures for the moderator and reflector graphites will likely be limited on the hot side by the temperature of the lithium metal or molten salt in the tritium generating blanket and

RADIATION-INDUCED VOLUME CHANGES IN GRAPHITE AT VARIOUS TEMPERATURES

Figure G.1

on the cold side by the temperature of the coolant between the graphite and the radiation shield. For the ORNL standard blanket, depicted in Figure G.2, the maximum temperatures are expected to be about 600°C if liquid lithium is used for the blanket or *BOOae* if the molten salt (LiF-BeF_2) is used. Minimum graphite temperatures for the moderator are predicted to fall in the range of 200-600°C.

Likewise the design does not fix the total neutron fluence but, assuming a 20 year plant life, the graphite would be expected to receive irradiation exposure producing in excess of 75 displacements per atom (10^{23} neutron/cm²).

Experimental Program Required

As already indicated graphite for CTR may be subjected to a fast neutron fluence producing about 75 displacements/atom at temperatures ranging from 200-800°C. Although continuing efforts have been expended to obtain graphite data for several years, these programs have emphasized fission reactor conditions and, thus, information directly applicable to CTR is very lacking. For example, the highest fluence data available extend only to about 25 displacements/atom and this is limited to a temperature range of between 600-700°C. At lower temperatures the information is even more limited. Figure G.3 depicts graphically the anticipated data required for a CTR along with that currently available. Data are needed on dimensional changes, physical property changes, irradiation creep behavior, compatibility of the irradiated graphite with molten salts and lithium, and stored energy at the temperatures and fluences shown in Figure G.3.

RANGE OF ANTICIPATED CTR GRAPHITE TEMPERATURES IN ORNL STANDARD CTR BLANKET CONFIGURATION

Figure G.2

NEEDED AND AVAILABLE IRRADIATION DATA ON GRAPHITE FOR CTR

FIGURE G.3

On the basis of the current dimensional change data, there is little hope that conventional graphites can sustain the projected CTR fluences and maintain structural soundness. On the other hand, a significant improvement has been made in irradiation stability in new types of graphite. At present it is uncertain whether or not further improvements can be made which will permit the fabrication of a system sufficiently stable for CTR purposes. An experimental program is needed to evaluate the potential of graphite for these applications.

H. THERMAL AND HYDRAULIC ANALYSIS OF A HYBRID MACHINE - A. M. Sutey

To determine the technical feasibility of fusion-fission hybrid systems, PNL is performing parameteric analytical studies of natural uranium, graphite, helium cooled lattices. A conceptual design has been developed in an attempt to characterize a fission blanket system which will provide maximum power multiplication of DT source neutrons. To date, the dimensions of the lattice concepts have been arbitrarily chosen to be suitable primarily from a physics standpoint. Of course, it is necessary to evaluate these concepts on a thermal hydraulic basis. Such an evaluation is needed to further define the technical feasibility of the hybrid concept.

Description of Work and Expected Results

It is suggested that a thermal-hydraulic evaluation be performed of the hybrid CTR conceptual design (which has been previously described in this report). A one-dimensional, conduction and convection heat transfer analysis should be performed to determine whether the various regions of the hybrid system can be sufficiently cooled using the coolant, configuration, dimensions" and extent of coolant channels assumed for the hybrid concept. A cylindrical or donut shaped unit cell of the hybrid blanket (as shown in Figure B.3) with proposed dimensions and estimated flux profiles should be considered. Because the heat transfer associated with the unit cell is expected to be similar to that of an HTGR, analytical methods similar to those used for gas cooled reactors should be employed. Helium is a good choice to consider as the primary coolant and should be used to cool the graphite, lithium, and other blanket regions of the system. Coolant channels will comprise approximately 10% of the system. It is expected that the percentage

of coolant channels will decrease with distance from the vacuum wall since the heat load will decrease with distance from the vacuum wall. Temperature profiles and maximum power densities in the hybrid blanket should be determined. In particular, a thermal hydraulic analysis of the power peaking which appears to be exhibited at the vacuum wall is of interest.

This analysis should help identify the relative merits of the hybrid system under consideration with regard to choice of coolant and geometry and is expected to help define potentially desirable system concepts for more detailed engineering studies.

I. HYDROGEN ISOTOPES EFFECTS: SOLUBILITY AND DIFFUSIVITY OF HYDROGEN ISOTOPES IN CANDIDATE CTR WALL METALS - A. B. Johnson, Jr., L. A. Charlot and R. E. Westerman

Recent advances in controlled thermonuclear physics led to the concept of controlled thermonuclear reactors (CTR) as a future power source. Of the major metal design problems in a projected CTR plant, the permeation of hydrogen isotopes through materials of construction is important for either containment of tritium or for regulated transfer of tritium to where it can be recycled to a reactor fuel system. Present estimates⁸³ for CTR plant design indicate that core construction metals, such as niobium, vanadium or molybdenum, will be exposed to temperatures between 800 and 1000°C and tritium partial pressures of less than 10^{-3} Torr. Within these temperature and pressure ranges, for CTR candidate materials, little experimental data are available from which tritium permeation rates can be calculated.

An initial study⁸⁴ was undertaken to determine the low pressure solubility of hydrogen isotopes in niobium and selected niobium alloys, and to determine the diffusivity of hydrogen isotopes in these metals. Based on the results of this study and on what is apparently needed in CTR technology, it was decided that valid data could best be obtained by considering candidate CTR metal -- tritium interactions. The objective, therefore, is to build a suitable tritium gas handling system where low pressure--high temperature tritium solubility and diffusivity can be determined.

Hydrogen Isotope Effects: System

A vacuum system has been built so that the solubility and diffusivity of hydrogen isotopes, principally tritium, in candidate CTR metals can be determined at metal temperatures to 1000°C and gas overpressures from 100 Torr down to 10^{-3} Torr.

The vacuum system consists of a hydrogen isotope source, a furnace section, a gas transfer section and a gas measuring section. A uranium hydrogen isotope bed is used as the gas source. When tritium is to be used in the system, a total inventory of 150 curies tritium (or about 58 cc tritium STP) is available for uranium tritide. The hydrogen isotope gettering--off-gas character of uranium acts as either source or storage for the gas. The furnace section is a resistance heated tube fashioned so that a metal test sample can be maintained at a constant temperature. A mercury diffusion pump is used for gas transfer, both from the furnace section to the gas measuring section and from the gas measuring section to the uranium bed. The gas measuring section is essentially a set of constant volumes. The amount of gas taken by sample (for a hydrogen isotope solubility measurement) or released from a sample (for a permeation or diffusivity measurement) is obtained from gas-volume relationships.

Hydrogen Isotope Effects: Metals

Niobium, molybdenum and vanadium are the prominent construction metals mentioned in eTR vacuum wall design. During a 20-year lifetime of a eTR metal wall, the metallurgy of niobium, for example, would change by transmuted part of the niobium to yttrium and zirconium. Three metals have been selected for initial study, niobium, niobium-1 wt percent zirconium and vanadium. All three metals to be used in this study are greater than 75 percent reduced from cast condition in the form of a 1/2 in. diameter annealed cylindrical rod. The chemical composition of each alloy is shown in Table 1.1.

TABLE 1.1

METAL COMPOSITION-INGOT ANALYSIS

	<u>Ni obi unl</u>	<u>Vanadium</u>	<u>Niobium-l Zirconium</u>
B	< 1 ppm	< 1 ppm	< 1 ppm
C	50	110	50
Cd	< 5		< 5
Co	<10	<10	<10
Fe	<50	59	<50
H	< 5	2	5
Hf	<50		<50
Mn	<20	<20	<20
Mo	<20	<20	20
N	20	10	50
Ni	<20	<20	<20
o	50	<50	50
Pb	<20	<20	<20
Si	<50	400	<50
Ta	1070		670
V	<20		<20
W	160		76
Zr	100		
Nb	BAL. WT %		BAL. WT %
Zr			
V		BAL. WT %	

Hydrogen Isotope Effects: Proposed Experiment

The isothermal-pressure-composition equilibria are to be determined for each alloy using first hydrogen and then tritium. The temperatures of interest are 800, 900 and 1000°C, and the pressures of interest are between 100 Torr and 10^{-3} Torr. These data will help to characterize the test metals and to provide a comparison to data appearing in the open literature. The bulk of the isotope effects study is concerned with determining the steady state permeability of hydrogen isotopes through the test metals.

The permeation specimen consists of a closed end tube (thimble) of the test metal welded to a molybdenum tube. On one side of the thimble, the hydrogen isotope pressure will be kept at some constant value. On the other side of the thimble, the hydrogen isotope pressure will be maintained near zero by transferring accumulated gas to the standard volumes of the vacuum system. The rate of pressure increase in the standard volumes defines the permeation rate through the sample.

Hydrogen Isotope Effects: Preliminary Results

Due to test material procurement delays only the vacuum system has been built and a limited number of isothermal pressure-composition determinations have been made. Figure 1.1 shows hydrogen solubility curves for niobium. These results are in reasonable agreement with data taken from the literature.

Solubility determinations have also been made on the niobium-zirconium alloy at 1000°C. Although not shown on Figure 1.1, the data points parallel and lie between the 1000°C and 900°C lines.

Figure 1,1
Logarithmic Plot of Isotherms in the
Niobium-Hydrogen System

J. SPUTTER-DEPOSITION OF HIGH FIELD SUPERCONDUCTORS
FOR LARGE CTR MAGNETS - S. D. Dahlgren

Current research on Nb_3Al (Niobium-Aluminum) and $\text{Nb}_3(\text{Al-Ge})$ (Niobium-Aluminum-Germanium) superconductors shows high rate sputter-deposition can produce better superconducting materials for high field CTR magnets than other methods. Superconductors for fields up to about 150 kilogauss presently are fairly satisfactory. Materials for higher field applications are at present not satisfactory. Fields higher than 150 kilogauss may be needed for economical CTR operation because the usable output power of a reactor is said to increase with the fourth power of the magnetic field. Fortunately, practical amounts of sputter-deposited superconductor materials can be made by scaling up Pacific Northwest Laboratory's unique high-rate sputtering capability. Deposition source design, deposition parameters, and heat-treatment procedures (all at or below 750°C) were established for laboratory quantities of superconductor materials under an AEC-Division of Physics Research Contract.

Sputter-deposited superconductors have all the characteristics needed for good electrical properties. They have fine grain size and uniform composition and structure throughout which yields measured current carrying capacities of $2.6 \times 10^5 \text{ amps/cm}^2$ in a magnetic field of 100 kilogauss. Superconducting transition temperatures are high (18°K) and excellent reproducibility of superconducting properties has been demonstrated. Materials having the highest known critical fields can be made by sputter-deposition, and there is no apparent limitation on coating thickness. Furthermore, superconductors can be continuously sputter-deposited in strip form directly onto a stabilizing shunt material. The deposit bond to the shunt material substrate has been found to be excellent.

High field superconductors cannot be mechanically formed because they are extremely brittle intermetallic compounds. Consequently, investigators developed 1) arc melting, 2) diffusion, and 3) vapor deposition fabrication methods. Each of these methods has shortcomings that can be overcome by sputtering. For example, problems created by high reaction temperatures (methods 1 and 3), large differences in elemental vapor pressure (methods 1 and 3), uncontrollable or non-uniform composition (segregation or decomposition) (methods 1, 2, and 3), and lack of grain size control (methods 1, 2, and 3) are either not encountered or are amenable to control with sputter-deposition. Researchers have developed ingenious methods of alloying for controlling grain size with diffusion methods, but these often introduce deleterious side effects. Even with alloying, only a very thin layer of superconductor can be formed for the highest field superconducting ribbon available; additional reaction time increases the superconducting compound thickness but grain size control is lost and conductor current carrying capacity is only slightly increased.

Sputter-deposition is also very attractive for research work in searching for new and better superconducting compounds. Samples can be made rapidly from the elements of most solids, and compound compositions are not restricted by phase diagram and thermodynamic considerations. Thus, research quantities of potentially attractive compounds, impossible to make by other methods, can be prepared easily by sputter-deposition.

K. CTR VACUUM WALL SPUTTERING AND WALL LIFE CRITERIA - N. Laegried
and S. D. Dahlgren

Physical sputtering may be the most important among all the processes affecting CTR wall life and plasma conditions. But even physical sputtering may not cause severe wall erosion. General equations are presented that are easily used for estimating erosion rates and wall life. Specifically these have been applied to wall materials proposed for conceptual pulsed and steady-state (mirror and toroidal) experimental reactors. Sputtering factors applicable to CTR vacuum wall erosion are also discussed.

Sputtering Erosion Rates

In order to gain perspective on the wall erosion problem due to physical sputtering some general sputtering rate calculations are reviewed. The results of the calculations can be used to determine limiting flux-sputtering yield products ($S_{\mu}\phi_{\mu}$) and thus to evaluate reactor design practicability.

The number of atoms (N_s) sputtered from a surface is:

$$N_s = (\sum_{\mu} S_{\mu}\phi_{\mu})\Lambda t \quad (1)$$

where: S_{μ} = sputtering yield in atoms/particle for a bombardment species (μ)

ϕ_{μ} = particle flux density of bombardment species (μ)

Λ = area being bombarded

t = duration of bombardment.

The term $\sum_{\mu} S_{\mu}\phi_{\mu} = S_1\phi_1 + S_2\phi_2 + S_3\phi_3 + \dots$ includes all bombardment species, i.e. 1) neutrons, 2) protons, 3) deuterons, 4) tritons, 5) alpha particles, etc. Yield (S_{μ}) is dependent on particle energy, therefore an average yield must be ascertained for each bombardment species because of the spectrum of particle energies for each species.

11/so:

$$N_s = d(t)A \left(\frac{\rho N_0}{A_w} \right) \quad (2)$$

where: $d(t)$ = thickness removed in time t

A = area being bombarded

ρ = wall material density

A_w = atomic weight for the wall material

N_0 = 6.023×10^{23} atoms/mole.

The product $d(t) \cdot A$ is the volume sputtered and the term in parentheses in Eq. 2 gives the number of wall atoms per unit volume. Eliminating N_s from equations 1 and 2 and solving for the thickness of material removed in time (t) yields:

$$d(t) = \frac{(\sum_{\mu} S_{\mu} \phi_{\mu}) t}{N_0} \frac{A_w}{\rho} \quad (3)$$

Therefore, the rate of material removal per year (3.15×10^7 seconds) is:

$$R = d(1 \text{ yr}) = 5.23 \times 10^{-17} (\sum_{\mu} S_{\mu} \phi_{\mu}) \left(\frac{A_w}{\rho} \right) \text{cm/year.} \quad (4)$$

If a certain percent reduction in wall thickness is not tolerable over a practicable reactor lifetime, the sputtering erosion rate, R = thickness/lifetime, must not exceed a fixed value. Design lifetime will not be exceeded if

than limiting for the other materials. For zirconium to have a wall life of 20 years or more when 0.2 cm wall thickness reduction is allowable (20% reduction in a 1 cm thick wall), Equation 5 shows that $\sum_{\mu} S_{\mu} \phi_{\mu} < 1.4 \times 10^{13}$ wall atoms/cm²/sec. Consequently, sputtering should not be a problem for suggested wall materials when $\sum_{\mu} S_{\mu} \phi_{\mu} \approx 10^{13}$, and this number should be a handy one to keep in mind when considering reactor design and operation for 1 cm thick walls. A table of values for $\sum_{\mu} S_{\mu} \phi_{\mu}$ (Table K.1) is included for 0.2 cm wall erosion and 2, 4, 10, and 20-year lifetimes of candidate wall materials. Other $\sum_{\mu} S_{\mu} \phi_{\mu}$ numbers can be derived from Equation 5 for other materials, potential lifetimes, and thickness reductions. It is inconceivable that a reactor vacuum wall would ever be less than 1 mm thick, and a 20% tolerable erosion of a 1 mm thick wall would correspond to a maximum $\sum_{\mu} S_{\mu} \phi_{\mu} \approx 10^{12}$. Thus sputtering erosion problems should not be encountered for any reactor with $\sum_{\mu} S_{\mu} \phi_{\mu} < 10^{12}$.

Solving Equation 3 for t , the time to erode a thickness $d(t)$ from the wall, and changing the constant to give erosion time in years (t') rather than seconds (t) yields:

$$t' = (1.91 \times 10^{15}) \frac{d(t')}{\sum_{\mu} S_{\mu} \phi_{\mu}} \frac{\rho}{A_w} \quad (6)$$

Thus the lifetime of a reactor wall can be calculated if all S_{μ} and ϕ_{μ} are known and allowable thickness reduction is predicted.

Niobium Sputtering

Inserting A_w and ρ for Nb in Equation 4 and changing the constant to obtain R_{Nb} in mm/year rather than in cm/year yields

$$R_{Nb} = 5.65 \times 10^{-15} \left(\sum_{\mu} S_{\mu} \phi_{\mu} \right) \text{ mm/year} \quad (7)$$

TABLE K.I

Values of $\sum_{\mu} S_{\mu} \phi_{\mu}$ calculated from Equation 5 giving maximum allowable flux-sputtering yield products for several candidate wall materials and life-times of interest. Tolerable sputtering erosion is assumed to be 0.2 em.

Wall Material	$\frac{A_w \text{ cm}^3}{\text{p mole}}$	$\sum_{\mu} S_{\mu} \phi_{\mu} (x 10^{13})$ wall atoms/cm ² /sec for wall removal of 0.2 em (20% of 1 em)			
		20-year lifetime	10-year lifetime	4-year lifetime	2-year lifetime
V	8.6	2.3	4.6	11.5	23
Mo	9.2	2.2	4.4	11.0	22
Nb	10.9	1.8	3.6	9.0	18
Ta	10.9	1.8	3.6	9.0	18
Zr	14.2	1.4	2.8	7.0	14

for the material removal rate from a Nb wall. Inspection of Equation 7 shows that R_{Nb} is given by the sum of the rates for the individual species (μ); i.e. $R_{\text{Nb}} = R_1 + R_2 + R_3 + \dots + R_{\mu} + \dots$ where $R_{\mu} = 5.65 \times 10^{-15} S_{\mu} \phi_{\mu}$. Plotting R_{μ} as a function of S_{μ} for several values of ϕ_{μ} (Fig. 1) allows the rapid estimation of sputtering rates for individual species (μ) if values of S_{μ} and the magnitudes of ϕ_{μ} are available. The total erosion rate is obtained by summing the individual rates.

Rates indicated by the horizontal dashed lines in Fig. K.1 correspond to 0.2 em sputtering erosion (20% of 1 em wall thickness) during the time indicated by the number of years to the right of the line. Combinations of S_{μ} and ϕ_{μ} below the horizontal lines give rates which remove less than 0.2 em in the designated time. Thus if S_{μ} is known, tolerable fluxes (ϕ_{μ}) for the

WALL SPUTTERING EROSION RATE FOR Nb VERSUS SPUTTERING YIELD (S_{μ}) CALCULATED FROM EQUATION 7 FOR SEVERAL ASSUMED VALUES OF PARTICLE FLUX (Φ_{μ}). REGIONS BELOW THE HORIZONTAL DASHED LINES CORRESPOND TO LIFETIMES IN EXCESS OF THE NUMBER OF YEARS SHOWN AT THE RIGHT. LIFETIMES WERE BASED ON 0.2 cm WALL LOSS (20% OF 1 cm).

FIGURE K.1

required lifetime can be determined readily for the case where one species (μ) dominates, or the sum of the rates can be adjusted for tolerable flux levels when more than one species contributes significantly to sputtering erosion. Stated another way, the condition for less than 0.2 cm Nb wall thickness reduction in 20 years as shown in Fig. K.1 and given by Equation 5 is

$$\sum_{\mu} S_{\mu} \phi_{\mu} < 1.8 \times 10^{13}. \quad (8)$$

Sputtering in Conceptual Reactors

Particle fluxes and yields are sufficiently high that sputtering erosion could be a problem in proposed pulsed reactors. Steady state fusion reactors would appear at this time to have fewer problems unless average bombarding particle energies are much lower than anticipated.

The θ pinch prototype pulsed reactor suggested by Burnett and Ellis(1) of Los Alamos has a plasma density of 10^{16} particles/cm³. The suggested configuration and pulse rate (1 sec⁻¹) results in an average wall flux of 1.75×10^{16} particle/cm²/sec. From Equation 5, $\sum_{\mu} S_{\mu} \phi_{\mu} < 1.8 \times 10^{13}$ for Nb so that a yield $S_{\mu} \leq 10^{-3}$ atoms/particle is required to obtain a 20-year wall life (20% reduction of a 1 cm thick Nb wall). Measured yields for high energy particles on Nb(2) indicate an average yield of $S_{\mu} \approx 4 \times 10^{-3}$ atoms/particle, which is too high by a factor of four. Some reactor designs incorporate a thin electrically insulating coating on the reactor first wall. Sputtering yields for such insulating coatings would have to be even lower than for Nb because less thickness reduction would be tolerable for a thin coating. With or without a coating, redeposition of sputtered wall material may

lower the wall erosion to an acceptable amount even with the particle fluxes and yields suggested for the 0 pinch pulsed reactor.

In the proposed steady state machines (mirror and toroidal), the maximum suggested flux at the first wall is approximately 3×10^{14} particles/cm²/sec and the average particle energy is expected to be greater than 10 keV.(3,4) Again using the criterion of Equation 5, the maximum tolerable yield would be 6×10^{-2} atoms/particle. Such a high yield would not be expected unless the average particle energy is very low (e.g. 1 to 5 keV), which is not anticipated at present. The small amount of data available for sputtering yields of light particles having 1 to 5 keV energy indicate higher yields for 1 to 5 keV particles than for particles with energies greater than 5 keV.

Sputtering by helium and neutrons is not expected to be significant in any of the reactors because reported helium and neutron fluxes(S) are low $\ll 2 \times 10^{15}$ particles/cm²/sec). Yields are also low due to high (MeV) particle energies. Helium and neutron yield values high enough to cause concern, however, have been suggested by Kaminsky.⁽⁶⁾

Additional Factors Influencing Sputtering

Sputtering yields of the wall material can be reduced by surface roughness and alloying additions. Thus correspondingly higher particle flux densities would be allowed according to Equation 5. Wall erosion rates may also be significantly reduced by sputtered material redeposition.

Surface roughness can be increased by sputter etching (particularly where certain impurities are present), blistering, and uneven redeposition.

Increased surface roughness also can be produced intentionally for yield reduction. The effect of certain impurities on surface roughness is markedly demonstrated in Figure K.2. One millimeter high cones were formed on a silver target due to elevated target temperatures and the close proximity of a simultaneously sputtered copper target. Copper coatings protected the cone tips from sputter erosion and promoted surface roughness by cone formation. Impurities distributed in sputtering targets, even at the atomic level, have been shown to cause low sputtering yields due to cone formation.⁽⁷⁾ Elevated CTR vacuum wall temperatures, which encourage the formation of cones and surface roughness by impurities, are necessary for efficient CTR operation.

Yields can be lowered by suitable alloying additions even when surface roughness does not occur. Dahlgren and McClanahan found yields were reduced to half with nominal additions of low sputtering yield second phases.⁽⁸⁾ Typical results are shown in Figure K.3. Low target temperatures eliminated cone formation.

Depending on the operating conditions, it should be expected that most of the sputtered material will be redeposited on the walls. In that case the allowable $\sum_{\mu} S_{\mu} \phi_{\mu}$ could be significantly increased unless localized depressions develop where sputtering rates are much greater than redeposition rates. Peeling of the redeposited material must be prevented. Investigators experienced in sputtering, however, have developed surface conditioning procedures for particular material systems to avoid peeling. Similarly, techniques have been developed to prevent coating of exposed components •

Ag-Cu BOUNDARY

LARGE, 1mm HIGH CONES ON A POLYCRYSTALLINE Ag TARGET CO-SPUTTERED WITH Cu. SOURCE OF Cu ATOMS IS AT THE Ag-Cu BOUNDARY INDICATED. THE SPACING OF THE Ag CONES INCREASED WITH DECREASED COPPER FLUX AT INCREASED DISTANCE FROM THE BOUNDARY. (SEM 75X)

FIGURE K.2

SPUTTERING YIELDS VERSUS COMPOSITION FOR TWO-PHASE TARGETS SHOWING RAPID DECREASE IN SPUTTERING YIELD WITH SMALL ADDITIONS OF Ni OR Co TO Ag, TARGETS WERE SPUTTERED WITH 1500 eV Kr IONS.

FIGURE K.3

Acknowledgments

The authors wish to thank Dr. R. D. Nelson for reviewing the manuscript.

L. IRRADIATION INDUCED SWELLING IN CTR MATERIALS - J. L. Brimhall

Two recent publications have reviewed the materials problems associated with CTR operation. (93,94) It was concluded that severe radiation damage to the first wall of the CTR, particularly in the form of voids, was a critical problem area. Void formation in materials has also been reviewed in two recent publications. (95,96) Several investigations of neutron irradiated refractory metals (Mo, Nb, V) have shown that voids easily form in these materials during neutron irradiation. (97,98,99) It has also been shown that high energy, heavy ion bombardment can simulate in a short time, the high displacement damage expected after long time CTR operation. (100,101,102)

In a previous report, the microstructural damage induced by high energy ion bombardment had been studied in a cursory manner in several refractory metals. (103) In this report, a more detailed study of the void swelling phenomenon in ion bombarded niobium and Nb-1% Zr alloy is presented.

Experimental Procedures

Material from several different stocks of nominally pure niobium were used. The chemical composition of the pure niobium and Nb-1% Zr is given in Table L.1.

The specimens were prepared in the form of foils 0.05 to 0.2 mm thick. All material was annealed at 1600°C for approximately 30 minutes in a vacuum of 1×10^{-7} Torr. After annealing, the specimen surfaces were electropolished prior to ion bombardment.

TABLE L.1

CHEMICAL ANALYSIS OF MATERIAL USED IN STUDY

	Commercially pure Niobium	High purity Niobium	Nb-1% Zr
Ta	500 - 700	100	500
W	30	10	70
Other metallics	20 - 40	1	20 - 50
°2	50 - 100	24	60 - 100
N ₂	40	4	35
H ₂	4	0.2	5
C	30	10	40

The specimens were irradiated in the tandem accelerator at Ion Physics Corporation using 7.5 MeV Ta⁺⁺⁺ ions. The bombardment temperatures were 800 and 900°C and the vacuum varied between 4×10^{-7} to 4×10^{-6} Torr. The ion currents used produced atom displacement rates of 0.01 to 0.02 dpa/sec (dpa = atom displacements per atom).

Samples were prepared for electron microscope examination by selective electrothinning in a manner similar to that described previously.(104) According to calculations, the displacement damage peaks at about 8000 Å from the surface. Generally 2000 - 4000 Å was removed from the surface. All quantitative measurements were made on transmission electron micrographs. Estimation of the foil thickness was made using stereo microscopy whenever possible. The spread in the calculated void concentration was still about $\pm 30\%$. Due to uncertainties in measuring void size and magnification, the accuracy of the calculated void swelling values approaches $\pm 50\%$.

RESULTS

Microstructure of Niobium and Nb-1% Zr

Voids were observed in all specimens of pure niobium bombarded at dose levels from 1 to 300 dpa. The microstructure of niobium bombarded at 800°C is shown in Figure L.1. At low doses, the voids are randomly distributed and have a cubic shape with the faces parallel to {100}, Figure L.1a. With increasing dose, a void lattice begins to emerge. The ordering of the voids is first noticed after 60 dpa and at 90 dpa is well developed, Figure L.1b. At a higher dose of 300 dpa, the only change is an increasing perfection of the lattice, Figure L.1c,d. The void lattice is body centered

(a) 15 dpa

(b) 90 dpa

(c) 300 dpa

Cd) 300 dna

Figure L.1. Microstructure of niobium bombarded with 7.5 MeV Ta+++ ions at 800°C to the dose levels indicated.

cubic which is superimposed on the parent atomic bcc lattice. The lattice spacing is 380 Å average. The void morphology at the high doses is still cubic. Similar observations were made at 900°C, Figure L.2. The voids are larger after 900°C bombardment than at 800°C. The lattice spacing is 750 Å or about twice that measured at 800°C. The voids are strongly faceted on the {110} planes at the higher doses, Figure L.2c. This faceting appears to be a function of void size. The larger the void, the greater the faceting.

The general appearance of the void microstructure in the Nb-1% Zr alloy was similar to that of pure niobium at least at 900°C. There were some anomalies. Figure L.3a shows typical microstructure at a low dose of 20 dpa. Figure L.3b shows an anomalous high concentration of voids along a grain boundary. In this particular region, very few voids were observed away from the grain boundary. At a higher dose of 50 dpa, the void distribution was very non-homogeneous. The size and density varied noticeably over the bombarded region as illustrated in Figure L.3c and L.3d. No voids were detected at the highest dose of 100 dpa. However, it is possible that too much had been removed from the surface on this sample and observations were made on material beyond the damage zone.

No voids were detected in Nb-1% Zr samples bombarded at 800°C. Damage was evident in the form of dislocation loops and dislocation tangles. The samples showed evidence of heavy oxidation which may have affected the bombardment process. Some experiments will be repeated to determine if this lack of voids at 800°C is a true phenomenon.

(a) 10 dpa

(b) 60 dpa

(c) 280 dpa

(d) 280 dpa

Figure L.2. Microstructure of niobium bombarded with 7.5 MeV Ta⁺⁺⁺ ions at 900°C to the dose levels indicated.

(a) 20 dpa

(b) 20 dpa

(c) 50 dpa

(d) 50 dpa

Figure L.3. Microstructure of Nb-1% Zr bombarded with 7.5 MeV Ta+++ at 900°C to the dose levels indicated.

Void Parameters as a Function of Dose

The void size increases with dose at both 800 °C and 900 °C. At 800 °C, the size increases from 35 Å to 108 Å at 800 °C and from 40 Å to 250 Å at 900 °C in pure niobium over the dose range studied. There is little size increase after a dose of 100 dpa at 800 °C. The size of the voids is smaller in Nb-1%Zr alloy than in pure niobium for equivalent doses and temperatures. At a dose of 20 dpa, the average void size is 74 Å in Nb-1%Zr compared to 135 Å in pure niobium. The results are in contrast to those of Wiffen, who found larger voids in irradiated Nb-Zr alloys than in pure niobium. (97)

The void concentration increases with dose at 800 °C, but decreases with dose at 900 °C in pure niobium. At both temperatures, the concentration levels off to a constant value after about 100 dpa. The void concentration of the Nb-1%Zr alloy varied considerably within the observed areas. An average value would place it in the same range as that for pure niobium for an equivalent dose.

The calculated void swelling is plotted as a function of dose for pure niobium in Figures L.4 and L.5. At both 800 °C and 900 °C, the swelling increases approximately linearly with increase in dose from to 100 dpa. There is a definite leveling off of the swelling in doses exceeding 100 dpa which indicates saturation. This is quite evident at 800 °C where the saturated value appears to fall between 4 and 5%. The swelling values are greater at 900 °C for all dose levels. It also appears that saturation is occurring at 900 °C at doses greater than 100 dpa and in conjunction with the development of the void lattice. The saturated value appears to fall within the range 7 - 10%

Figure L.4

VOID SWELLING IN NIOBIUM BOMBARDED WITH 7.5MeV Ta IONS AT 900°C

Figure L.5

It must be remembered that the very high dose rates used in the ion bombardment experiments produce an effective temperature shift whenever correlating the results with those from neutron irradiation. (101, 105) For the dose rates reported here, the effective neutron irradiation temperatures are 630 °C and 700 °C for the ion bombardment temperatures of 800 °C and 900 °C, respectively.

DISCUSSION

Effect of Temperature and Dose on Swelling

The void size is larger and the concentration lower at 900 °C than at 800 °C. This is in agreement with the general observation of the void size increasing and the concentration decreasing with increase in temperature. The magnitude of the swelling was always greater at 900 °C than at 800 °C. Previous results have shown a maximum in the swelling versus temperature curve at 900°C for a low dose of 5 dpa. (103) The maximum does not shift toward lower temperatures at high doses, however, it is not possible to say whether there may be a shift to higher temperatures with increasing dose.

The dependence of swelling on dose is similar at the two temperatures. However, the atomic mechanisms are not the same. At 800 °C, nucleation of voids is still occurring up to a relatively high dose of 100 dpa. At 900 °C, nucleation has stopped at a relatively low dose as shown by the fact that the void concentration is decreasing with dose. The observation of both an increasing and decreasing void concentration at two different temperatures is not unreasonable. Similar behavior has been observed in ion bombarded nickel and stainless steel. (106,107) The reasoning is as follows. All voids have a sphere of influence around them within which

vacancies will go to the void rather than nucleate a new void. When these spheres of influence begin to overlap, no further voids will be nucleated and a maximum in the void concentration will have been reached. The vacancy migration rate increases with temperature so the sphere of influence will increase with temperature. Therefore, the value of the void concentration at its maximum will be less at higher temperatures and consequently will be attained at a lower fluence. At 900 °C, the lowest dose was apparently at or past the point of maximum void concentration. Further irradiation only produces coarsening of the existing voids. In this case the void swelling is solely due to the growth of the voids.

Saturation in the swelling does occur and this saturation is associated with the formation of the void lattice. It does not matter whether the void concentration is increasing or decreasing or how the void size is changing. The spacing of the void lattice is, however, a function of temperature. The ratio of the lattice spacing to the void size is 6.9 at 800°C and 6.0 at 900°C. These ratios are in good agreement with the approximate calculations of the theory of Bullough and Malen.(108) However, a more rigorous calculation by Stoneham has shown that the ratio should be between 2 - 4.(109) Further, these theories, which depend on lattice elastic anisotropy, can not explain the presence of the void lattice in tantalum.(110) A recent model by Tewary does not depend on elastic anisotropy but only on void faceting.(111) Void faceting must be important in the development of the void lattice. At 800°C, the voids are near perfect cubes and the lattice is very well defined. At 900°C, the increased truncation on other planes produces a void shape closer to a sphere and the void lattice is

consequently much less perfect. It may well be that at a higher temperature, the lattice would not be observable at all. Brown has developed a model of void stability based on the attainment of steady state conditions.(112) The model predicts an equilibrium void spacing and the volume fraction of voids but does not necessarily predict a lattice structure.

Influence of Zirconium on Void Swelling

Some of the results from the Nb-1% Zr alloy were anomalous, and it is difficult to arrive at any firm conclusions. Basically, there was less swelling at 900°C than in pure niobium and no voids at all, hence no swelling, at 800°C. Work in other materials has generally shown that alloying tends to inhibit void formation,(97,113,114) although the precise mechanisms are many and varied. In this regard, the results on Nb-1% Zr are consistent. The results are at some variance to those of Wiffen who found larger voids in neutron irradiated Nb-1% Zr than in pure niobium.(97) The void swelling was approximately the same in the two materials. The concentration of voids varied considerably in the neutron irradiated Nb-1% Zr even within one grain. This is in agreement with ion bombardment results which showed a wide variation in both size and concentration of voids.

It is suspected that oxygen is playing a very significant role in the development of the void structure. Oxygen would interact with the vacancies and interstitials thereby affecting the recombination rate and nucleation of voids. Niobium dissolves considerable oxygen, but zirconium is a good oxygen getter. Therefore, the relationship between the zirconium

atoms, oxygen atoms, vacancies, interstitials, and larger defect aggregates can become quite complex. The uneven distribution of the oxygen and zirconium atoms may account for the large variations in void concentrations and the strings of voids observed along grain boundaries.

Significance of Results to CTR Operation

Niobium and Nb-1% Zr are prime candidates for the vacuum or first wall in a CTR. The results presented here give a preview of the microstructural changes to be expected after long time exposure in the CTR. For example, a dose of 300 dpa would be produced in 5 to 10 years of CTR operation.

Two important questions which need to be answered in these types of studies are: does swelling continue to increase indefinitely, and if not, what is the maximum swelling? The results show that swelling does saturate at high exposure. At a bombardment temperature of 900°C, the ultimate swelling will fall between 7 and 10%. At 800°C, the maximum swelling expected is approximately 5%. The magnitude of the swelling has been found to be relatively insensitive to dose rate at least in the temperature region of maximum swelling. This is not necessarily true for void size and concentration. However, it is felt that these swelling values can be applied to the radiation doses expected during CTR operation.

The temperature shift must be taken into account when relating the values to CTR temperatures. A temperature range of 800 - 900°C would correspond to approximately 630 - 700°C in a reactor as mentioned previously. Wiffen has shown a maximum in the swelling in neutron irradiated niobium in

the temperature range 600 - 630°C.(97) The maximum swelling occurs at a slightly higher temperature in ion bombardment. However, some of the diffusion data used in calculating a temperature shift are not too well known, so close correlations of temperature cannot be expected.

There are still important unanswered questions, It is very likely the void lattice does not form at higher temperatures so predictions of saturation at higher temperatures cannot realistically be made. No hint of the void lattice was observed in the Nb-1% Zr alloy, so predictions about saturation in the alloy may have to be modified. However, the results do show the swelling problems which will be encountered, at least within a certain temperature range, after long time reactor operations.

M. Measurement of 14 MeV Neutrons Slowing Down in Graphite -
R. J. Hall, J. B. Freim, R. M. Jones

Experiments which were reported(25) last year were continued to measure selected foil reaction rates from 14 MeV neutrons incident on room temperature graphite. The purpose of the latest experiment was to obtain further benchmark data to assess the accuracy of calculational methods for eTR and to obtain additional data to resolve the relatively poor correlation with calculations obtained in the initial studies. Experiments were conducted in the High Temperature Lattice Test Reactor. (25) In addition to obtaining benchmark data, the experiments were designed to be similar to the previous one so that the effects of the experimental environment on the measurements could be assessed. Thus, while the basic geometry remained relatively similar to the previous experiment, the measurements were made in a larger system consisting of a 10 ft x 10 ft x 6 ft (minimum thickness) graphite assembly located in the central volume of the HTLTR. Two experiments were performed to better assess the calculational uncertainties; the first using the foils employed in the previously reported experiment; the second using additional foils.

The foils were arranged as previously in the center of the pile of graphite. The source was positioned at one end of the graphite and the foils removed from the other end. The foils that were added to those used in the earlier experiment were Oy, W, In, Fe, and depleted and enriched U in order to measure the reactions

Using an activation ratio technique in which the 1/v dysprosium (n,y) reaction is used as a reference, the resonance and thermal contributions to the activation of Au(n,y), Mn(n,y), Cu(n,y), In(n,y) and W(n,y) were

separated out. This will yield additional information on the lower energy portions of the neutron spectrum. The W, In (n,y reaction) and the ^{23}SU foils were added to further define the thermal and resonance portions of the spectrum. To better define the higher energy (~ 1 to 10 MeV) portion of the spectrum, three additional threshold foils [In^{115} (n,n' reaction), ^{238}U (n,f reaction), Fe^{56} (n,p reaction)] were added.

The irradiations and experimental measurements were completed by year end. The data is in the process of being analyzed and compared to calculated results as well as the previous experimental results. Preliminary results from the current experiments indicate slightly better correlation with calculations than did the previous experiment." The reduced neutron leakage from the larger graphite system is quite apparent in the results. The effect of the experimental environment (reflection from buss bars, thermal insulation, and radiation shielding) appears to perturb the experimental results; however, if necessary, these effects can be unfolded from the results.

The next step in the systematic plan to provide worthwhile benchmark data to assess the accuracy of calculations is to move the 14 MeV neutron source into the center of the 10 ft cube of graphite in order to provide a better source-graphite geometry. Once that is accomplished, the HTLTR and our understanding of the environment on the experimental measurements should make it possible "to go into a more extensive program of providing benchmark data. Because of some of the unique characteristics of the HTLTR it is a useful facility for neutronic and other technical studies of blankets for eTR devices. In addition to its usefulness in providing integral benchmark data at room temperatures, the test volume is thermally insulated and surrounded by a radiation shield so that neutronic characteristics of blankets can be readily studied in the presence of strong neutron sources and at increased temperatures up to 1000°C when reactions at higher temperatures become important. In addition, it could provide a facility for studying characteristics of fission blankets since it was designed originally as a critical fission reactor facility.

- N. A Calculational/Experimental Comparison of a Neutron Spectra in Graphite from a 14 MeV D-T Source - J. Freimt^t, E. Linn Draper, Jr.^{tt}, and W. C. Wolkenhauer

The neutron flux in a graphite mock-up of a fusion reactor blanket has been determined by foil activation and the iterative unfolding scheme SAND II.(117) The flux so calculated has been compared to that predicted by TWOTRAN,(118) a transport theory code using ENDF/B cross sections. TWOTRAN is a discrete ordinates code, and SAND II uses activation data to iterate on the neutron flux to minimize the difference between the calculated activity and the experimental activity.

The test blanket described in detail in reference 25 was 37-1/2" x 37-1/2" x 45" graphite stack in which foils placed at five inch intervals were neutron activated from a 14 MeV D-T source. In all ten positions the inclusion of Cu and Mn data introduced unphysical structure in the iterated spectra found by SAND II. Since neither the validity of the Cu nor Mn data could be established, both sets of data were deleted. Because of the relatively low source strength (7×10^{10} n/s) the number of foils with significant activity decreased rapidly with depth of "penetration into the block. Since SAND II accuracy is dependent on having a relatively large number of reaction rates, only the first four positions were used.

In Figure 1 the TWOTRAN flux at position one is compared to the SAND II iterated flux using the TWOTRAN flux as an initial guess. Although the general shape of the two curves is the same, a discrepancy exists from one to fourteen MeV where the SAND II predicts a lower flux. Although the number of foils sensitive in the thermal range is small, SAND II shows a higher thermal peak at the first three positions and a higher fusion peak at positions two and three. The results corroborate previous observations that the TWOTRAN code underpredicts the penetration of fast neutrons into the graphite.(25, 119) The activities and peak values are given in Table 1 for each position in the graphite.

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TABLE I

Position

				2	3
AL27A	TWOTRAN SAND II	-72.4 -15.1	-41.0 0.04	-35.0 - 4.2	-42.5 -18.3
AL27P	TWOTRAN SAND II	-74.9 -22.4	-32.4 10.6		-28.0 1.7
AU197G	TWOTRAN SAND II	155.3 - . 0.95	81 . 1 - 0.46	18.8 - 0.7	-17.3 - 1.6
MG24P	TWOTRAN SAND II	-65.0 7.7	-41 . 1 - 0.56	-34.0 - 1.6	-29.3 2.6
NI582	TWOTRAN SAND II	-57.3 21.0	-15.6 11 . 5	100.9 23.9	207.9 36.3
NI58P	TWOTRAN SAND II	-67.3 2.8	-36.9 - 2.0	-28.4 0.17	
TI48P	TWOTRAN SAND II	-64.7 6.9	-51 . 1 -19. 1	-38.1 -17.6	-35.7 -20.6
Average	TWOTRAN SAND II	144.7 14.6	79.4 10. 1	59.2 13.4	87.7 20.4
Fusion Peak	TWOTRAN SAND II	4.75+8 1.59+8'	7.09+6 5.34+6	7.27+5 1.35+6	1.65+5 4.47+5
Thermal Peak	TWOTRAN SAND II	1.30+13 3.44+13	6.84+13 1.24+14	1.47+14 1.76+14	2. 15+14 1.81+14

FIGURE N.1 FLUX COMPARISON

IV. COMPUTER CODE ABSTRACT

FUSION FIREBALL CRITICALITY ANALYSIS
(BATTELLE-NORTHWEST PROGRAM FUSION)

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MACHINE-ANALYSIS ABSTRACT

Battelle-Northwest Program Fusion analyzes spherically symmetric fusion ignition by multiple-zap multiple-beam laser implosions. Analysis versatility includes coupled variation of design specifications to meet performance requirements.

PHYSICS

At criticality,. the fusion fireball is represented as a fully ionized neutral plasma within an expanding spherical detonation shock wave driven by fusion energy, with fusion energy release rate just balanced against shock feed energy rate, neutron energy leakage rate, and photon energy leakage rate.

Temperature-variant fusion cross sections are assembled by thermally averaging quantum theory analytic forms fit directly to cross section measurements. Shock dynamics are based upon the shock-frame sonic-exit detonation solution of the mass, momentum, and energy conservation equations. Photon leakage is based upon analytic integration of the photon transport equation for bremsstrahlung emission attenuated by both isotropic scatter and absorption.

MACHINE TIME

Thermally averaging a ten-temperature set of fusion cross sections for the six most important fusion reactions requires about three minutes on the Univac-11GB computer. Criticality search on fireball radius, using fusion cross sections generated in a preceding run, requires about thirty seconds to reach four-figure convergence.

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