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PACIFIC NORTHWEST LABORATORY ANNUAL REPORT FOR 1974 ON CONTROLLED THERMONUCLEAR REACTOR TECHNOLOGY

ERDA RESEARCH AND DEVELOPMENT REPORT

Battelle
Pacific Northwest Laboratories
Richland, Washington 99352

FEBRUARY 1975

Prepared for the U.S. Energy Research
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ERDA RESEARCH AND DEVELOPMENT REPORT

By

Staff of Battelle Northwest

February 1975

On January 19, 1975, research and development programs of the U.S. Atomic Energy Commission (AEC) became part of the newly formed Energy Research and Development Administration (ERDA). In this report, since it refers to work done in 1974, most references are to AEC programs.

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PREFACE

The Annual Report for 1974 on Controlled Thermonuclear Reactor Technology consists of progress summaries of research conducted by the staff of Pacific Northwest Laboratories (PNL). The ERDA Division of Controlled Thermonuclear Research is a major sponsor of the work. However, fusion-related work sponsored by others is also included as appropriate.

The summaries are presented in three major categories of:

- System Design and Analysis
- Materials Research and Radiation Environment Simulation
- Environmental Effects

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

Loren C. Schmid
Fusion Programs Manager

Other Reports in the Series:
Annual Controlled Thermonuclear Reactor Technology Report-1971, BNWL-1604.
Annual Report on Controlled Thermonuclear Reactor Technology-1972, BNWL-1685.
Annual Report for 1973 on Controlled Thermonuclear Reactor Technology, BNWL-1823.

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ANNUAL REPORT FOR 1974 ON
CONTROLLED THERMONUCLEAR REACTOR TECHNOLOGY
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SUMMARY

Over the last 25 years controlled thermonuclear research has emphasized the containment of fusion plasma to demonstrate scientific feasibility of obtaining more energy from fusion reactions than is supplied to the plasma. Recent successes in plasma containment have caused scientists to look beyond plasma confinement problems and identify fusion reactor engineering problems which need solutions before useful power can be produced. Some fundamental and applied scientific problems could quite possibly limit thermonuclear reactor development even after scientific feasibility is attained. Evaluations by PNL over the last five years have resulted in the initiation of research and development efforts in some engineering technology areas.

The PNL staff has been studying fusion technology in areas such as economics, fusion-fission hybrid concepts, materials, neutronics, environment and safety. These studies have been scoped to make efficient use of ERDA resources, and to complement and support efforts at other laboratories.

Areas of study with other laboratories in 1974 include the design of fusion reactors, the understanding of material properties, and the environmental effects of fusion reactor power plants. In fusion reactor design, PNL staff contributed to Tokamak reactor design efforts at the University of Wisconsin in the areas of economic assessment, the compatibility of helium and lithium coolants with structural materials, and the technology of tritium handling and containment. A cooperative study of the design of a fusion-fission hybrid reactor and based upon the mirror concept of plasma confinement was carried out with the staff of Lawrence Livermore Laboratory. In addition, PNL staff contributed to studies carried out at Lawrence Livermore Laboratory on design of a Fusion Engineering Research Facility (FERF).

In materials research, PNL surface scientists worked with Lawrence Livermore Laboratory, Argonne National Laboratory, Massachusetts Institute of Technology and Atomics International, and participated in discussions at Harwell (United Kingdom) and at several laboratories in Russia. A study, Body Centered Cubic Interlaboratory Ion Bombardment Correlation Experiment, was established involving seven U.S. laboratories. Studies are planned on ion bombardment in cooperation with the Naval Research Laboratory, and on mechanical properties in a joint effort with Hanford Engineering Development Laboratory. Documentation of the status of insulator technology was addressed in cooperation with

Battelle's Columbus Laboratories, and a review of fabrication technology has been initiated with McDonnell Douglas and Battelle-Columbus.

In depth discussions have been held with scientists from several laboratories on the environmental effects of fusion power plants. These discussions included fusion reactor designs, magnetic technology, biological effects of magnetism and tritium technology.

Progress during 1974 are summarized in areas of systems design and analyses, materials research, and environmental effects. Since controlled thermonuclear reactor designs are only conceptual at this time, their characteristics are very tentative and the economics of various designs can only be studied in a very preliminary way. An envelope of reactor characteristics is being developed and analyzed to determine CTR operating characteristics necessary to achieve a satisfactory return on research investment. One concept that has some unique benefits is the fusion-fission hybrid in which a fission blanket is placed around a fusion core. Thus, studies are included to identify economic regimes for the hybrid concept.

Improvements in analytical capability are being sought to improve the validity of technical bases for CTR design and the transfer of technology to industrial capability. Present studies directed toward these goals involve the acquisition of improved nuclear data and data files, and the transfer of technology to industrial capability. These studies are being pursued as part of the hybrid reactor design program.

Structural and coolant materials applicable to CTR concepts are being studied to determine damage effects that could affect fabrication and performance of CTR components. For example, damage mechanisms which might affect the first wall, or the use of graphite in the blanket and/or shield regions, are being investigated. In addition, studies are being made on material compatibility with helium coolant and the effects of helium on the mechanical properties of vanadium and niobium. Methods for producing certain insulator coatings and barrier surfaces are in the experimental stage and tests of the resulting coatings are being planned.

Environmental assessments are being made on the basis of information available, to help identify impacts that might indicate modification of research and development studies. Technical analysis has begun where information is available. A review of available information is in progress to make recommendations for additional data where needed to help assure the protection of the public and the environment.

The contribution which fusion reactors can make to the U.S. power economy depends not only on their performance but on the performance of competing systems. An economic regimes program identifies the characteristics that a CTR must have to achieve a satisfactory return on the research investment. Two studies have been completed in a preliminary fashion. Under the energy characteristics assumed for the studies, it was found that the CTR penetration of the market is independent of LMFBR doubling over an 8-15 year range and that the 30-year present worth cost of $390 per kWe is a desirable maximum CTR cost.

Assistance was provided to the University of Wisconsin study team in defining the operating cycle, the plant factor and the electricity production cost of the UWMAK-I conceptual power plant. Estimates include operating costs, fueling costs, and certain capital costs. The results are being reported in University of Wisconsin reports for the conceptual power plant. Plans have been made for the development of similar operating and economic characteristics for the UWMAK-II conceptual power plant.

Analyses of controlled thermonuclear reactors containing a fission blanket has revealed some unique benefits. As a result, studies are being conducted to identify CTR fusion-fission hybrid economic regimes and to evaluate possible technology combinations. A regional electrical supply model with parameters in a form which can be compared to alternate power sources are being studied to determine the present worth costs and benefits which will allow substantial penetration of the energy market by the hybrid CTR. The results will provide future trends of performance characteristics for pure fusion CTR, and CTR hybrid, designs with acceptable benefit to cost ratios. Additional work is expected to include the definition of optimum fissile to electrical production ratios and to look for synergistic hybrid designs.

The fusion-fission hybrid reactor is a concept which involves elements of both fission and fusion technology. Because of the many combinations possible, an approach has been developed to evaluate the potential of various combinations of technology. The possible combinations are viewed as elements of a matrix having fusion technology as one dimension and fission technology as the other. The objective of the study is to identify the technical feasibility of electrical power production by various combinations of fission and fusion technology. The combinations can then be ranked to identify the most promising ones as candidates for later, more extensive engineering design studies.

A consistent hybrid design based on the mirror fusion reactor was developed as part of a cooperative program with the Lawrence Livermore Laboratory. Wherever possible, current technology was used for the fission and mirror fusion segments of the hybrid. Basic design objectives were to produce electrical power, produce as much tritium as consumed, and produce more fissionable material than consumed. It was concluded that it is possible to build a mirror hybrid fusion reactor which could be a viable electrical power source. In addition the probable breeding of both fissile material and tritium has been confirmed by detailed calculations of the behavior of the blanket through an extended operating period. A safety assessment has shown that the fission lattice is expected to remain subcritical and that fission fuel meltdown should not occur with loss of coolant.

Assistance was provided Lawrence Livermore Laboratory in the development of a concept for a Fusion Engineering Research Facility. Primary input was in the area of remote assembly and disassembly with other inputs in the test loops and sample handling areas.

Valid technical bases for CTR design and transfer of the technology to industry capability require the acquisition of improved nuclear data and data files, and an improved analytical capability. Data deficiencies are being defined by critical review of data used in the analysis of fusion reactor systems including hybrid reactors. Improved data and data files are obtained by evaluation and through participation in the cross-section evaluation working group of the U.S. Nuclear Data Committee.

Two technical developments were made in the process of performing mirror hybrid design analyses which enhance analytical capability for CTR design and analyses in general. Programs FUSEP and TRUTH were developed for the calculation of the energy distribution of source D-T neutrons from injected machines and the calculation of steady-state and transient thermal performance of the mirror hybrid blanket, respectively.

CTR Economic Regimes Overview

DE Deonigi, JR Young

The contribution which fusion reactors can make to the U. S. power economy will not only depend on their performance but on the performance of competing systems. The economic regimes program identifies the characteristics that a CTR must have to achieve a satisfactory return on the research investment. Table 1 illustrates the relationship between energy system characteristics and individual plant characteristics where the 30-yr present worth total cost is a unit which is common to both systems. The electrical system characteristics define the required 30-yr present worth cost total that the CTR must achieve. The plant characteristics defines this present worth cost total to determine the individual plant's performance. Two preliminary studies use the following set of energy system characteristics.

- Energy Demand is projected in Figure 1 through year 2030 with growth rates indicated for the best estimate that was used.
- The Desired Benefit Cost ratio is 10, based on recent LMFBR studies.
- The LMFBR Performance is assumed to be introduced in year 1990 with the performance characteristics described in Table 2.
- R&D Cost of $9.5 billion is assumed which translates into a present worth R&D cost investment of $2.8 billion when the time dependent pattern of expenditure is included.
- Introduction Date for the CTR is assumed to be year 2000.
- Discount Rate for R&D expenditures and benefits is 7.5% per year based on recent LMFBR studies.
- Introduction Rate for the new CTR technology is restricted to 130 GWe in the first eight years.

TABLE 1. CTR Economic Regimes

Plant Characteristics		Energy Systems Characteristics
Capital O&M Conversion Efficiency Fuel Costs Interim Capital Replacement Discount Rate Load Factor	30-yr Present Worth Cost Total	Energy Demand Desired B/C Ratio FBR Performance R&D Cost Introduction Date Discount Rate Rate of Introduction

The results of preliminary cases indicate that:

1) For the energy demand growth rate assumed, the CTR penetration of the market is independent of LMFBR doubling time over the 8- to 15-yr range indicated in Table 2.

2) The CTR participation in the electrical power economy will be 3,000 GWe by year 2030.

3) Thirty-year present worth cost total of $390/kWe is the maximum CTR cost desired.

The economic regimes program is continuing to investigate other energy system characteristics sensitivities to better define the target cost. A topical report is being prepared which will summarize the results.

TABLE 2. LMFBR Characteristics

Capital Cost, Total	$460/KW
Heat Transfer	270
Reactor System	190
O&M	$18,000,000/yr
Size	2,000 MWe
Net Thermal Efficiency	40%
Doubling Time	8 to 15 years
Fuel Cycle, Total	$320/Kg
Fabrication	150
Reprocessing	80
Shipping	40
Load Factor	80%

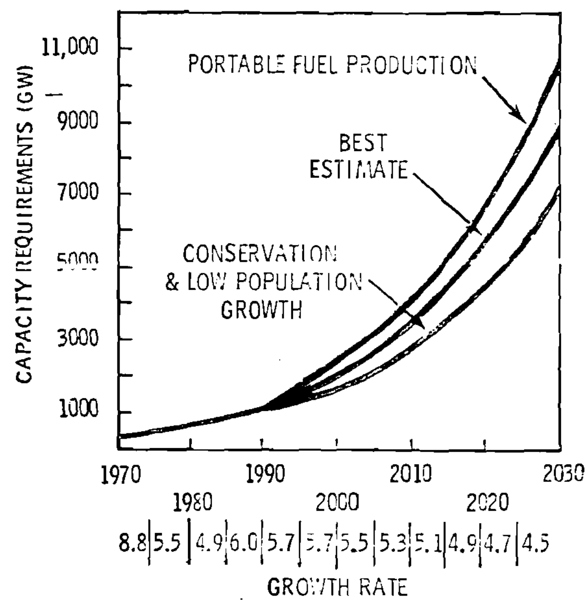


FIGURE 1. Projected Electrical Capacity Requirements

Preliminary Economic Analysis

JR Young

Operating and economic characteristics of the UWMAK-I conceptual power plant was developed by PNL and the University of Wisconsin study team. A definition was made of the reactor operating cycle, the plant factor, and the electricity production cost.

The operating cycle is the sequence of events between two consecutive times when the reactor is empty and ready to operate. Between two such times the reactor is loaded with fuel, the fuel is ignited , there is a fuel burn period, the burn is terminated, and the residual fuel is pumped out of the reactor. Definition of the operating cycle requires study of the operating characteristics of the reactor and its auxiliary equipment to determine the sequence and timing of the operating events necessary to complete the operating cycle.

The plant factor is defined as the percent attainment of the power generation that would occur if the plant operated 100% of the time at the design power level. Determination of the plant factor required estimation of four factors:

1) The length of the biennial wall replacement outages.
2) The amount of other outage time due to miscellaneous maintenance, plant modifications, and transmission system abnormal conditions.
3) The average length of the reactor burn periods.
4) The average length of the reactor rejuvenation periods between burn periods.

General correlations relating the plant factor to these four factors were developed. Analysis of these correlations showed that an 80% plant factor can be achieved with reasonable values for these four factors.

The electricity production cost was determined by estimating the capital, operating, and fueling costs for UWMAK-I. The capital costs were estimated by use of the standard Federal Power Commision capital cost summary forms. The costs for the major equipment items and structures were provided by Babcock and Wilcox, and Sargent and Lundy Engineers. The remainder of the capital costs were estimated by PNL and the University of Wisconsin by use of standard cost estimating procedures. Operating costs were obtained by estimating the size of the operation and maintenance force, the nature of the maintenance activities, and the probable use of materials and supplies. Normal operating costs such as support services, general and administrative, and working capital requirements were included. Fueling costs were determined by estimating fuel consumption and losses and then multiplying by the current market prices.

Preparations were made for development of these operating and economic characteristics for the UWMAK-II conceptual power plant early in CY-1975.

CTR Fusion-Fission Hybrid Regimes

DE Deonigi, RL Watts

In the analysis of controlled thermonuclear reactors adding a fission blanket to the fusion core reveals unique benefits. These benefits are reduced first wall loading, a reduction in the plasma performance required for overall thermodynamic breakeven, and improved utilization of excess neutrons produced by the plasma.

Ordinarily most of the energy recovered in a CTR will be generated in the plasma, transferred through the first wall and recovered in a suitable blanket external to the plasma. In a hybrid reactor having a fission blanket, much of the energy can be generated in the blanket and thus the first wall loading can be reduced by a factor as large as 10 to 1. This reduced loading can lengthen the life from the 2 years typically assumed. The economics of CTR energy production can be improved through the increased plant availability resulting from reduced first wall repairs and replacement.

The power amplification in the blanket which reduces the first wall loading will also reduce the plasma performance required for thermodynamic breakeven of the plant. The excess neutrons can be utilized in an appropriate blanket to generate fissile fuel for light-water or high-temperature gas cooled reactors.

Specific blanket designs will vary in the amount of power produced and the amount of fuel bred. Such blankets might be characterized as power producers at one extreme and fuel factories at the other.

Since detailed designs do not exist the estimated benefits and increased hybrid costs are expressed in discounted present worth terms in a form which can be compared to alternative power sources. Assumed values can be used in PNL's regional electrical supply model to determine the present worth costs and benefits which will allow substantial penetration of the energy market by the hybrid CTR. The discounted present worth of the hybrid benefits are compared to the discounted present costs (including Research & Development) to obtain a benefit-cost ratio.

The reduction in plasma confinement requirements indicates that the hybrid might be a transitional phase achievable before full commerical CTR's. Benefit-cost ratios will be affected if hybrids are introduced before standard CTR's.

Results of our preliminary analysis follow, and they indicate future trends in defining desirable performance windows in CTR and CTR-hybrid reactor designs that are needed for acceptable benefit-cost ratios. Additional objectives of future efforts are to define optimum fissile/electrical production ratios and discover synergistic hybrid designs.

STUDY BASIS

The inventory and generation of fissile fuel is a key cost element in a hybrid plant. Table 3 shows the fuel inventory and burnup characteristics assumed for the regional electrical supply study.

TABLE 3. Fissile Inventory and Annual Change

	Fissile Inventory KG	Change Fissile Annual
Power producer hybrid CTR	3,200	+1,900
Fuel factory hybrid CTR	800	+4,340
HTGR	1,975	- 466
PWR	1,756	- 520
FBR	2,150	+ 323

The fissile inventory is principally plutonium but may contain substantial amounts of ^{233}U and/or ^{235}U which are valued in the model.

Capital costs are assumed to vary according to market entry. The following table shows arbitrarily chosen capital costs as a function of time.

TABLE 4. Capital Costs, $/kW

	1980	1990	2000	2010
LWR	404	388	374	360
HTGR	405	389	374	360
FBR		458	434	410
Hybrid, Case H-1				
Power		638	628	598
Hybrid, Case H-2				
Power		575	561	548

Cost per kWt for the fuel generating reactors is assumed to be the same as that for the power producing reactors in Case H-1 and H-2.

LINEAR PROGRAMMING MODEL RESULTS

A base case was generated (Table 5) assuming that the CTR and CTR Hybrid were not available. This base case satisfied the energy demand by specifying enough LMFBR's to make up the deficit between lower cost sources and demand. In this solution the plutonium generated in the LMFBR is consumed by the LWR's and HTGR's.

TABLE 5. Electrical Generating Capacity (Number of 1000 MWe Plants)

Year	LWR	HTGR	LMFBR	Fossil	Hybrid	Total
			Base Case			
1970-79	78	0.3		167		246
1980-89	224	56	4.4	164		448
1990-99	427	120	217	47		811
2000-09	346	577	849	303		2,075
2010-19	261	496	2,365	1,310		4,432
2020-29	271	699	5,083	3,435		9,488
			Case H-1			
1970-79	78	0.3		167.4		246
1980-89	210	67		159	12	448
1990-99	182	345		25	245	797
2000-09	42	1,448	185		400	2,075
2010-19	42	2,613	778		999	4,432
2020-29	42	4,686	2,971		1,789	9,488
			Case H-2			
1970-79	78	0.3		167		246
1980-89	204	69		163	12	448
1990-99	183	341		23	264	811
2000-09	42	1,604			429	2,075
2010-19	42	3,219			1,171	4,432
2020-29	42	5,243	1,828		2,375	9,488

In Case H-1 and H-2, the CTR Hybrid was added. The fuel factory did not enter the solution at the assumed cost but reduced costs were obtained to allow entry of the fuel factory at any time. The generation of fuel is greater in the hybrid than in LMFBR's, therefore, the introduction of the hybrid in the solution results in more comparatively low cost LWR's and HTGR's being built in Case H-1 and H-2 than in the base case.

The greater fuel generation of the hybrid causes a decrease in the "market price" or shadow price of the Pu as indicated in Table 6 where plutonium values in the $5 range contrast with the $30-$40 range of plutonium value in the circa 2000 period for the base case.

TABLE 6. Plutonium Value, $/g

Year	Case H-1	Case H-2	Base
1974	2.08	2.27	3.42
1980	5.90	6.23	8.15
1990	16.97	16.49	25.97
2000	9.76	6.26	45.45
2010	7.75	4.73	52.73
2020	7.24	5.11	40.90
2030	3.86	2.89	20.24

BENEFIT COST RATIOS OF THE HYBRID

Substantial benefit-cost ratios result from the hybrid CTR program at the assumed capital costs for Case H-2 with the cost of the hybrid CTR program at approximately $10 billion, which is reduced by present worth discounting to about $3 billion at 8%.

Case H-2 shows a present worth of future benefits of about $14 billion giving a benefit to cost ratio of about 5 where the capital costs of the hybrid were assumed to be $575/kWe at 1990 decreasing to $548/kWe at 2010 and beyond (due to the learning curve).

The information obtained from this analysis is better seen in perspective as shown in Figure 2, where the assumed capital cost in $/kWe is shown as a function of the fissile fuel production. The CTR without breeding blanket is shown at the left. The fuel factory CTR which would generate about 4,200 kg/yr of Pu is shown on this figure reflecting the reduced values obtained from Cases H-1 and H-2 even though they didn't enter the solution.

The FBR is the natural reference for the study since it is the alternate means of satisfying energy demands in the future. The $460/kWe capital costs shown as a dotted line yielded satisfactory benefit-costs ratios in previous studies of the FBR.

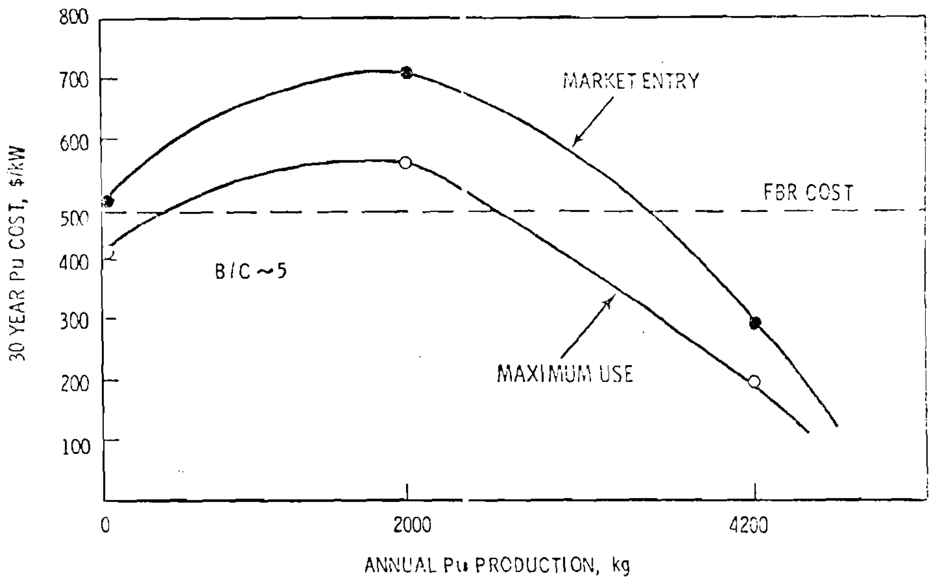


FIGURE 2. Annual Pu Production, kg

Future work is planned to better define the economic windows for the hybrid CTR. Additional values are needed to reasonably define the relationship between allowable capital costs and fuel versus energy production. Synergetic combinations and optimum values can result. Important parameters such as introduction date and electrical demand levels should be varied to determine their effect on the benefit cost ratio obtained.

Evaluation of Possible Fusion-Fission Technology Combinations

DT Aase, RM Fleischman, BF Gore, CM Heeb, BR Leonard, Jr., DF Newman, WC Wolkenhauer

The fusion-fission hybrid reactor is a concept which involves elements of both fission and fusion technologies. This type of reactor might very well play a useful role in the scenario of exploitation of fission power and the development of fusion power to commerical scale. The hybrid reactor could be developed sooner than CTRs on the basis of less stringent plasma requirements. Alternatively, the hybrid reactor could extend the supply of resources for fission reactors by producing fissile materials such as ^{233}U or plutonium. It might be applicable for reduction of waste produced in fission power reactors.

The concept of fusion-fission hybrid systems involves the coupling of the energetic neutrons from fusion reactions with fissile or fertile nuclei to produce a multiplication of the fusion neutron source strength and, ultimately, the production of energy from nuclear fission. There appear to be two distinct classes of hybrid reactors: 1) a reactor which serves as an adjunct to fission reactors by producing fissionable materials or reducing fissionable materials or reducing fission reactor waste, and 2) a self-contained electrical power plant which meets its own fissionable and fusionable material requirements.

Studies described in this report focused on the second class of hybrid system by combining a mirror fusion device with a urania-fueled, graphite-moderated, gas-cooled fission blanket to make a self-contained electrical power plant. This is just one of the many possible combinations of fusion and fission systems. The range of possibilities may be seen by noting that fusion neutrons from plasmas can be generated using injected machines or ignition machines either of which may operate on either D-D or D-T fuel cycles. The fission blanket may use either thermal neutron systems or fast neutron systems and there exist many combinations of neutron moderator and heat transfer systems for uranium, thorium, or plutonium fuel cycles. To reduce the number of possibities to a manageable set for this study, the fusion and fission systems are considered to operate only on current, or currently planned fuel cycles. Thus, the fusion devices are considered to operate on a D-T cycle and the fission fuel cycles are restricted to current or near term industrial capability (e.g., urania and/or urania-plutonia pellet fuels for LWR's LMFBRs, and GCFBRs).

The possible combinations of fusion and fission systems into a hybrid reactor may be viewed as elements of a matrix having fusion technology as one dimension and fission technology as the other. Such a matrix is shown in Figure 3, with candidate fusion technologies listed across the top and developed fission technologies listed from top to bottom in order of the state of their development.

FUSION TECHNOLOGY

FISSION TECHNOLOGY	TWO COMPONENT TORUS	TOKAMAK	MIRROR	THETA PINCH	LASER
LIGHT WATER MODERATED AND COOLED SYSTEMS					
HEAVY WATER MODERATED AND COOLED SYSTEMS					
GRAPHITE MODERATED GAS COOLED SYSTEMS		●		●	
LIQUID METAL COOLED FAST BREEDER SYSTEMS		●	●	●	●
GAS COOLED FAST BREEDER SYSTEMS					
MOLTEN SALT SYSTEMS		●	●		

● SOME STUDIES HAVE BEEN MADE IN THIS AREA

FIGURE 3. Evaluation of Possible Hybrid Reactor Combinations

The approach being used to evaluate the possible combinations of technologies is illustrated in Figure 4. Current descriptions of CTR and fission reactor systems are being used to define the technology of each. The range of fission technology considered is restricted to that of current manufacturing capability or planned for near term manufacture by industry. This restrictive view is taken to eliminate the need for substantial additional investment in fission technology for hybrid development. Design compatibility parameters for both the plasma and the fission blanket are defined to aid the development of criteria for preliminary screening of the possible combinations. The design compatibility parameters developed to date are listed in Tables 7 and 8. The design compatibility parameters along with other criteria in Table 9, form the bases for making a preliminary screening of possible combinations.

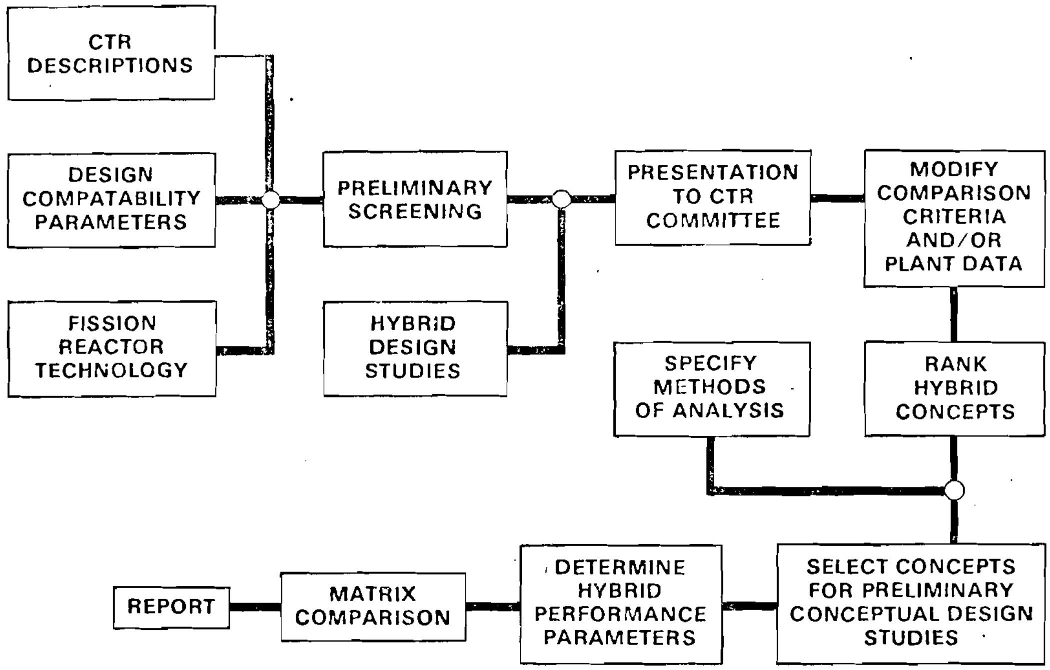


FIGURE 4. Evaluation Approach for the Hybrid Matrix Study

TABLE 7. Design Compatibility Parameters: Fission Blanket

- Coolant Characteristics

  Temperatures

  Pressures
  Flowrates

- Magnetic Field Characteristics

  Piping Requirements

- Neutronic Lattice Characteristics

  Reactivity

  Heat Generation

- Power Density - Average and Technical Specification Values

- Range of Flexibility in Fission Lattice Characteristics

- Current Subsystems Costs

TABLE 8. Design Compatibility Parameters: Plasma

- Type of Neutrons and Distributions
- Blanket Volume Characterization
    - Magnet Configuration
    - Structural Requirements
        - Plasma Injection
        - Divertor Design
        - Vacuum System
- Cycle Time and Duration
- Current Performance Range Considering Lawson Criteria, Wall Loadings, Etc.
- Subsystem Costs and Sensitivity to Operating Parameters

TABLE 9. Preliminary Screening Criteria

- Design Compatibility
    - Neutronic Compatibility
    - Structural Compatibility
    - Coolant Compatibility
    - Operational Compatibility
- Range of Flexibility in Plasma & Fission Lattice Technologies
- Potential Functional Capability (i.e., The Role)
- Potential Hybrid Cost Reductions

The potential functional capability (i.e., role) can be a criterion for screening. However, for the purposes of this study, the role is limited to an electrical power plant producing more fissionable and fusionable material than it consumes in a given fuel cycle. This narrow view is taken primarily because the design studies performed to date are mostly based upon this premise.

Data sources used in making a preliminary ranking of concepts for presentation to a peer CTR committee are shown in Figure 4. When a final ranking is established after peer review and comment, analytical study of key performance parameters for the most likely concepts will be made to firm up the technical basis for the ranking of the systems.

Preliminary Conceptual Design of a Mirror Hybrid Reactor

PNL/LLL Cooperative Study

WC Wolkenhauer, BR Leonard, Jr., UP Jenquin, CW Stewart, AM Sutey of Pacific Northwest Laboratories; RW Moir, JD Lee, RW Werner of Lawrence Livermore Laboratory, University of California

This study was to determine the potential of the hybrid as incorporated in the mirror fusion reactor. The underlying guideline was to employ, wherever possible, current technology for the fission and mirror fusion segments of the hybrid. The basic design objectives were to:

1) Produce electrical power.
2) Produce as much tritium as consumed.
3) Produce more fissionable material than consumed.

A safety criterion was to maintain the reactor subcritical at all times expecting that the system would not have to employ emergency coolant devices.

The major technical studies accomplished were:

- Development of a scope engineering design hybrid blanket within the general physical and geometrical guidelines established by LLL.
- Development, with LLL, of the parameters of the chosen reference system.
- Performance of neutronic analyses to parametrize the performance of the fission lattice pitch, fuel particle size, initial ^{235}U enrichment, initial ^{235}U to ^{239}Pu conversion ratio, fission lattice multiplication, and fission power multiplication.
- Evaluation of the steady state and transient thermal performance of the blanket.
- Calculation of the radioactive inventory of blanket materials for safety evaluation.

The conceptual design was built around the plasma characteristics as defined by the Livermore group. These plasma conditions are expected to be attainable with reasonable extrapolations of present technology. These plasma characteristics are shown in Table 10. The Yin-Yang coil system for this mirror hybrid was fed by four high-energy neutral-beam injectors.

PNL designed a hybrid lattice which was optimized for power production and fissile-fertile fuel utilization to fit in the design configuration. The blanket design consisted of a convertor region and a thermal fission lattice region. The inner blanket region, called a convertor, consisted of depleted-uranium-dioxide, with the uranium being 0.3% ^{235}U atom percent. The convertor and thermal fission lattice region are shown in Figure 5. The

TABLE 10. Mirror Hybrid Plasma Characteristics

PLASMA

Ellipsoid
3.5 m radius

NEUTRAL BEAM INJECTION

150 keV
500 Amperes
68 MW(e)

MAGNET

10 m radius
Central Field = 1.9 Tesla
Mirror Field = 7.1 Tesla

fission region consists of slightly enriched UO_2 fuel (1.35 wt% ^{235}U), graphite moderator, and helium coolant. The fission lattice is used for power multiplication.

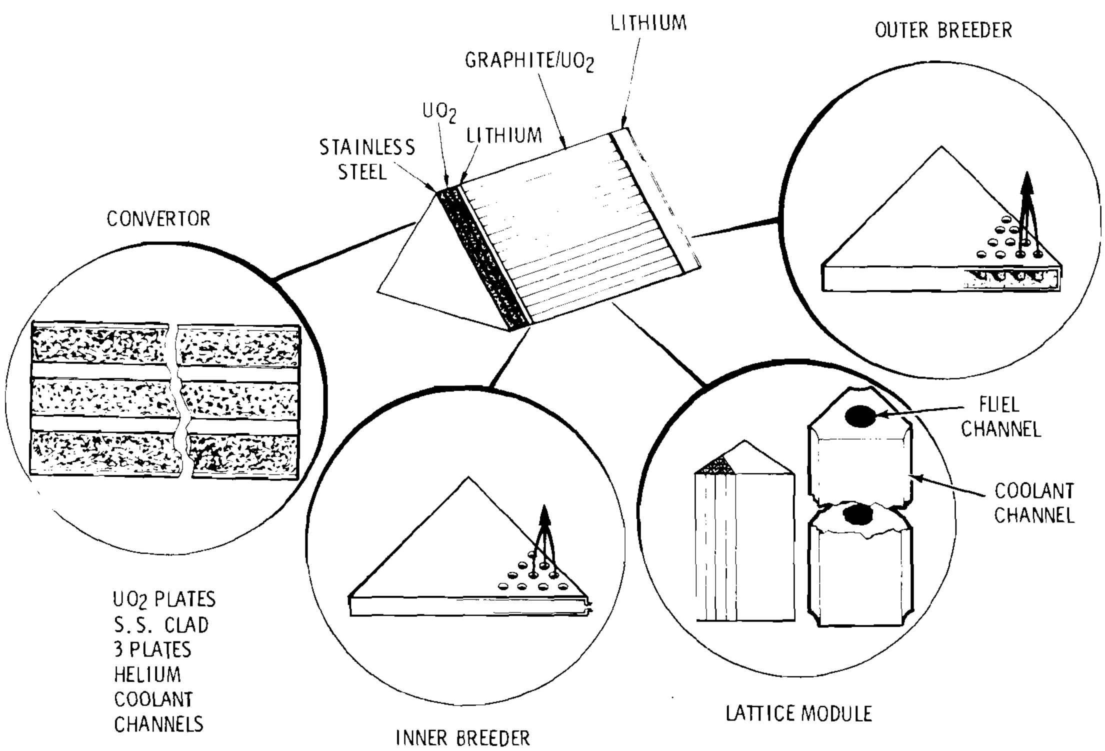


FIGURE 5. Mirror Hybrid Blanket Modules

A systems study inlcuding thermal hydraulics was done on this system which resulted in the characteristics shown in Table 11. An artist's conception of the reactor is shown in Figure 6 with a synopsis of the major milestones met in the project.

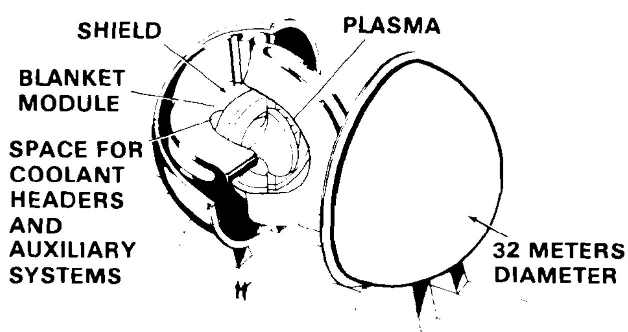
TABLE 11. Mirror Hybrid Performance Parameters

Electrical Power	664 MWe net
Circulating Power	240 MWe
System Efficiency	32%
Plasma Energy Multiplication	0.94
Blanket Power Multiplication	40 x 14.1 MeV

The status of the LLL mirror hybrid design results was reported in two papers presented at the First Topical Conference on the Technology of Controlled Nuclear Fusion at San Diego in April and in a paper presented at the 8th Symposium on Fusion Technology, Noordwiijkerhout, Netherlands, in June. A detailed technical report, BNWL-1835, covering the work has been issued. Summaries of two papers resulting from this study and the PNL extended evaluation in this report were also prepared and presented at the Fifth IAEA Conference on Plasma Physics and Controlled Nuclear Fusion Research held in Tokyo in 1974.

MIRROR HYBRID REACTOR

- DEVELOPMENT OF A CONSISTENT HYBRID DESIGN BASED ON THE MIRROR FUSION REACTOR



- FIRST ASSESSMENT OF HYBRID REACTOR SAFETY
  —FISSION LATTICE ALWAYS SUBCRITICAL
  —NO CORE MELTDOWN WITH LOSS OF COOLANT IN HIGH HEAT CAPACITY SYSTEMS
- ESTABLISHMENT OF A COOPERATIVE EFFORT IN HYBRID REACTOR DEVELOPMENT
  —LLL/PNL

FIGURE 6. Accomplished Milestones

Preliminary Conceptual Design of a Mirror Hybrid Reactor

Extended PNL Evaluations

BR Leonard, Jr., UP Jenquin, WC Wolkenhauer

Additional technical information was developed on the mirror hybrid performance characteristics which could be used in judging the technical feasibility of the concept. The analytical basis of the PNL/LLL Hybrid CTR conceptual design study was extended by additional studies evaluating:

- Burnup effects in the fissile blanket.
- Effects of thickening the blanket inner lithium layer.
- Sensitivity of calculated blanket performance to certain nuclear data deficiencies.
- Neutronics effects of the broad energy distribution of mirror CTR source neutrons.

In the initial design study certain marginal aspects of blanket performance were identified. These were:

- The degree of blanket subcriticality.
- The tritium production rate.
- The spatial power distribution in the fission lattice.

The importance of these performance aspects, and the possibility of changes resulting from a more complete investigation, provided the incentive for performing further analyses.

The reference system for these studies is that produced by the PNL/LLL cooperative study, and is summarized in Table 12. The general conclusion drawn from these studies is that the previous work is valid. A mirror hybrid fusion reactor could be built based upon the reported blanket design which would be a viable electric power source. Additional results and conclusions are also reported.

TABLE 12. General Features of the LLL/PNL Mirror Hybrid Conceptual Design

Region	Dimension
Plasma Radius	3.5 m
Stainless Steel Inner Wall Radius	5.0 m
Depleted (0.3 wt% ^{235}U in U) UO_2 Converter Region Clad in Stainless Steel (Cooled with Helium)	8.5 cm thick
Natural Lithium Region for Breeding Tritium	1.5 cm thick
Fission Lattice Region, 1.35 wt% UO_2 Clad in Stainless Steel	180 cm thick
Natural Lithium Region for Tritium Production	10 cm thick

BURNUP BEHAVIOR

The energy multiplication of the blanket fission lattice is directly related to the neutron multiplication value k_{eff}. To obtain high energy multiplication requires that k_{eff} not be small. However, for the thermal fission lattice studied here, k_{eff} varies significantly with temperature, increasing as the lattice is cooled from hot operating conditions to room temperature. Consequently, a lattice with a high energy multiplication at operating temperatures will have a k_{eff} which approaches unity at room temperature. During operation fissile material will be produced in the lattice, and burnup calculations must be performed to ensure that k_{eff} does not become unity at any time.

A series of neutron balance calculations were performed for the reference lattice operating at the design specific power level of 4.3 MW/m^3. These calculations were performed at burnup steps corresponding to 50 days operation. The calculations were admittedly approximate but provide an adequate estimate of the time dependent neutronic behavior of the blanket. The major approximations made in the calculations were:

- Representation of the average flux for the entire blanket.
- Recalculating the spectrum averaged cross sections at only one time step (100 days) of burnup.
- Neglect of captures in 2.35 d ^{239}Np.
- Use of an incomplete fission product model.

Burnup calculations were not made for the convertor region, since the isotopic changes of materials in this region are assumed to be small and, therefore, not have a significant effect on the neutronic behavior.

The change in lattice multiplication value during irradiation is listed in Table 13 for both hot operating and room temperature conditions. As shown, the k_{eff} value at hot operating increases by $\sim$ 5% in the first 50 days of power operation. This is due to plutonium buildup in the lattice. After 50 days, the multiplication value decreases. Initially, the saving in k_{eff} between room and hot operating temperatures is about 14%.

TABLE 13. Lattice Multiplication Values as a Function of Burnup

	Multiplication Value, k_{eff}	
Full Power Days	Hot Operating	Room Temperature
0	0.842	0.981
50	0.886	- -
100	0.871	0.916
150	.856	0.883
200	.838	- -

With burnup, the multiplication defect between hot operating and room temperature conditions decreases during power operating, decreasing to 25% at 100 days. This is due to the relative difference in neutron competition between uranium and plutonium at hot operating conditions (i.e., Pu competes more favorably for neutrons at hot conditions). On the bases of these results, the lattice is assumed always to be subcritical during normal operating conditions.

Some of the other changes in blanket performance parameters due to burnup are listed in Table 14. After 200 days of full power operation, the fuel has accumulated 5000 MWd/MT of exposure. Assuming a linear exposure accumulation (i.e., 25 MWd/MT/day) then in 300 days of full power operation (which is about the plan of current LWR power plants), the exposure accumulation would be $\sim$ 7500 MWd/MT. This is about the annual exposure accumulation of boiling water reactor (BWR) fuel. In contrast, fuel in pressurized water reactors accumulates roughly 11,000 MWd/MT annually whereas HTGR fuels accumulate about 22,000 MWd/MT annually.

During 200 days operation 315 kg of ^{235}U are burned and 340 kg of plutonium has formed. Though more plutonium has been generated than ^{235}U burned, the power output has been reduced. This is evidenced by the lower value of k_{eff} (hot operating) given in Table 13 and the blanket thermal energy multiplication value given in Table 14.

TABLE 14. Performance Parameters of Hybrid Blanket with Power Operation

Parameter	Initial	200 Days Operation	Average Value
Fuel Exposure in Fission Lattice	0	5000 MWd/MT	25 MWd/MT/day
^{235}U Inventory (Enriched Lattice)	1000 kg	685 kg	
Total Plutonium Inventory	0	340 kg	
Blanket Thermal Energy - 14.1 MeV	40	35	45
Tritium Produced per DT Event	1.065	0.88	1.07[a]

[a] Not corrected for source neutron losses through plasma and beam ports.

The initial increase in hot multiplication of the lattice (50 days burnup) results in a significant increase in tritium breeding. By 200 days operation, however, the tritium production has reached the value of 0.88 and is steadily decreasing.

After 200 days of full power operation more tritium is being consumed than is being produced. The energy multiplication in the blanket, and thus the power output, is decreasing. These conditions may signify that the useful blanket residence time has been reached.

MODIFICATION OF LITHIUM REGION

The reference lattice design had an initial conversion factor of 0.973. Inability to breed tritium was not considered to be a major design constraint for two reasons. First, the tritium consumption in the mirror hybrid is so small that full conversion may not be a necessity. Secondly, the tritium conversion was expected to improve with buildup of Pu, as was later confirmed by calculation. Through a minor modification of the reference design it has proven possible to increase the tritium conversion ratio above unity, with insignificant changes in other reaction rates in the blanket. This is accomplished by increasing the thickness of lithium in the convertor region from 1.5 cm to 4 cm, and maintaining the blanket thickness by reducing the outer lithium layer correspondingly. The effect of the additional lithium is to increase the captures of epithermal neutrons leaking from the lattice. The tritium production shown in Table 14 includes the effect of this change, and also the effects of some nuclear data changes.

NUCLEAR DATA

The ENDF/B-III data used as the reference set for these calculations have known deficiencies in the ^{238}U data. These deficiencies are primarily in the description of the energies of the secondary neutrons from n,2n reactions, and from neutron inelastic scattering for incident 14 MeV neutrons. The ENDF/B-III descriptions give secondary neutrons from these reactions whose average energies are much too low to be physically realistic. Haight and Lee compared calculations using ENDF/B-III data in thick ^{238}U hybrid blankets with the results obtained using more realistic LLL evaluated data and found significant differences.[1] Consequently, a set of newly evaluated data were developed for inelastic scattering and for the secondary energy distributions for the n,2n and n,3n reactions on ^{238}U and these were used to replace the ENDF/B-III descriptions.

The effects of these modifications on the performance of the convertor region of the hybrid blanket were estimated by making ANISN calculations in which the thermal fission lattice was replaced with graphite to eliminate lattice effects. The results of the calculations, compared to results of identical calculations using ENDF/B-III data, showed expected differences. However, the differences were small, primarily because the convertor is fairly thin to the mean free path of 14 MeV neutrons.

The most significant changes observed were an increase in tritium breeding of about 2.5% and in an increase in the maximum neutron flux in the graphite ranging from 3% in the thermal group to about 9% for a large portion of slowing down flux. The increased flux would be expected to give an increase in the energy multiplication of the blanket of 5 to 10% or the improved convertor performance could justify use of a thicker convertor. Flux levels that were predicted in the graphite region in calculations using the ENDF/B-III data could be achieved using the new data even when the convertor thickness was increased from the reference value of 8.5 cm to 12 cm. In addition, with the thicker convertor tritium breeding was increased by about 1% and Pu production in the convertor was increased by about 60%.

14 MeV NEUTRON SOURCE SPECTRUM

The energy spectrum of D-T source neutrons which result from a uniform isotropic Maxwellian velocity distribution are quite broad and could result in large changes in calculated reaction rates, particularly for threshold reactions, relative to the rates resulting from low temperature (10-20 keV) plasmas.[2] The effect of this phenomenon appears to have received almost no attention in mirror or Tokamak plasma concepts driven by energetic neutral beams. Since the ion energy distribution in the driven mirror reactor is not Maxwellian, a computer program was written which performs calculation of the source neutron energy distribution from a tabulated ion energy distribution.

An ion energy source spectrum calculated by LLL for injected beams with an average energy of 100 keV has been used to calculate the source neutron energy spectrum as shown on Figure 7. The measured full width at half-maximum of the distribution is 1.70 MeV centered at 14.20 MeV. This distribution is only slightly narrower than that obtained by Muir for a Gaussian approximation and the wings are truncated relative to a Gaussian at both low and high energies because the ion energy spectrum is similarly truncated.

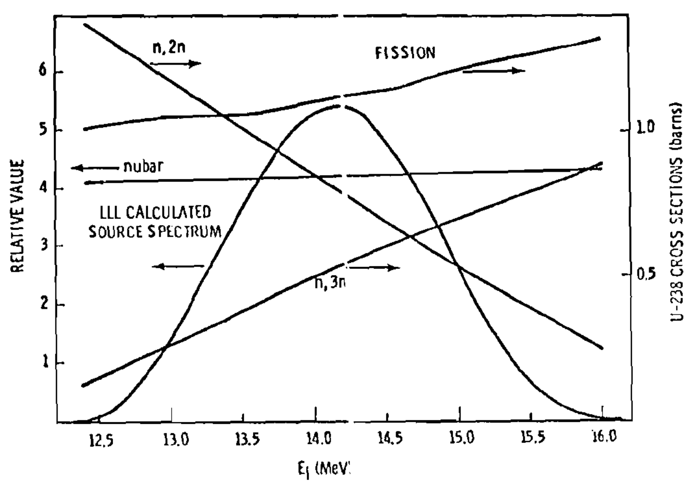


FIGURE 7. The LLL Calculated Source Spectrum is the Neutron Source Spectrum Calculated from a 100 keV Injected Ion Energy Spectrum. Overlayed are the Important Reaction Cross Sections as Described in ENDF/B-IV.

Also shown on the figure are the descriptions of the pertinent cross section data for ^{238}U contained in the ENDF/B-IV files. To estimate the importance of this source effect, cross section values for ^{238}U obtained from averaging over this spectrum have been calculated and compared to the 14.07 MeV values. The largest effect was an 8% increase in the n,3n reaction rate with a net effect of +3% for eta (the number of secondary neutrons produced per non-elastic collision). These changes may produce significant effects in subsequent generations of neutrons and do not address the fact that the average energies of secondary neutrons are also changed.

The energy multigroup structure used in our reference lattice calculations has been modified and extended to include five energy groups in which the neutron source is contained. In addition, a new set of multigroup cross sections has been obtained for all lattice materials using ENDF/B-IV data and including the extended neutron source as part of the weighting function. Only small changes in integral performance parameters of the hybrid lattice were observed. More significant effects might result from different data descriptions.

SUMMARY AND CONCLUSIONS

The major conclusion of our previous studies remains unchanged -- a mirror hybrid fusion reactor could be built based on this blanket design that could be a viable electrical power source. In addition, the probable breeding of both fissile material and tritium has been confirmed by detailed calculations of the behavior of the blanket through an extended operating period. Neutronics calculations at operating temperatures during the operation have also demonstrated that these calculations are a necessity in arriving at valid projections of the performance and safety of the hybrid system. The sensitivity of the performance of this system to nuclear data has been assessed and shown that design improvements could be made using better estimate data. The effect of the extended source neutron energy spectrum has been investigated and shown to be of possible significance in the nuclear performance and design characteristics of plasma devices driven by energetic neutral beams.

Remote Assembly/Disassembly of Fusion Reactors

PL Peterson, WS Kelly, MG Patrick

The reactor core of the fusion reactors to be operated in the future will need to be designed to be remotely assembled and disassembled following startup due to induced radiation. Ancillary equipment and systems near the core will require a similar capability. Inadequate equipment and procedures will extend reactor down time and add to reactor operating costs.

Pacific Northwest Laboratory assisted Lawrence Livermore Laboratory during 1974 in the development of a concept for a Fusion Engineering Research Facility (FERF).* PNL's primary input was in the area of remote assembly and disassembly with other inputs in the test loops and sample handling areas. These contributions were based on experience accumulated by PNL personnel in the design and operation of remote operating and control systems for nuclear reactors and other experimental facilities at Hanford.

The procedures for remote assembly and disassembly of the FERF reactor assembly and associated equipment were formulated with the following objectives in mind:

- completely remote operation (i.e., no personnel in the vault)
- all devices to operate in a fail-safe manner
- operations to be automated if possible
- redundancy of backup systems to be provided where feasible
- lifting by single point suspension to be avoided if possible

Essentially the same equipment would be used for both assembly and disassembly. The handling systems will fall into two general categories: 1) primarily fixed or rigid systems and 2) portable or flexible systems with universal application. The fixed systems would consist of supporting dollies permanently attached to the large reactor subassemblies operating on tracks imbedded in the floor of the vault. Subassemblies such as the expander tanks, first and second wall assemblies, and half tanks would be mounted on the dollies which would support the entire reactor assembly.

Reactor system components that cannot be supported by permanent dollies or that cannot be mounted by pure translational movement would be handled by a combination of lifting

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* Details of the work are reported in the LLL document Conceptual Design of a Mirror Reactor for a Fusion Engineering Research Facility (FERF), UCRL 51617, August 28, 1974.

devices, portable dollies and special purpose temporary fixtures. Positioning and clamping of the mating surfaces of individual components prior to pumpdown would be similar for all reactor subassemblies.

The initial concept drawings of the reactor vault indicated a vault approximately 60 ft high and 55 ft wide. A slot or trench down the centerline of the floor of the vault was provided to reduce the height of the assembled reactor above the floor by permitting projecting portions of one half tank and one expander tank to extend below the floor level. The trench provided for separation of the major subassemblies to permit access to the inner components.

However, the trench would limit the distance and direction that the major subassemblies can be backed off. This could be avoided by mounting the entire reactor structure above the floor level. Two sets of parallel tracks along each wall of the vault with appropriate switching mechanisms would permit the components from the inner core to be placed on dollies which could then be shunted around the subassemblies that had been backed off. Mounting of the entire assembly above the floor would further simplify mounting and dismounting of components beneath the reactor structure by avoiding operating within the confines of the trench. The overall height of the vault would not have to be significantly increased because no lifting of components up and over other subassemblies would be required.

All assembly and disassembly operations would be monitored and conducted from a central control room using remotely operated TV monitors. A console within the control room would contain all of the required system operating controls, TV monitoring screens and instrumentation readouts.

The atmosphere within the reactor vault was assumed to be nitrogen at all times. Therefore, all equipment and instrumentation within the vault would have to be capable of operating in nitrogen. Personnel entering for any reason would require a self-contained breathing apparatus.

One conclusion arising from participation in the study with Lawrence Livermore Laboratory was that equipment and procedures available at the time of reactor core design could put significant constraints on the design in terms of maximum size or configuration. Prior development of equipment for remote handling components common to all types of fusion reactors such as large scale vacuum seals, manifolds for cooling fluids, alignment devices and clamping fixtures will permit greater flexibility in the reactor core and overall system design.

Nuclear Standards - BR Leonard, Jr., KB Stewart

Obtaining improved nuclear data and data files for the neutronic analysis of fusion reactor blankets is the objective in this area. Data deficiencies are defined by the critical review of data used in the analysis of fusion reactor systems, including hybrid reactors, at PNL. Improved data and data files are obtained by evaluation and through participation in the Cross Section Evaluation Working Group (CSEWG) and the U.S. Nuclear Data Committee (USNDC). Since the last report, meetings of the CTR and the Standards Subcommittees of the USNDC and a meeting of CSEWG was attended. As part of that effort, the new data file for ^{6}Li submitted for Version IV of ENDF/B was reviewed, minor errors corrected, and the file approved for ENDF/B-IV. In addition, needs for hybrid reactors and for fission product transmutation in CTRs which had been previously identified[3] were submitted for inclusion in the USNDC document "Compilation of Requests for Nuclear Data."

A review was made of the evaluated data files of ENDF/B-III for ^{238}U which has been used in PNL hybrid neutronic analyses. Comparisons of the ENDF/B and LLL descriptions of the secondary neutron energies distributions resulting from inelastic scattering and n,2n reactions of initially 15 MeV neutrons on ^{238}U are shown in Figures 8 and 9. The LLL descriptions are judged to be better founded physically and would lead to larger neutron multiplication and deeper penetration of incident 14 MeV neutrons in ^{238}U. As a result of the deficiencies identified in the ENDF/B-III ^{238}U data files, the magnitude and secondary neutron energy distributions of inelastic scattering and the secondary neutron energy distributions of n,2n and n,3n reactions were re-evaluated and prepared to modify the ENDF/B-III files. An example of one of the newly evaluated secondary neutron energy distributions from n,2n reactions is shown in Figure 10.

A review of CTR nuclear data requirements was made in cooperation with investigators from other laboratories and the results were reported in a session of invited papers on Fusion Blanket: Shielding and Cross Section Studies at the Winter ANS Meeting held in Washington, D. C.

The processing code ETOG was modified in order to be able to process nuclear data in the format specified for the ENDF/B-IV data file.

A set of multigroup (30 group) cross sections from ENDF/B-IV was developed for use in ANISIN in the mirror hybrid studies. The tape contains microscopic data for the 15 isotopes

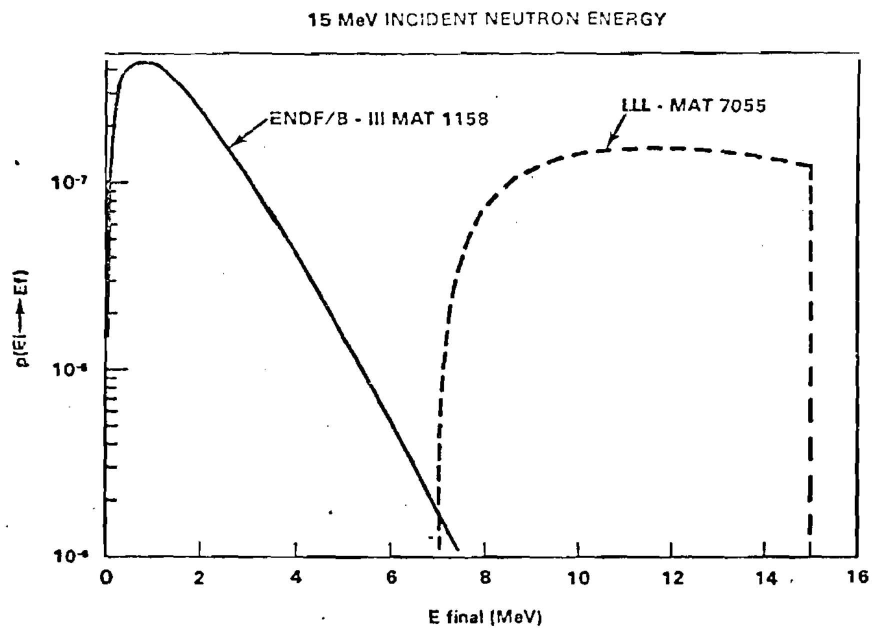

FIGURE 8. Energy Spectrum of Neutrons Inelastically Scattered on U-238

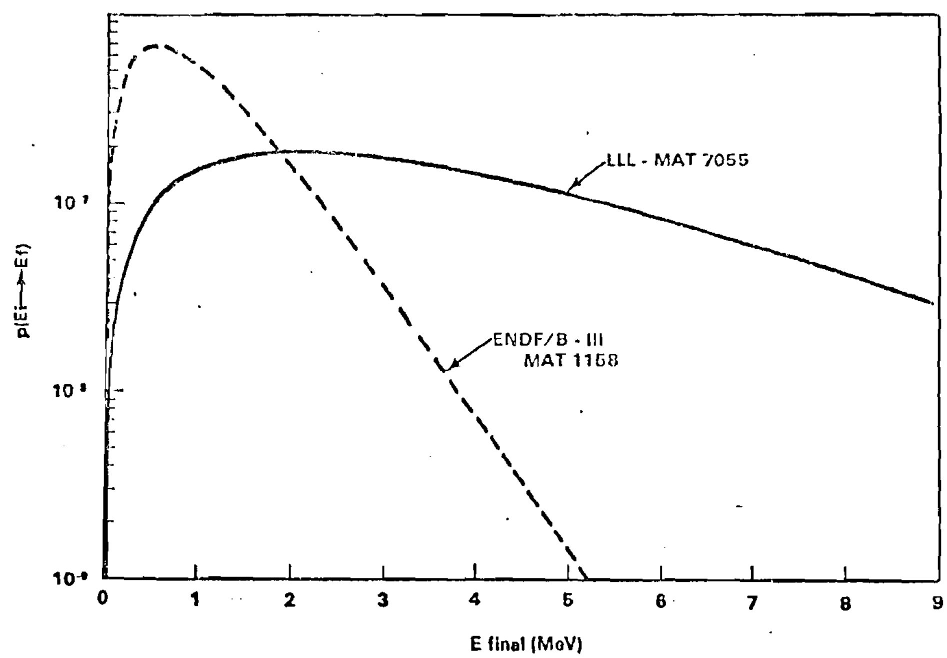

FIGURE 9. Energy Spectrum on Secondary Neutrons from n,2n Reaction on U-238

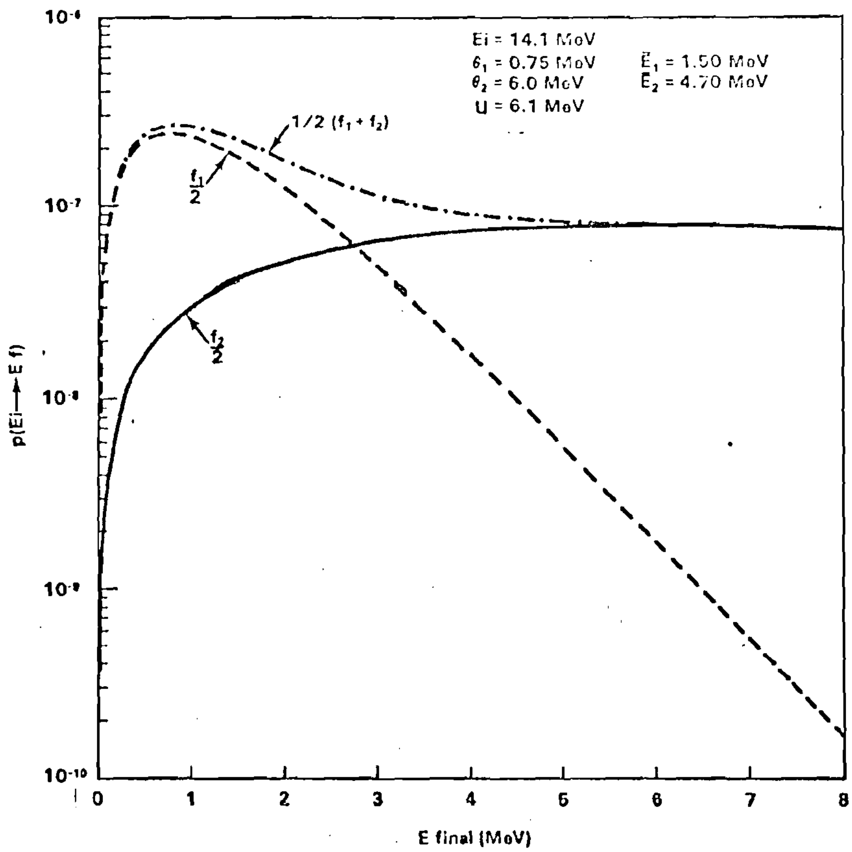

FIGURE 10. A New Evaluation of the Energy Spectrum of Secondary Neutrons from n,2n Reactions on U-238

listed in Table 15. The group structure is shown in Table 16. The weighting function consisted of a 1/E flux shape up to $\sim$ 13.2 MeV with a near-Gaussian centered about 14.2 MeV. The spike has a full width of 1.7 MeV at half maximum. Cross sections from thermalization calculations have been substituted into groups 28, 29, and 30. For materials 8 through 18, the data have been self-shielded in groups 16 through 27. The data represent a fuel zone without any moderator. All the other data assume infinitely dilute cross sections. Stainless steel was created by mixing iron, chronium, and nickel together with atom fractions of 0.7, 0.2, and 0.1, respectively. Each isotope has P_0, P_1, P_2, and P_3 microscopic data. Each group contains 37 entries with the total cross section in position 8. Thus, $\nu\sigma_f$ is in position 7 and σ_a is in position 6. Position 6 does not contain n,2n and n,3n cross sections. These are in positions 3 and 4, respectively. The scattering matrix accounts for neutron multiplication via the n,2n and n,3n reactions. Position 5 contains the fission cross section and position 1 contains the capture cross section, n,γ. Position 2 contains reactions which result in alpha production.

A copy of this tape has been sent to the Radiation Shielding Information Center for other investigators who are interested in using these data in their research.

TABLE 15. Isotopes on the Tape

Number	Isotope
1	Carbon
2	Oxygen
3	Silicon
4	Helium
5	Lithium-6
6	Lithium-7
7	Stainless Steel
8	Uranium-235
9	Uranium-238
10	Plutonium-239
11	Plutonium-240
12	Plutonium-241
13	Plutonium-242
14	Uranium-233
15	Thorium-232

TABLE 16. 30-Group Structure for Fusion-Fission Calculations

Group	Upper Energy, eV	ΔU	U
1	18.22×10^6	.15	-.60
2	15.68×10^6	.05	-.45
3	14.92×10^6	.05	-.40
4	14.19×10^6	.05	-.35
5	13.50×10^6	.1	-.3
6	12.21×10^6	.1	-.2
7	11.05×10^6	.1	-.1
8	10.00×10^6	.5	0
9	6.07×10^6	.5	1.0
10	3.68×10^6	.5	1.0
11	2.23×10^6	.5	1.5
12	1.35×10^6	.5	2.0
13	$.821 \times 10^6$	.5	2.5
14	$.498 \times 10^6$	.5	3.0
15	$.302 \times 10^6$	.5	3.5
16	$.183 \times 10^6$	1.0	4
17	67.4×10^3	1.0	5
18	24.8×10^3	1.0	6
19	9.12×10^3	1.0	7
20	3.36×10^3	1.0	8
21	1.23×10^3	1.0	9
22	$.454 \times 10^3$	1.0	10
23	$.167 \times 10^3$	1.0	11
24	61.4	1.0	12
25	22.6	1.0	13
26	8.32	1.0	14
27	3.06	.5	15
28	1.855	.464	15.5
29	1.166	.624	15.964
30	.625		16.588

Analytical Capability

Fusion Neutron Spectrum Code (FUSEP) - DL Lessor
Thermal Hydraulic Model (TRUTH) - CW Stewart, AM Sutey

Two technical developments needed in the process of performing the mirror hybrid design analysis which enhance analytical capability for CTR design and analyses were the calculation of the energy distribution of the source D-T neutrons from injected machines, and the calculation of steady state and transient thermal performance of the mirror hybrid blanket.

FUSEP

Program FUSEP was developed to calculate neutron energy spectra produced by fusion reactions for use in fusion reactor blanket calculations.

FUSEP takes as input the energy distribution of ions in an isotropic plasma of up to 3 ion species with neutron-producing fusion reactions between them. The energy spectrum of the fusion neutrons is calculated by integrating the product of ion velocity distributions, relative velocity, and cross section per unit final neutron energy over ion velocity space. Fitted cross section expressions having Coulomb barrier penetration and nuclear resonance factors are used.[4] Neutron emission is assumed isotropic in the center of mass frame of colliding ions. FUSEP has been used to calculate neutron spectra from D-T and D-D reactions in Maxwellian plasmas and from a calculated plasma spectrum from 100 keV neutron injection into a mirror machine.[5]

TRUTH

A digital computer code called TRUTH was developed to aid in determining the steady state and transient thermal performance of the blanket. The mathematical model consists of an energy balance between any arbitrary collection of interconnected subvolumes. The resulting equations are solved with an explicit numerical procedure which computes the total heat exchange between subvolumes by the use of previously calculated temperatures and thermal resistances. The temperature of each subvolume is updated on the basis of its total heat capacity and the energy it receives from adjacent subvolumes during a time step.

The thermal resistance between nodes optionally includes the effects of conduction, forced and free convection, radiation and fluid flow. The conduction resistance is considered to be constant whereas the respective surface coefficients for the other heat transfer mechanisms are treated via correlations with temperature or flow rate as independent variables.

MATERIALS RESEARCH AND RADIATION ENVIRONMENT SIMULATION

Knowledge of high energy neutron interaction with materials is a high priority need in the development of a controlled fusion power plant. Significant factors in the design and operation of future CTRs as well as experimental fusion machines include neutron sputtering and radioactive atom ejection resulting from direct nuclear recoils in the near surface regions. Experiments have been initiated to determine the absolute sputtering ratios for candidate first wall materials and to obtain the absolute yield of radioactive atoms ejected from materials as a result of (D,T) neutron bombardment. Upper limits for sputtering ratios have been obtained which, when confirmed, would indicate that neutron sputtering will not present a major problem with first wall erosion of future CTRs. However, major radioactivity problems can be expected to occur in the cooling ducts and heat exchangers of fusion reactors due to radioactive material ejected from surfaces under neutron bombardment. The results from these experiments are expected to provide important input in the design of future fusion reactors.

Radiation induced structural damage in potential first wall materials is being characterized by heavy ion bombardment. Currently the effect of parameters uniquely associated with ion bombardment are being evaluated to more thoroughly understand the ion simulation technique and make better correlation with neutron irradiation. Results from studies of the surface effects in ion bombarded molybdenum and the effect of helium on void formation in molybdenum are presented. A decrease in surface void size with dose rate was found to agree reasonably well with the difference reported for neutron irradiated molybdenum. A strong helium effect was not found for the bombardment conditions. This is in agreement with nucleation theory and is attributed to the very high displacement rates used in the experiment. An interlaboratory ion correlations experiment has also been planned using molybdenum from a common source. The plan is to compare and correlate the results from the experiments conducted using common experimental conditions. Support is being provided to the Naval Research Laboratory in the area of ion bombardment studies. In addition to participation in the BCC Ion Correlation experiment, NRL will examine some neutron irradiated molybdenum specimens for comparison with ion bombardment results.

Since charged particle bombardment may simulate fusion neutron damage levels with rates in excess of existing fission reactor radiation facility, the extent to which penetrating ions simulate mechanical property damage in metals is being studied for application to candidate CTR alloys. The data input and major tasks required for such a program have been identified and a technical advisory committee established. Arrangements are also being made for integration of this work with related studies at other laboratories. For example, a joint effort is being conducted with Hanford Engineering Development Laboratory for damage analysis and neutron irradiation of relevant CTR materials.

Many of the CTR concepts currently being developed contain graphite in the blanket and/or shield regions. In addition, some recently proposed concepts have suggested the use of graphite or graphite cloth between the plasma and first structural wall. These new concepts pose some special problems which are being assessed. Specific areas being investigated include the evaluation of radiation damage effects, the rate of sputtering of carbon from the surface, outgassing problems, potential for carbon removal and transport, and reactions with impurities in gaseous coolants. Plans are also being made for experimentally testing the correlation between damage production rates in fission and fusion reactor spectra.

A portion of the literature dealing with the radiation effects on aluminum, a metal important to fusion reactor design, has been reviewed to identify applicable data. The volume of information does not compare with that for copper, nickel, or the noble metals or approach the enormous amount of literature dealing with stainless steel. However, aluminum has been studied in some detail and information is available on defect characteristics, voids, transmutations, and mechanical properties. Surface damage effects must be extrapolated from data for refractory metals since adequate data does not exist for aluminum. In addition, evaluation of aluminum for fusion design applications generally requires higher temperature and fluence irradiations than have been conducted so far.

Although pressurized helium has several advantages for fusion reactor coolant

Although pressurized helium has several advantages for fusion reactor coolant application, the best obtainable helium purity is not good enough to entirely prevent some corrosion reaction with reactive metal at elevated temperatures. Thus, experiments are being initiated to study the compatability of fusion reactor materials with helium using gas loops and autoclaves in a vacuum facility designed for studies of metal reactions in contaminated helium. The materials to be studied include aluminum base, iron base, and nickel base alloys and refractory metals.

Helium embrittlement is one of the radiation-induced damage mechanisms which may limit the life of first wall material. Since helium generation rates will be higher in fusion systems than in fission systems, methods for He-charging other than fission irradiation need to be investigated. One method, the tritium trick, is currently being studied to determine its applicability to the study of helium effects in metals. The study includes the determination of the mechanical properties of niobium and vanadium at various helium concentrations. Embrittlement did not occur for low (< 100 appm) helium concentrations in either metal; higher helium content specimens are currently being tested. Helium bubbles at grain boundaries, a postulated requirement for embrittlement, were not found in either metal at the test temperature of one-half the absolute melting point. Bubbles were detected in vanadium grain boundaries after a 1200°C anneal; this pretest anneal technique will be employed to determine helium bubble effects on mechanical properties.

Current thermonuclear reactor technology requires the use of electrical insulators in various locations which will be subjected to severe irradiation and thermal conditions. A program is underway to study and evaluate the electrical and related properties of potential insulators for CTR design. A joint Battelle-Columbus and Battelle-Northwest review and evaluation of electrical insulators was completed and published. The results of this review were used as the basis for initiating a program to experimentally measure the electrical and irradiation properties of insulators. The study is directed toward measurements of electrical resistivity and dielectric breakdown. Measurements have begun on reference oxides and on less known insulators which are isotropic and open crystallographic structures, e.g., yttrium oxide, alumina-magnesia spinel, and yttrium aluminate.

A study has been initiated to assess the current and projected availability, producibility, and fabricability of candidate first wall structure materials. The objective is to identify major resource or fabrication problems in time to develop needed supply and/or component fabrication technologies or to redirect design efforts to circumvent the problems. Data on current and projected material supply, resources, and industrial capacity relevant to CTR material demands are being compiled. The information ultimately will be reduced to a summary presentation which can be updated periodically.

Sputter deposition may provide an excellent method for producing certain materials such as multifilamentary superconductors, insulator coatings, and barrier surfaces. Methods have been developed to make multifilamentary Nb-Al-Ge and Nb-Al superconductors containing 10^6 or more filaments per square centimeter in a void-free matrix of high conductivity stabilizer such as copper. Coatings of Al_2O_3 up to 14 mils thick (sufficient for the theta pinch fusion reactor) have been prepared by high rate sputter depositions. Experimental work on the deposition of aluminum nitride are also underway, as well as major improvements in the sputtering technique. Plans have been made to test the electronic properties of the coatings by staff members at BNL, LASL, and PNL.

Surface Science Research Related to Fusion Reactor Technology

OK Harling, MT Thomas, DL Styris, RL Brodzinski, LA Rancitelli

Studies in the surface science area involve 1) the determination of absolute fast neutron sputtering ratios for candidate CTR structural materials including several refractory metals, and 2) measurements of the absolute yields of radioactive isotopes emitted from the surfaces of candidate CTR materials as a result of (D,T) neutron bombardment.

The structures of fusion reactors, (CTR's) will be subjected to intense radiation fields including fast neutrons. In the first wall region, 14 MeV neutron currents are expected to range between 4×10^{13} and 4×10^{14} n/cm^2-sec(1-10Mw/m^2). Furthermore, the total flux of MeV neutrons will be considerably higher than the primary source current.

A variety of neutron effects have been identified as significant factors in the design and operation of future CTR's as well as experimental fusion machines. Neutron effects on the surfaces of the CTR first wall include neutron sputtering and radioactive atom ejection resulting from direct nuclear recoils in the near surface region. The ejection of radioactive as well as non-radioactive material from the surfaces of the CTR wall structure, has a number of important implications. These include potential thinning and weakening of structures, plasma contamination, and problems with radioactivity in the cooling ducts and heat exchangers of the CTR.

During CY 1974 experiments were initiated to determine the absolute sputtering ratios for candidate CTR first wall materials. PNL has also begun measurements to obtain the absolute yields of radioactive atoms ejected from CTR materials as a result of (D,T) neutron bombardment. Some initial results for neutron sputtering ratios for Nb and Au are given in Table 17.

TABLE 17. (D,T) Neutron Sputtering Yields by Neutron Activiation Analysis

TARGET MATERIAL	FLUENCE (D,T) NEUTRONS $\times 10^{-15}$	FORWARD atoms/ (D,T) neut.	BACKWARD atoms/ (D,T) neut.
Nb			
Single X'l	5.6	$\leq 4 \times 10^{-4}$	$\leq 6 \times 10^{-4}$
Annealed Foil	5.1	$\leq 4 \times 10^{-4}$	$\leq 3 \times 10^{-4}$
CW Foil	3.6	$\leq 8.3 \times 10^{-4}$	$\leq 8.3 \times 10^{-4}$
CW Foil	12.2	$\leq 1.1 \times 10^{-5}$	$\leq 1.5 \times 10^{-5}$
Au			
Annealed Foil	6.2	$\leq 3 \times 10^{-5}$	$\leq 1 \times 10^{-5}$

The sputtering ratios in Table 17 are upper limits. These upper limits are relatively low and are several orders of magnitude lower than results reported by a group at the Argonne National Laboratory. Confirmation of these results by further measurements of very low sputtering ratios would indicate that neutron sputtering will not present a major problem in first wall erosion of future CTR's.

Extensive measurements of radioactive recoil atoms resulting from (D,T) neutron bombardment of CTR metals, have been made. Initial results are presented in Figure 11 for several metals including some refractories which are candidates for first wall construction.

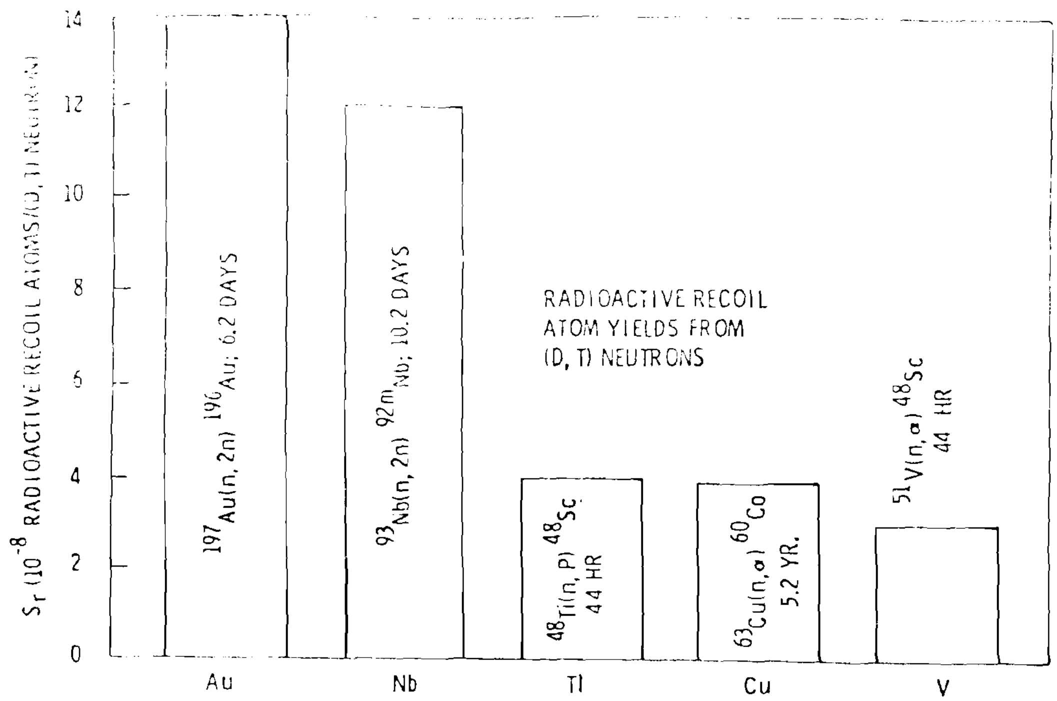

FIGURE 11. Radioactive Atom Yields from Nuclear Recoils, in the Forward Direction, Resulting from a (D,T) Neutron Current on Various Metal Foils

Major radioactivity problems can be expected to occur in the cooling ducts and heat exchangers of fusion reactors due to radioactive material ejected from surfaces under neutron bombardment. For example: in the UWMAK-1 design if the first wall structure is made of Nb, the equilibrium radioactivity from ^{92m}Nb ejected from the cooling duct surfaces is expected to be ~5000 curies for a 1 Mw/m^2 wall loading. Results of the type shown in Figure 11 will provide important input in the design of future CTR's.

Structural Damage Characterization

JL Brimhall, ER Bradley, EP Simonen, HE Kissinger

This work characterizes radiation induced structural damage in potential CTR first wall materials. The investigation is specifically directed at determining the nature of displacement damage at high doses by heavy ion bombardment. Currently, the effect of parameters uniquely associated with ion bombardment are being evaluated to more thoroughly understand the ion simulation technique and make better correlation with neutron irradiation.

SURFACE EFFECTS IN ION BOMBARDED MOLYBDENUM

High purity molybdenum discs have been bombarded at 1000±25°C with 5 MeV Ni^{++} ions at displacement rates of 3×10^{-4}, 1.8×10^{-3}, and 8×10^{-3} dpa/sec, and the microstructures have been examined at the bombarded surfaces by transmission electron microscopy. Voids were observed in the thinnest regions of all three foils. The width of the denuded region in the specimen bombarded at the lowest dose rate was determined by stereo analysis to be less than 100 Å. The width of the denuded zone should be less in the specimens bombarded at the higher dose rates since there is an inverse relationship between dose rate and the width of the denuded zones.

The dose rate dependence of the average size of the voids observed at the surface of the specimens is given in Table 18. The decrease in size with increasing dose rate can be attributed to the dose rate effect and its corresponding temperature shift which is calculated to be approximately 100°C between the high and low dose rates. The ~30 Å difference in size agrees reasonably well with the ~25 Å difference reported for neutron irradiated molybdenum.[10]

TABLE 18. Void Size as a Function of Dose Rate for High Purity Molbydenum, Bombarded with 5 MeV Ni^{++} Ions at a Temperature of 1000° ± 25°C.

Dose Rate (dpa/sec)	Dose (dpa)	Average Void Diameter (Å)
At Bombarded Surface		
3×10^{-4}	6	56±3
1.8×10^{-3}	6	41±3
8×10^{-3}	6	≤30±3
∿3000 Å Below Bombarded Surface		
4×10^{-4}	8	51±3
2.4×19^{-3}	8	45±3
1×10^{-2}	8	34±3

The dose rate dependence of the average size of the voids observed in specimens with ∿3000 Å removed from the bombarded surface is given in Table 18. Comparison between these data and data taken at the surface indicate that the void size increased in specimens bombarded at the higher two dose rates and decreased in the specimen bombarded at the lowest dose rate. An increase in void size is expected due to the increased displacement damage below the surface. The decrease in size observed in the low dose rate specimen may signify a surface effect where void growth is enhanced near a free surface.

The preliminary data show that in ion bombardment studies using high dose rates, near-surface effects such as void denuding are not significant. At low dose rates, surface effects may have to be considered.

BCC ION CORRELATION EXPERIMENT

An interlaboratory ion correlation experiment has been developed as a result of a meeting at Battelle Seattle Research Center. The goal of the experiment is to make an interlaboratory comparison of the microstructure in an identical material by various ion irradiations at different laboratories. Those laboratories participating in the experiment will be Pacific Northwest Laboratory, University of Wisconsin, Massachusetts Institute of Technology, Atomics International, Naval Research Laboratory and Argonne National Laboratory/University of Cincinnati. In the experiment, molybdenum from a common source will be distributed to the laboratories for ion bombardment and analysis of the microstructure. All of the laboratories will carry out the experiments using experimental parameters mutually agreed upon. The principal conditions are an irradiation temperature of 900°C, a dose of 20 dpa and a dose rate of $2x10^{-3}$ dpa/sec. The plans are to compare and correlate the results from the experiments when they become available. It is expected that results from such an interlaboratory comparison would allow conclusions from various laboratories to have common bases.

EFFECT OF HELIUM ON VOID FORMATION IN MOLYBDENUM*

Helium ions have been injected simultaneously with nickel ions into molybdenum to determine the effect of helium on void nucleation. Simultaneous injection of helium should provide a better simulation of actual reactor conditions than preinjection. At a temperature of 900°C and a displacement dose rate of $1.4x10^{-2}$ dpa/sec, there was no difference in void size between specimens in which the helium was injected simultaneously and those in which there was not helium injected. There was a small effect if the helium was pre-injected into the specimens. Table 19 shows the measured void sizes for the three irradiation conditions. The absence of a strong helium effect under these bombardment conditions is in

*This work supported by Division of Physical Research

agreement with nucleation theory and is related to the very high displacement rates used in the experiments. Experiments are underway using a lower displacement rate to determine if helium effects can be generated. The results show that the dose rate and temperature will be critical factors in determining the magnitude of the helium effect on void nucleation.

TABLE 19. Void Size in Ion Bombarded Molybdenum, Bombarded Under Different Conditions of Helium Injection

Bombardment Condition	Dose (dpa)	Void Size (Å)
No Helium	6 53	38±3 54±3
He Preinjected Prior to Ni ion Bombardment	6 53	28±3 45±3
He Simultaneously Injected During Ni ion Bombardment	6 53	41±3 51±3

Ion Beam Simulation of Fast Neutron Damage for Mechanical Property Measurements

RH Jones, DE Styris, OK Harling, RP Marshall

Present neutron sources cannot provide the neutron energy spectrum and fluences which will be present in future fusion reactors. Even at lower energies a CTR fluence of $3x10^{23}$n/cm2 would require five to ten years in the experimental reactor EBR-II. Charged particle bombardment can simulate the fusion neutron damage levels at rates in excess of existing fission reactor irradiation facilities. Ion beam measurements are relatively easy to instrument as compared to in-reactor measurements. High levels of radioactivity are avoided with in-beam irradiation. These advantages of ion-beam simulation are expected to allow measurement to be made with greater sophistication, speed, ease, and therefore, less cost than is possible with reactors. Studies have been made to determine the effect of CTR relevant neutrons on the mechanical properties of candidate CTR materials by bombardment in other particle beams.

The extent to which penetrating ions simulate neutron damage to mechanical properties in metals is being studied for application to candidate CTR alloys for determining the damage-property relationships in the fusion reactor environment. Depending on the degree of simulation attainable, ion beam simulation of fast neutron damage will prove useful in one or all of the following categories:

- Qualitative, where mechanical tests during or after ion-bombardment would indicate the relative magnitude of damage and permit valid comparisons of chemistry, structure, or metal system parameters. These data would not be absolutely predictive of CTR neutron damage.
- Phenomenological, where property tests in the particle beams would be used to investigate rate, temperature, time effects on deformation behavior that are difficult or costly to pursue in a neutron environment.
- Quantitative, where the results on ion-beam tests can be made predictive of mechanical behavior in CTR engineering devices.

Identifying the critical experimental problems, program planning, establishing a committee of technical advisors, visiting U.S. and European CTR design teams, reviewing the literature on micro-specimen preparation, testing and deformation modes, and ion damage in candidate metals are areas which are receiving attention in the early stages of this program. Figure 12 illustrates the data input and major tasks required. At this time, Spence Bush of PNL, E. R. Parker of University of California at Berkeley, Arden Bement from MIT, and Gerald L. Kulcinski of the University of Wisconsin have joined our technical advisory committee. Visits to Lawrence Livermore Laboratories, University of Wisconsin, Princeton Plasma Physics Lab., MIT, Harwell, and Russian experimental fusion reactor groups have been made. Arrangements are being made for close cooperation between PNL and other laboratories involved in related radiation damage studies.

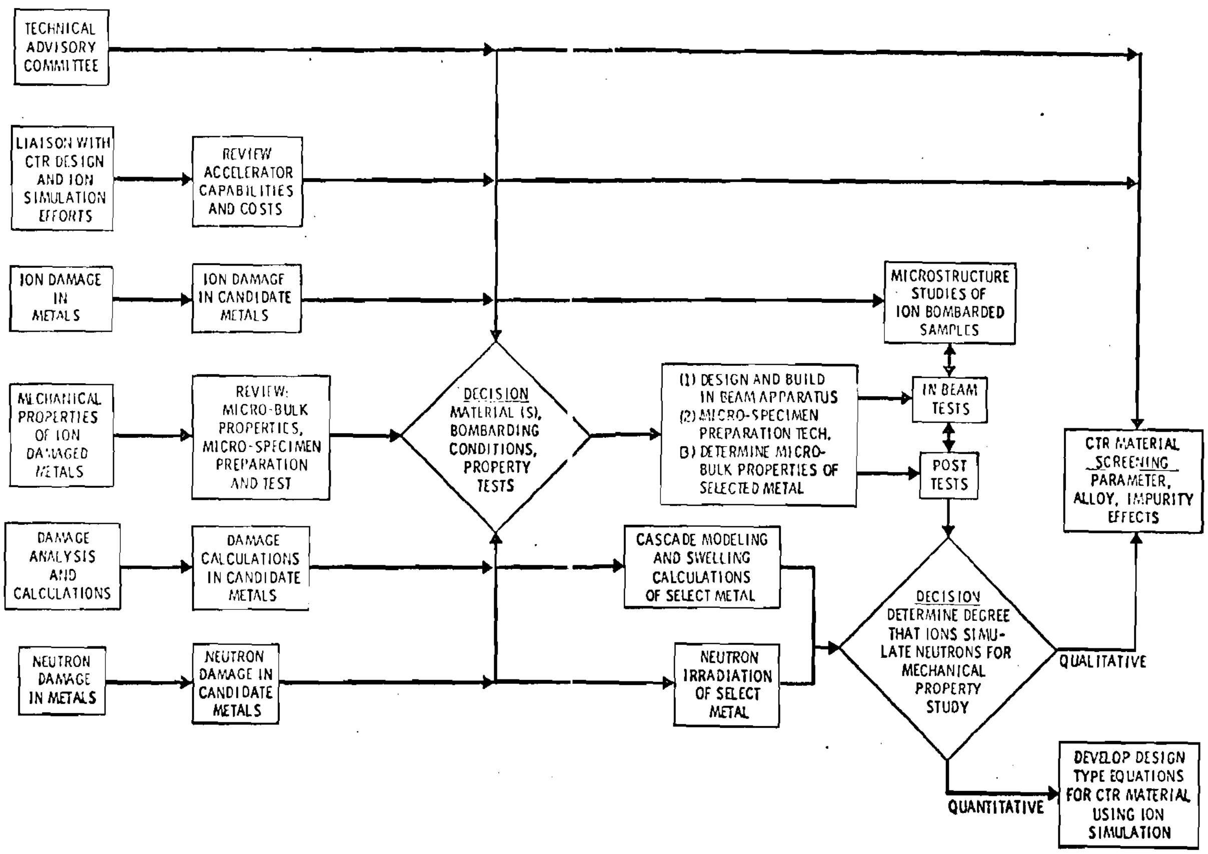

FIGURE 12. Data Input and Major Tasks Required for Program

Some areas requiring concentrated effort in order to realize the program goals are relating the micro-specimen properties to bulk properties, apparatus design for temperature control and strain measurement during in-accelerator testing, the experimental details for relating the properties of neutron and ion irradiated samples and the effect of surface chemistry and defects on properties in high surface/volume ratio specimens.

The value of penetrating ion beams for simulating neutron damage in metals for mechanical property measurements is expected to be assessed within a three to four year period. The application of this technique for alloy selection and development and the formulation of quantitative design equations for CTR purposes is a longer range goal.

Evaluation of Carbons and Graphites for Controlled Thermonuclear Reactor Applications

LR Bunnell, WJ Gray, WC Morgan, GL Tingey

Many CTR concepts call for graphite in the blanket and/or shield regions. In addition, some recently proposed concepts[6,7,8] have suggested the use of graphite or graphite cloth between the plasma and first structural wall. Graphite temperatures in the latter applications could range up to 2000°C. In some of these new concepts, lithium and/or beryllium bearing compounds would be placed in the graphite regions to improve the tritium breeding ratio. Location of the graphite inside the vacuum wall and adjacent to the plasma requires the assessment of potential problems and increases the importance of others. Specific areas being investigated include:

- Evaluation of radiation damage effects.
- Experimental testing of the correlation between damage production rates in fission reactor spectra and those from higher neutron energies found in a CTR.
- Evaluation of the rate of sputtering of carbon from the surface of the first wall into the plasma. This work is being done in conjunction with other surface science research presented in this report.
- Evaluation of outgassing problems including hydrogen and helium sorption and desorption properties.
- Evaluation of potential for carbon removal and transport due to reactions with hydrogen and the potential for reactions with impurities in gaseous coolants.

Results of literature reviews and other preliminary work in a number of areas follow in summary together with some plans for the future.

Graphite Lifetimes

Conn, et al.[7] have projected graphite cloth lifetimes in a CTR as about two years. For their projection to be valid, the graphite cloth must be assumed to have a radiation stability comparable to solid nuclear grade graphites. In our opinion, the lifetimes of cloths may be much different (either longer or shorter). Temperature must also be considered when projecting lifetimes. Using the atomic displacement rates at the inner surface of the graphite region calculated by Conn, et al.,[7] we have projected the minimum expected lifetimes for solid graphites as a function of temperature, as shown in Figure 13. Four things should be noted:

1) The temperature range 900-1200°C produces the shortest lifetimes.
2) There are insufficient data above about 1300°C to project lifetimes -- however, we believe lifetimes above 1300°C should be equal to or greater than those found below 800°C.

3) Some improved graphites have significantly longer lifetimes than typical nuclear grade graphites but cost as much as 10 times more.
4) The projected lifetimes were somewhat arbitrarily set equal to the time required for the graphite to first shrink and then expand back to its original dimension. Longer lifetimes could be realized if allowances were made for some net expansion.

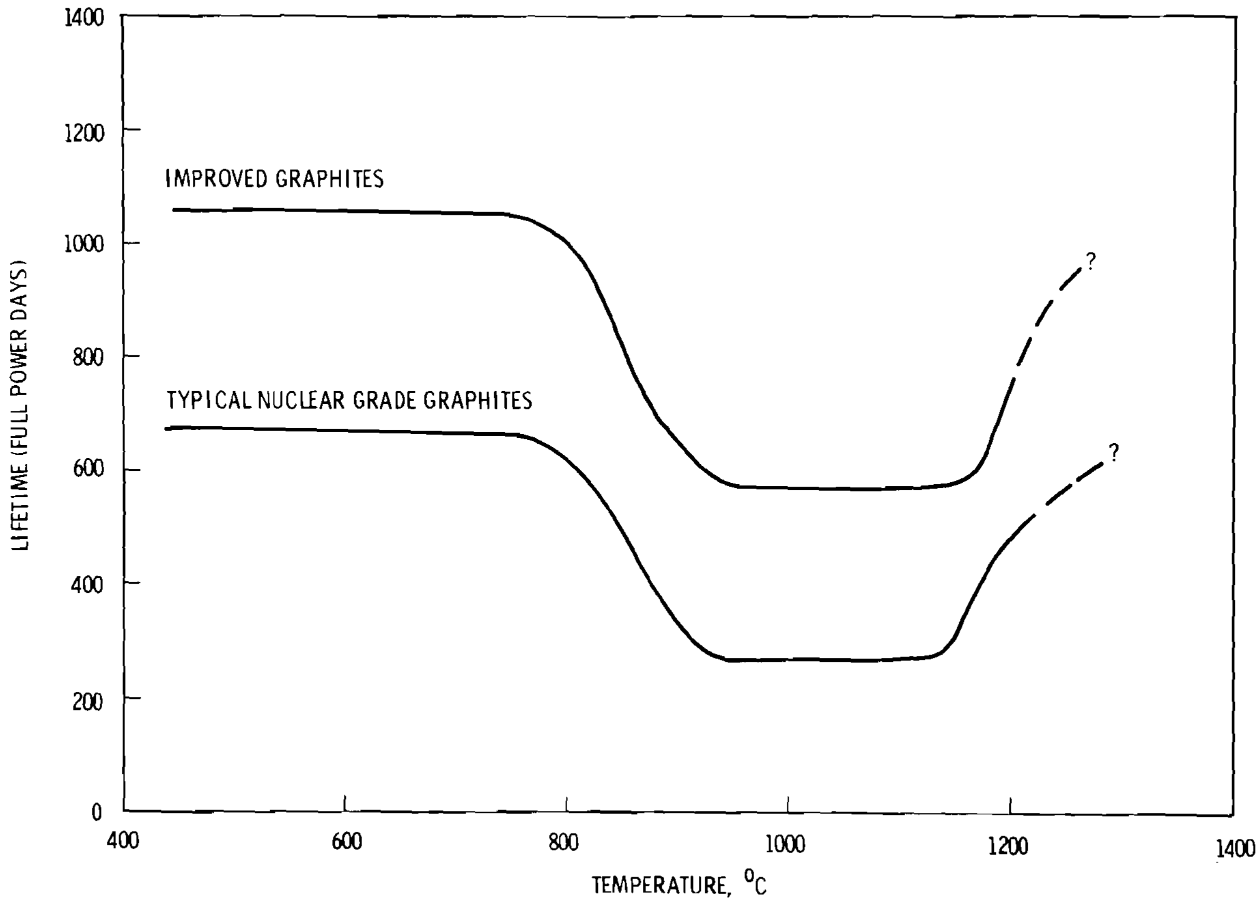

FIGURE 13. Minimum Expected Lifetime of Graphite in First Wall CTR Applications at 1 MW/m^2 Wall Loading

While a great deal is known about radiation effects in solid nuclear grade graphites, almost nothing is known about radiation effects in cloths. Because the use of carbon or graphite cloths is now receiving attention, arrangements are being made to irradiate some of these materials in the EBR-II beginning in the spring of 1975. Fluences approaching $10^{22} n/cm^2$ will be achieved after about one year's irradiation.

Atomic Displacement Rates

Changes in the Young's modulus of stress-annealed pyrolytic graphite can be detected after fluences as low as 10^{12} n/cm^2. PNL plans to irradiate samples of this graphite in various high-energy neutron sources and, by measuring the relative changes in Young's modulus, determine the relative atomic displacement rates in the different neutron spectra. Techniques are currently being developed to prepare this very fragile graphite in the small size required and for measuring the Young's modulus.

Sputtering Measurements

Measurements of carbon sputtering rates should be done on single-phase homogeneous materials such as very highly oriented graphite or very isotropic pyrolytic carbon so as to avoid ambiguities in the event that sputtering rates are dependent on structure and/or orientation. Quantitative determination of the number of sputtered carbon atoms appears difficult at best. Labeling the samples with carbon-14 seemed appropriate but the cost appears prohibitive. An exception to this cost dilemma may be to label glassy carbon using a labeled phenolic resin as a precursor. Glassy carbon is also a very homogeneous, very isotropic material that represents one extreme of a whole spectrum of possible types of carbon and graphite materials. PNL has asked for bids on the labeled phenolic precursor intending to order some of this material. Delivery times are expected to be long, and actual sputtering measurements will probably not begin for several months.

Outgassing

The degassing of graphite is a rather complex problem. Most of the studies to date have been done at fairly high pressures (10^{-3} Torr) and the remainder of the pumpdown is much less clear. In recent high vacuum work, polycrystalline graphite specimens only 0.01 cm thick were used which had previously undergone a 24-hr, 300°C outgassing treatment and the outgassing was found no worse than a tungsten filament.[9] The ease with which the graphite was degassed may be optimistic because of the relative absence of diffusion effects which would be present in large pieces. Water should be desorbed at a temperature that is low enough to prevent decomposition reactions. Recent data show that oxygen thus produced can be very difficult to remove. Our literature search is continuing but it appears that much more experimental data will be required to resolve the outgassing questions for CTR applications.

Hydrocarbon Formation

A study of the potential for carbon removal by reactions with hydrogen requires knowledge of pertinent equilibrium constants such as shown in Figure 14. All carbon located between the plasma and the first structural wall will likely be maintained at a high temperature so that heat transfer will occur primarily by radiation. The high temperature limit is restricted by the sublimation rate of carbon. Thus for the purposes of this analysis, the carbon temperatures were assumed to be restricted to the range of 1000 to 2500°C with maximum H_2 concentration of 10^{-5} atmospheres. This concentration is higher by a factor of at least 100 than the vacuum chamber but was purposely chosen to be high to account for cases where tritium is generated inside the carbon and must diffuse out. Table 20 shows the maximum hydrocarbon concentration which would be in equilibrium with 10^{-5} atm of hydrogen and solid carbon within the temperature range. As observed, C_2H_2 will have the highest concentration (10^{-7} atm) at 2500°C. CH_4 concentrations will never exceed 10^{-12} atm and all other hydrocarbon concentrations will be very much lower.

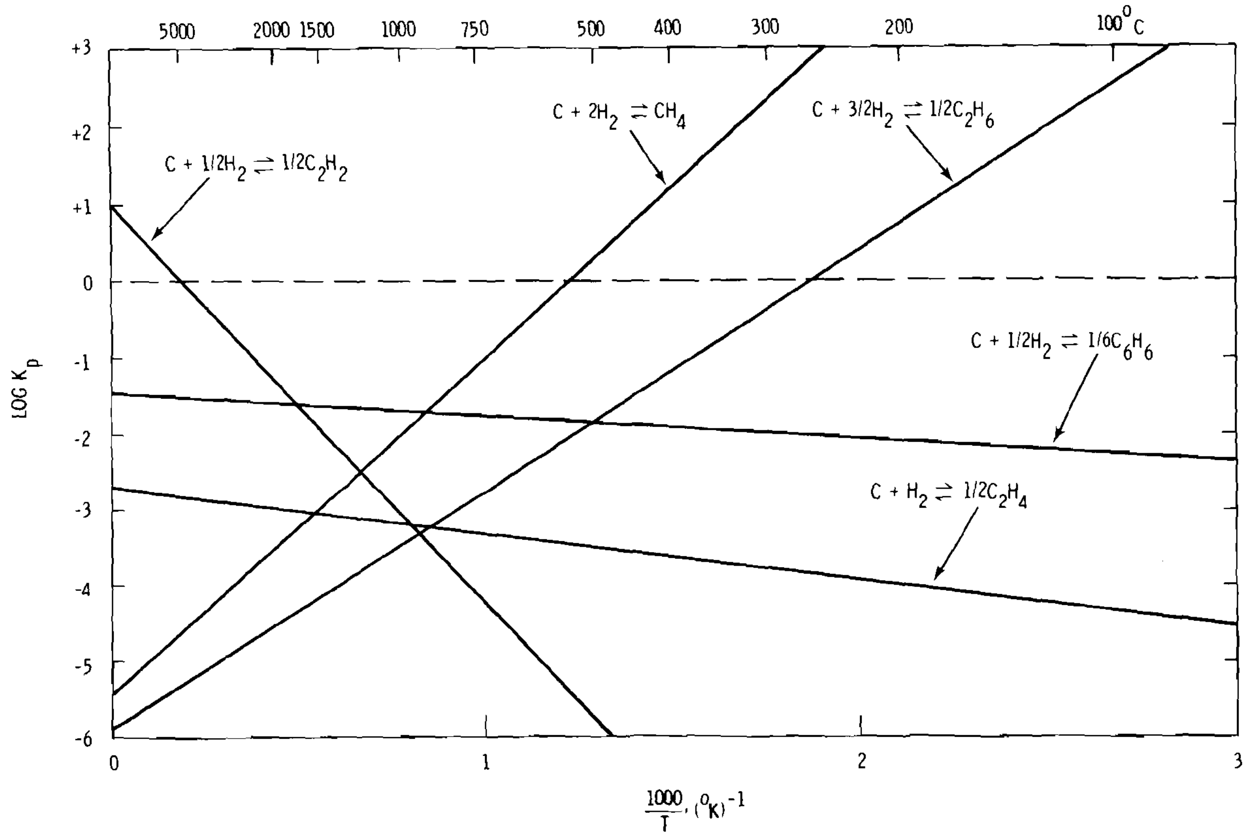

FIGURE 14. Equilibrium Constants for the Formation of Selected Hydrocarbons

The rate of carbon removal is not necessarily controlled by the equilibrium concentrations but rather by the rate of the reaction and by the rate of transport to a sink for the hydrocarbon. The reaction rates are expected to be high at these temperatures and the transport rate will be a direct function of concentration. Thus, it appears that carbon transport will be very slow at the extremely low hydrocarbon concentrations predicted. High energy processes (particles and electromagnetic) will perturb the thermal rates and a further evaluation will be required to define their importance.

TABLE 20. Maximum Hydrocarbon Concentration

Reaction	Maximum Equilibrium Constant	Temperature	Hydrocarbon Concentration At $[H_2]=10^{-5}$ atm
$C+1/2H_2 \rightleftarrows 1/2C_2H_2$	$10^{-0.97}$	2500°C	10^{-7} atm
$C+2H_2 \rightleftarrows CH_4$	$10^{-1.95}$	1000°C	10^{-12} atm
$C+3/2H_2 \rightleftarrows 1/2C_2H_6$	$10^{-3.38}$	1000°C	10^{-22} atm
$C+1/2H_2 \rightleftarrows 1/6C_6H_6$	$10^{-1.53}$	2500°C	10^{-24} atm
$C+H_2 \rightleftarrows 1/2C_2H_4$	$10^{-2.90}$	2500°C	10^{-16} atm

*Radiation Effects in Aluminum in Relation to Fusion Reactor Design**

HE Kissinger

The literature dealing with irradiation effects in aluminum cannot compare in volume with that for copper, nickel, or the noble metals nor approach the tremendous amount of literature dealing with stainless steel. However, aluminum has been studied in some detail and since the metal is important to fusion reactor design a portion of the literature which is considered applicable has been reviewed.

LOW TEMPERATURE IRRADIATION, 4.2°K to 20°K

Experiments conducted at temperatures between 4.2°K and 20°K were generally intended to study the physics of point defects. General references[11] and symposia[12] exist which summarize the bulk of this work. Samples usually received quite moderate fluences in cryogenic environments. Electrical resistivity was a common technique with resistivity changes on annealing providing data from which the defect properties could be derived.

In comparison to other metals studied, aluminum has a low displacement energy and a low vacancy formation energy. These factors combined with the low atomic weight yield a high damage rate approximately three times that for copper.[13]

VOIDS

Irradiation of most metals at appropriate temperatures may result in the formation of voids. Voids produce a volume change, or swelling, which may be as much as 10% or more. The mechanisms of void nucleation and growth are still unclear; a comprehensive review summarizes the available data and theoretical work.[14]

Aluminum is one of the few metals in which voids are formed during room-temperature irradiation. Voids form most readily (i.e., at lower fluences) in highly pure Al but less readily in impure Al and certain solid-solution alloys. Precipitation-hardened alloys such as the 6000 series are most resistant to void formation. A study found the fluence to produce a given density change was 2 orders of magnitude higher for 6061 Al than for a high-purity sample.[15]

The upper temperature limit for void formation in Al has been reported to be between 220° and 250°C and that cold working the sample before irradiation (at 55°C) appeared to increase the swelling.[16] In most metals, cold-work inhibits swelling. The influence of grain boundaries was demonstrated in a series of irradiations at 125°C and 150°C.[17] Regions near grain boundaries contain few voids, a phenomenon called "grain boundary denuding." The width of the denuded zone increases with increasing irradiation temperature.

*Work sponsored by the Division of Physical Research

The denuded zone is not free of voids; usually a band of very large voids is formed on or very near the grain boundary itself. At temperatures approaching the upper limit for void formation the denuded zone may extend entirely across the grain leaving only the voids in the near-grain-boundary region.[18]

It is anticipated that a fusion reactor with aluminum components would operate at temperatures of the order of 200° to 300°C. Voids and void-induced swelling are not expected to be a serious problem for irradiation temperature above 200°C. Dimensional changes would probably arise from the accumulation of transmutation products.

TRANSMUTATIONS

A thermal neutron reaction:

$$_{13}Al^{27}\ (n,\gamma)_{13}Al^{28} \xrightarrow{\beta} {}_{14}Si^{28}$$

generates a silicon impurity which can affect the mechanical properties of aluminum.[19] A fluence of 9×10^{22} n/cm^2 (thermal) produced 1.5 wt% Si in a 6063 Al alloy.[20] In experiments using a different neutron spectrum, 1.6 wt% Si was found in 1100 Al after 6.5×10^{22} n/cm^2 (thermal).[21]

Fast neutron reactions generate hydrogen and helium by the reactions

$$_{13}Al^{27}\ (n,p)\ {}_{12}Mg^{27} \xrightarrow{\beta} {}_{13}Al^{27}$$

$$_{13}Al^{27}\ (n,\alpha)\ {}_{11}Na^{24} \xrightarrow{\beta} {}_{12}mg^{24}$$

Analyzed samples of irradiated Al consistently show a greater hydrogen content than can be accounted for by transmutation.[20] The additional hydrogen is probably injected by proton recoils in the cooling water.

Since silicon has a different density than aluminum, the density of the irradiated Al changes with Si content. Helium and hydrogen may influence void nucleation,[17] or at a critical temperature, (>300°C) form gas-filled bubbles.[22] Some swelling must therefore be expected even if void formation can be suppressed.

These transmutation products also have an effect on the mechanical properties. The suggestion has been made that a solid solution of Mg in Al may precipitate the Si as Mg_2Si on a fine scale, thus preventing deleterious effects.[23] Since the 6000 alloy series is hardened by dispersed Mg_2Si precipitates, the suggestion appears feasible.

SURFACE DAMAGE

Surface damage effects on metals is being explored rather carefully at the present time. Unfortunately, the emphasis has been almost entirely on refractory metals. Some of the general conclusions, however, may be extrapolated to Al.

A general equation for wall erosion has been given:[24]

$$\frac{\Delta \ell}{t} = \sum_{\mu} S_{\mu} \quad \phi_{\mu} \quad \frac{A_w}{N_o \rho}$$

Where $\Delta\ell$ is wall thickness removed, S_{μ} is the number of atoms removed by a single bombarding particle; μ, ϕ_{μ} is the flux of particles; μ, N_o is Avogadro's number and A_w and ρ are respectively atomic weight and density of the wall material. All other factors being equal, the rate depends upon the ratio A_w/ρ. For aluminum this ratio is approximately the same as for the refractory metals. The yields S_{μ} are unknown for Al but are of the order of 10^{-3} atom/incident particle (neutron and deuteron) for most metals studied to date. Alpha particle bombardment not only generates a sputtering yield but can produce "blisters" of helium gas just beneath the surface. A "chemical" or "reactive" sputtering process also may occur, where an impinging molecular species may react with the bombarded surface to produce a volatile compound. The role of the Al oxide layer on these processes is not known, although it should tend to inhibit the sputtering. A detailed review of these phenomena can be found in Reference 25.

In Al, 200°C is approximately 0.5 of the absolute melting point. The corresponding temperature for niobium is 1100°C. One review presents a number of estimates of niobium wall erosion, as much as 1-2 mm per year. An aluminum wall may reasonably be expected to erode at an equivalent rate.

MECHANICAL PROPERTIES

Early reactor designers assumed that radiation damage to aluminum would be negligible or non-existent. Aluminum components in research reactors provided satisfactory service for many years, apparently justifying the designer's assumptions. In the late 1960's, however, two failures of aluminum assemblies were reported. An alloy tube failed in HFIR because of embrittlement[26] and an ORR sample tray became unserviceable because of dimensional instability.[27] Subsequently components of MTR[21] and other reactors[20,23] were examined to empirically evaluate the serviceability of aluminum and aluminum alloy reactor subassemblies. Basic studies have been performed at Oak Ridge[28] and Saclay.[16]

The mechanical properties of 1100 Al, which is nearly pure aluminum with small additions of copper and zinc, have been investigated.[21] Exposure to 4.5×10^{22} n/cm^2 (E>1 MeV) increased the yield strength approximately 4-fold, from 5.0 ksi to 18.7 ksi when measured at 20°C. Measurements at 300°C showed comparable results. A loss of ductility was observed that became disproportionately severe during tests at 200°C and above. Electron microscopy showed no significant changes in microstructure, so the decreased ductility was ascribed to a redistribution of transmutation products, probably silicon. Helium and hydrogen were also present in quantity (5 ppm He, 26 ppm H) and may have had an effect.

Examination was made of a 6063 Al alloy which had been irradiated to 5.8×10^{21} n/cm fast fluence in Savannah River reactors.[20] An increase in yield strength and a decrease in ductility was observed. Tests were performed to 300°C where the alloy essentially recovered its pre-irradiation properties. Scanning electron microscopy showed the failure mechanism was intergranular fracture in low ductility samples. These investigations also indicated that migration of transmuted silicon to grain boundaries was responsible for the reduced ductility. According to one study, the helium effects should be most severe at elevated temperature.[29] A series of elevated-temperature creep tests on alloys irradiated in HFIR failed to reveal any effect which could be unequivocally associated with helium.[23] The extent of helium embrittlement in Al and Al alloys remains undetermined.

In few of the experiments described was the ductility loss severe enough to render the alloy unserviceable where irradiation temperatures were in the general range of 50° to 150°C. Higher-temperature effects mentioned refer to post-irradiation anneals. Evaluation of aluminum for fusion application requires higher temperature irradiations to higher fluences. The observation that damage appears to recover at 300°C suggests that irradiation near this temperature would not generate damage but this needs experimental verification since the mechanical properties of aluminum alloys deteriorate rapidly as the temperature increases.

CONCLUSIONS

Aluminum has a low displacement energy and a low vacancy formation energy. These factors combined with low atomic weight yield a high damage rate approximately three times that for copper. Voids and void-induced swelling are not expected to be a serious problem for irradiation temperature much above 200°C. Dimensional changes, should they occur, would probably arise from the accumulation of transmutation products. Neutron reactions generate silicon, hydrogen, and helium in the aluminum. These impurities are expected to cause swelling even if void formation can be suppressed. None of the experiments reviewed succeeded in relating the presence of voids to any change in mechanical properties. Helium, however, has long been recognized as a source of embrittlement so a similar effect could be expected in aluminum. Surface damage must be extrapolated from data for refractory metals since adequate data does not exist for aluminum. Evaluation of aluminum for fusion design applications requires higher temperature and fluence irradiations than have been conducted so far.

Helium Compatibility with Fusion Reactor Materials

DG Atteridge, AB Johnson, Jr. and RE Westerman

Pressurized helium has several advantages for fusion reactor coolant applications: it is not activated by the neutron flux; it does not interact with magnetic fields; it does not transport large volumes of radioactive corrosion products through solubility effects; and it is chemically inert. However, the best attainable helium purity is not good enough to entirely prevent corrosion reactions with reactive metals at elevated temperatures. In helium-cooled fusion systems there are three potential sources of impurities:

1) Impurities indigenous to the helium. Existing helium purification technology leaves traces of O_2, N_2, H_2, CO_x, and hydrocarbons in the purified gas.
2) Tritium and associated impurities. Tritium will almost certainly be found in the helium used to cool the breeding blanket. Some conceptual designs propose purposeful O_2 additions to the helium to convert the T_2 to T_2O, permitting the removal of the tritium by molecular sieves. The range of tolerable concentrations for the $T_2/O_2/T_2O$ system needs to be defined.
3) In-leakage. In a helium system which interfaces with steam, corrosion product hydrogen will diffuse through the heat exchanger. Also, unless major advances in heat exchanger technology occur, some leaks can be expected, allowing ingress of H_2O to the helium coolant. Other sources of in-leakage to the primary helium will undoubtedly occur, but are definable only by detailed analysis of specific designs.

Candidate materials for fusion reactor first wall and blanket structures include aluminum-base, iron-base, and nickel-base alloys, and refractory metals. The FY 1975 program will concentrate on iron-base and nickel-base materials, which are supported by an existing fabrication technology and therefore are prime candidates for first generation CTRs. Aluminum alloys also represent existing technology, but will not tolerate the temperature regime (600-700°C) planned for testing the iron-base and nickel-base materials. The refractory metals offer the most severe challenge to helium coolant applications, requiring that they be phased into the program.

Helium test equipment available for the studies includes:

- A large gas loop, which recirculates up to 500 lb/hr of helium at 300 psi and operates up to 2100°F.
- A small gas loop capable of operation up to 350 psi and 2700°F at helium flow rates up to 600 ft/sec.

- Two gas autoclaves designed for operation up to 350 psi at 2500°F.
- A vacuum facility used for studies of metal reactions in contaminated helium. This system includes a semi-micro automatic recording balance to continuously monitor specimen weight changes.

The literature on helium coolant compatibility relevant to fusion reactors was reviewed.[30] Further insight to helium coolant compatibility in stainless steel CTRs was gained when as part of this review by assessing helium coolant compatibility in UWMAK-II[31] (a helium-cooled Tokamak being designed at the University of Wisconson).

Based on these studies the experimental program was designed. The program includes five materials: 316, 321 and 347 austenitic stainless steels, Incaloy-800 (high-Ni), and 2 1/4 Cr - 1 Mo (Fe-base), representing candidate first wall and heat exchanger materials. The test specimens include both smooth and notched tensile specimens. The specimens will be exposed in two autoclaves to helium at 300 psi and 650°C. One system will be contaminated with $\sim$50 ppm H_2O, the other with $\sim$50 ppm O_2.

The tests will provide information on the corrosion interactions and changes in mechanical properties of the specimens.

The major activities in the program have been procurement of specimen materials, specimen fabrication and recommissioning of helium test equipment and analytical instruments used on previous PNL helium technology programs.

Helium Embrittlement in CTR Material Employing the Tritium Trick Method

AB Johnson, Jr., JF Remark, RE Westerman

Helium embrittlement is one of the radiation-induced damage mechanisms which may limit the life of CTR first wall materials. Helium generation rates will be higher in fusion reactor first wall materials than in fission reactor fuel cladding due to the higher n,α cross section of materials exposed to the 14 Mev fusion neutrons. Because helium accumulates slowly in most materials exposed to fission fluxes, other methods of helium injection need to be investigated. Helium injected by accelerators has a limited range in materials, placing limitations on mechanical property specimen design. The so-called tritium trick can be employed to inject ^3He, generated from the tritium decay, into alloys having substantial hydrogen solubilities.[32] The advantage of this method is that it can be used to prepare mechanical property specimens of standard size to assess helium damage effects in metals, apart from other radiation damage effects. The primary objective of the present study is to determine the applicability of the tritium trick method to the study of helium effects on the mechanical properties of niobium and vanadium alloys.

Commercial purity niobium and vanadium rod stock was obtained for this study. The initial program emphasis was placed on niobium. Twenty-five miniature buttonhead tensile specimens were machined from the niobium rod stock. Fifteen tensile specimens were prepared from the annealed material and ten prepared from material cold worked 25-35%. Six annealed specimens and four cold worked specimens were charged with tritium. Control specimens were of two types: as-received and deuterium-charged. Selected specimens were charged with deuterium to determine what effect, if any, the charging and discharging cycles had on the mechanical properties. The niobium specimens were charged in the following manner: the annealed tensile specimens were pre-annealed at 1050°C for 30-min at 10^{-5} torr; the tritium (4000-12,000 appm dissolved in the metal) was charged into the tensile specimen for 90-180-min at 500°C; the tritium was allowed to decay to ^3He; and in the final operation tritium was outgassed at 800°C. The specimens were given a pretest anneal at 1020°C for 30 min at 10^{-5} torr before being tensile-tested at 1020°C in a vacuum of 10^{-5} torr at a strain rate of 0.016-min^{-1}. The procedure for cold worked specimens was similar, except they were not subjected to the 1050°C preanneal.

Table 21 is a summary of the high temperature mechanical properties of the niobium and vanadium tensile specimens. Column 3 denotes the calculated helium content of the tensile specimens, while the experimentally determined values are listed in column 4. The high temperature mechanical properties listed show no significant change in the total elongation between the controls and the helium-charged specimens. Helium embrittlement did not occur for the three types of specimens examined. The yield strength and the ultimate tensile strength data suggest possible slight increases with increasing helium concentration. Information regarding helium and deutrium charging of vanadium was previously reported.[33]

TABLE 21. Summary of He-Charged Niobium (Nb) and Vanadium (V) Tensile Specimens

Material and Specimen Number	Metallurgical Condition	Helium Content, appm(a)	Helium(c) Analyses	High Temperature Mechanical Properties: Yield Strength, psi (0.2% offset)	Ultimate Tensile Strength, psi	Total Elongation, %
Nb-AN 3	Annealed	30		8,080	9,470	45.6
Nb-AN 6	Annealed	65	65,60	12,550	13,880	47.7
Nb-AN 10	Annealed	As-Received Control - 0(b)		7,040	8,900	48.8
Nb-CN 2	Cold worked	62	52,56	25,300	25,700	26.4
Nb-CN 3	Cold worked	30		19,500	19,700	17.6
Nb-CN 10	Cold worked	Control - 0		19,600	19,900	21.1
V-V 1	Annealed	Control - 0		9,380	13,650	42.3
V-V 3	Annealed	166	113,114, 112,113	14,900	18,800	42.3
V-V 4	Annealed	As-Received Control - 0		12,950	16,600	38.1
V-V 3(d)	Annealed	335	223			

(a) Calculated helium content based on the amount of tritium charged into the specimen and the holding (decay) time before outgassing the tritium from a given specimen.

(b) Two kinds of control specimens are tested in addition to those charged with helium. The first is as-received material, as-machined specimen with no test history prior to tensile testing. The second is a specimen having the same thermal and hydrogen isotope charging history as the helium charged specimens. The only exception is that these controls were deuterium-charged rather than tritium-charged.

(c) He analyses performed by Harry Farrar, Atomics International.

(d) Specimen cut from the buttonhead of V-V3 and allowed to continue to accumulate helium until helium analysis was performed.

Transmission Electron Microscopy (TEM) was used to determine the location of the helium bubbles in the metal lattice. Other investigators have indicated that helium bubbles located at grain boundaries are the primary cause of high temperature embrittlement.[34] TEM was performed on vanadium and niobium samples annealed at 900, 1200, and 1600°C. These samples contained helium concentrations of 110 and 60 appm respectively. No bubbles were observed in either specimen after a 900°C anneal. After the 1200°C (2 hours) anneal bubbles were observed in both specimens; however, only vanadium had bubbles at the grain boundaries. Helium bubbles were observed in the niobium grains at both 1200 and 1600°C but no concentration of bubbles was observed at the grain boundaries.

Helium apparently contributes most to high-temperature embrittlement when it is concentrated at the grain boundaries.[35] Results of the TEM studies indicate that helium migrates to the grain boundaries in vanadium much more readily than in niobium. This finding has encouraged increased emphasis on vanadium studies. Ten vanadium tensile specimens, both annealed and cold worked, have been machined from a portion of the vanadium rod stock. These vanadium samples are being charged with tritium. After the desired helium concentration has been attained, the vanadium specimens will be annealed at 1200°C for one hour prior to the mechanical properties evaluation. TEM will also be used to determine the size, concentration, and location of helium bubbles in the specimens. Sampling of the helium concentration will be performed by mass spectrometry to confirm the calculated concentration of helium that is born within the tritium charged tensile specimens. The helium-charged vanadium tensile specimens may be tested at various strain rates to determine the strain rate effect on the mechanical properties.

Investigations also are continuing on niobium. A niobium tensile specimen having a helium concentration of $\sim$ 200 appm will be annealed at 1800°C for one hour to determine if the helium migrates to the grain boundaries at this temperature. The mechanical properties of the specimen will then be determined.

One annealed and one cold worked specimen now have a concentration of $\sim$ 400 appm helium. They will be committed to mechanical property testing when the 1800°C annealing study has been evaluated.

A study reported a considerable loss of ductility for thin specimens of the alloy V-15 wt% Cr5 wt% Ti after being charged with He by α-particle bombardment to a level of 25 appm.[36] This ductility loss was observed at temperatures above 700°C. PNL is attempting to obtain a quantity of this alloy for charging with helium by the tritium trick method. A comparison can then be made between the effects of: 1) the helium charged by the tritium trick and the helium charged by alpna bombardment; 2) specimen size and configuration; 3) matrix composition.

Since the construction of Table 21, an annealed niobium specimen containing 518 appm (estimated) of helium was tested at 1020°C. It was given a pretest anneal at 1020°C for 1/2 hour. The sample exhibited an increase in strength and a decrease in ductility. Annealed niobium specimens containing approximately 250 and 350 appm are in the process of being examined under the same test conditions.

Electrical Insulators for Controlled Thermonuclear Reaction Application

JE Bates, JE Garnier

Current thermonuclear reactor technology requires the use of electrical insulators in various locations of the Theta-Pinch, Mirror, and Tokamak reactors. These insulators will be subjected to severe irradiation and thermal conditons, i.e, 800 to 1300°C and to 10^{16} nv. These insulators must retain their electrical, thermal, and mechanical properties in intense radiation, electron and particle fields and for long neutron exposures, $\sim 10^{22}$nvt. The objective of this program is to study and evaluate the electrical and related properties of potential electrical insulators for CTR application. The ultimate goal is to find or develop suitable electrical insulators which meet the varied requirements for CTR designs.

A joint Battelle-Columbus and Battelle Northwest review and evaluation of electrical insulators for fusion reactors was completed and published.[37] This review identified the potential electrical insulator needs of the Tokamak, Mirror, and Theta-Pinch reactors. Although the electrical properties of potential insulators are emphasized, irradiation behavior, thermal properties, compatibility, and fabrication were also considered. The report concludes that:

- The electrical properties of electrical insulators are dependent upon structure, impurities, atmosphere, measurement techniques, and fabrication methods.
- Wide variations in the electrical properties are reported for all insulators.
- The data describing the effects of irradiation on the electrical properties are very limited. Neutron damage effects and induced electrical effects from ionizing radiation have received little attention.
- Electrical property data measured during irradiation may be suspect due to experimental error.
- Insulators with isotropic cyrstal structures which also have an open crystal lattice appear to have the best resistance to irradiation damage.

This review is the basis for experimentally measuring the electrical and irradiation properties of insulators. The study will be directed toward the:

- Absolute measurements of the electrical resistivity of some selected electrical insulators.
- Dielectric breakdown measurements beginning with an intensive review and evaluation of breakdown measurement methods.

The electrical resistivity apparatus has been redesigned and assembled. The new three contact-guarded system incorporates methods to eliminate errors previously encountered in measuring insulators with very high resistivity values, such as surface and gas phase conduction.

Measurements have begun on a reference BeO sample and will be extended to include yttrium oxide, alumina-magnesia spinel, aluminum nitride, silicon nitride, and yttrium aluminate.

An evaluation of the possible methods for measuring dielectric breakdown related to CTR insulator applications has been initiated. Breakdown effects are often not intrinsic properties and, therefore, become very dependent on external factors, such a electrode shape and sample thickness. A suitable dielectric breakdown technique will be chosen and developed to provide meaningful data, useful to compare dielectric properties of various insulators which are related to CTR uses. The present work is directed toward an evaluation of the breakdown techniques.

Fabrication Technology

RS Kemper, GS Allison

This study was initiated in July, 1974 with the broad objective of assessing the current and projected availability, producibility, and fabricability of candidate CTR first wall structural materials for identification of major resource or fabrication problems. The early identification of these problems is necessary for the timely development of the supply and/or component fabrication technologies where needed, or redirection of the design effort to circumvent the problems.

Major candidate first wall materials for current CTR conceptual designs comprise 18-8 stainless steel, nickel-base alloys, and alloys of the refractory metals - niobium, molybdenium, vanadium, and possibly tantalum. Material specification, commercial availability and producibility, and fabrication technology have been highly developed for stainless steel and the nickel-base alloys for both industrial and nuclear reactor applications. This is not the case for the refractory metals whose technology was advanced primarily for aerospace applications. Reduction in demand for these refractory metal applications has slowed the growth of their technology in recent years. The current status of development needs to be evaluated in terms of supply and production potential, size limitations, quality, properties, cost and potential fabrication problems. One conceptual design (minimum activation) potentially employs Sintered Aluminum Product (SAP) as a structural material. This material, like the refractory metals, has not been used in recent years and the supply, production, and fabrication need to be evaluated with respect to potential CTR requirements.

The current CTR conceptual designs are being reviewed for potential problems in material supply, component fabrication, component assembly, and component maintenance for the first wall and blanket. The material quantities projected to the year 2020 are being based on a total CTR electrical power generation of 10^6 MWe. This information will be reduced to a summary presentation which will be updated periodically as required. This summary will show for each CTR material considered:

- Current and projected quantity required.
- Current and projected supply.
- Status of fabricability.

The CTR material potential demands are being compiled through continual conference with the Universities and National Laboratories who are engaged in conceptual design studies. The various designs are being studied from a fabrication standpoint. McDonnell Douglas Astronautics - East is preparing a state-of-the-art document for the fabrication of molybdenum, niobium, vanadium and tantalum alloys into CTR type hardware. They are drawing on their extensive experience in fabricating aerospace components of this type material.

The current and projected material supply data are being compiled through conference with Industry, Universities, U.S. Bureau of Mines and U.S. laboratories who have worked with the various materials. Dr. Earl T. Hayes, former Deputy Director - U.S. Bureau of Mines has been retained as a consultant on resource and industrial preparedness problems. Battelle-Columbus Laboratories, through the Metals and Ceramics Information Center are collecting data on resources and industrial capacity as affected by CTR material demands.

*Sputter-Deposited Multifilamentary Superconductors for Fusion Reactors**

SD Dahlgren

Methods have been developed to make multifilamentary Nb-Al-Ge and Nb-Al superconductors by high-rate sputter-deposition. Multifilamentary high-field superconductors are needed for fusion reactor plasma containment. The method permits production of 10^6 or more filaments per square centimeter in a void-free matrix of high conductivity stabilizer such as copper. The bronze method presently used for production of multifilimentary high-field superconductors is limited by 1) the restricted choice of superconductor composition, 2) formation of voids next to the filaments, 3) filament breakage during handling and 4) low conductivity of the stabilizer due to unavoidable solution of a superconductor reactant in the stabilizer.

Excellent superconducting properties have already been found for sputter-deposited Nb-Al-Ge and they can be duplicated in the multifilamentary structure. Sputter-deposited Nb-Al-Ge, for example, was found to have current capabilities of 4.4×10^5 A/cm^2 at 100 kOe, 1.3×10^5 A/cm^2 at 150 kOe, and 2.6×10^4 A/cm^2 at 200 kOe. The high current capacity was attributed to small grain size; grains 350Å in diameter were observed in transmission electron micrographs. Critical temperatures for sputter-deposited Nb-Al-Ge samples were 16°K to 18.5°K. The fabrication methods are sufficiently general that small grain size, and consequently high current capacity, should be producible in multifilamentary superconductors made from any of the promising A-15 phase compositions, e.g., Nb-Al, Nb-Sn or V-Ga.

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

Sputter-Deposition of Materials for CTR Application

R Wang, SD Dahlgren, N Laegreid, RW Moss

The objective of this program is to develop sputtered insulator coatings that meet fusion reactor requirements. Sputtered coatings of aluminum oxide, yttrium oxide and aluminum nitride were prepared for studies of the materials and the technology. A 14 mil thick deposit of Al_2O_3 was made at a rate of about 1 mil/hr, establishing that insulator material coatings can be made in the required thickness by high-rate rf sputter deposition. This thickness is sufficient for the theta-pinch fusion reactor, which requires an insulating first wall coating of about 13 mils. A high-purity sintered Al_2O_3 target was used to make the 14 mil thick deposit at 20°C. Breakage of high-purity sintered targets prevented earlier efforts to deposit insulating materials of the desired thickness. Therefore, the deposit was made in a sputtering system that was redesigned to allow target breakage to occur in a controlled manner to permit satisfactory deposition of the thick deposits.

Previously, a dc triode sputtering system, modified for rf experiments, was used to make thin, 0.4 mil and 6 mil thick Al_2O_3 deposits from a low-purity plasma-sprayed target at a deposition rate of 0.5 mil/hr. Specifically, three 6 mil thick deposits were made on 1 inch niobium disks maintained at approximately 20°C, 600°C, and 800°C. The 0.4 mil deposit was made at 20°C on copper.

Except for unexplained dark areas, deposits made from both sintered and plasma-sprayed targets were transparent and had a slight brownish tint. The brown tint indicated oxygen deficiency, even though oxygen additions were made during the sputtering run.

The deposits made at 20°C yielded an x-ray diffraction pattern characteristic of a glassy or amorphous structure. Two extremely broad diffraction lines corresponding to the most intense lines for gamma Al_2O_3 were observed in the patterns. Heat treatment of the 6 mil thick deposit at 800°C for 17 hours caused partial transformation to crystalline gamma alumina. Complete transformation to gamma alumina was observed after 1 hour at 900°C for the 0.4 mil thick deposit. Further heat treatment of these deposits for 7 days at 1000°C produced gamma Al_2O_3 crystal structures having a small grain size. Heat treatment at 900°C and 1000°C in air caused both deposits to lose much of the brown color that had been ascribed to oxygen deficiency. Finally, after 1 hour at 1420°C, the deposits transformed to white, opaque alpha alumina with no crystallographic texture characteristic of many deposited and heat treated materials.

The 6 mil thick deposits made at 600°C and 800°C also had the gamma alumina structure, but the diffraction lines were somewhat broad, indicating fine grain size or crystalline disorder. Crystallographic texture was not indicated for any of the gamma alumina. A possible advantage of gamma alumina, which is cubic, is that it may permit resistance to irradiation damage caused by anisotropic swelling. Literature data indicate gamma alumina slowly transforms to alpha alumina at 1000°C, but no evidence of the alpha phase was detected in sputter deposits heat treated for 1 week at 1000°C.

Experimental work on deposition of AlN was conducted in modified rf equipment using a sintered AlN target. The purpose was to find whether insulators of nitrides, such as aluminum nitride coatings, can be made by high-rate sputtering. The deposit had a thickness of 2 mils (50 μm) and had a color similar to the target material. The surface of the deposit was rough which may have been caused by the flaking and discharging during the final stage of the deposition. X-ray diffraction showed that the deposit was single-phase AlN. Both the color and the crystal structure of the deposit indicate that the composition is near stoichiometry. AlN coating on a Nb substrate also was prepared by high-rate reactive sputtering of pure aluminum in a nitrogen atmosphere. The coating was about 1.8 mils (45 μm) thick consisting of a 0.6 mil (15 μm) thick Al-AlN graded layer. This coating adhered excellently to the Nb substrate.

Major improvements of the sputtering technique are underway. The anode is being redesigned to prevent it from being insulated by build-up of the insulator coating. Target designs and fabrication methods are being developed to eliminate the target cracking problem. These changes should ensure the consistent quality of our sputtered insulator coatings.

Coatings prepared by the improved sputtering designs will be tested for electronic properties by the staff members at BNL, LASL and PNL within the next few months.

ENVIRONMENTAL EFFECTS

It is important that any impacts of the CTR be identified early in the research and development program so efforts can be modified to provide needed input to help assure minimization of the impacts. A study has been completed to determine the availability of the information needed for writing an environmental impact assessment for the CTR development program. As a result of that study, it was determined that sufficient information is available to permit writing an assessment of the impacts. Therefore, an environmental analysis of the development program has been initiated. As a result of the analysis, a draft document is expected to be prepared that estimates the environmental effects of CTR power plants early in the 21st century. The primary activities to date have been organization of the effort and initiation of the technical analysis. A synoptic outline of the final report has been prepared and appropriate sections have been assigned to a work team of 20 principal investigators.

Since extremely strong magnetic fields will be used in fusion reactors, consideration must be given to the possible consequences of exposure of people, other animals, or plant material to these fields. A review is in progress to determine what information is available concerning the biological effects of magnetic fields and to make recommendations concerning the need for additional data. Information obtained thus far indicate that biological effects occur under certain conditions but that there is a great deal of inconsistencies between experiments.

Information Requirements for Preparation of an Environmental Impact Assessment of the CTR Development Program

JR Young

A study was made to determine the availability of information needed for writing an environmental impact assessment for the CTR development program. It was concluded that enough information is available to permit writing an assessment.

The primary activities in the study were as follows:

- Describe the CTRs being developed.
- Determine the potential environmental interactions for each of these CTR concepts.
- Determine the unique environmental interactions in comparison to fission and fossil power plants.
- Determine the information needed to define those unique environmental interactions.
- Determine the availability of that information.
- Determine the development efforts necessary to provide the information that is not available.

The analysis identified five potential unique sources of environmental interactions:

1) Large tritium releases.
2) Large lithium releases.
3) Large magnetic fields.
4) Large vibrations.
5) Large material requirements.

Sufficient information is available to permit adequate prediction of the metabolic and radiological consequences of tritium releases. The metabolic and radiologic effects of tritium probably are better known than for any other radionuclide. The primary deficiencies in the knowledge of tritium lie in environmental transport and in assurance of no unforeseen long-term effects. Large-scale, multi-generation demonstration experiments are needed to confirm present knowledge of basic principles.

Ample information is available to permit prediction of the general environmental effects of lithium releases. Considerable information is available as a result of past industrial experience. Additional research is needed to assure complete understanding of the effects on both terrestrial and aquatic ecosystems. However, the lack of all detailed information on lithium effects does not preclude ability to prepare an environmental statement at this time. Information is available to permit an order-of-magnitude estimate of the effects as needed for preparation of an initial program environmental statement. Ample time should be available for obtaining the additional information needed for preparation of the environmental statements for the CTR prototypes.

Accurate prediction of the biological effects of chronic exposure to large magnetic fields is not possible at this time. Large scale, multi-generation demonstration experiments are necessary to obtain the adequate information. However, lack of this information does not preclude conducting an environmental analysis. Magnetic fields have relatively small impact regions. Because the fields affecting the general public can be reduced by proper building and equipment design and by use of exclusion zones, the environmental impact on the general public due to magnetic fields is not expected to be significant. Considerable research may be necessary to demonstrate that there will not be significant effects on plant employees.

Vibrations at laser-fusion facilities are expected to have a negligible environmental effect. Reduction of vibrations to a desirable level is possible by proper plant design, although the cost may be high. In addition, reduction of vibrations to an acceptable level for the reactor auxiliary systems, such as the laser, turbine generators, and instrumentation, should result in acceptable vibration levels for the general public.

Procurement of materials for construction of fusion reactors may cause unique environmental effects because of the significant increase in the quantities of materials used. This could result in a large increase in land alteration during mining of ores or storage of chemical wastes. Increased air and water pollution could also result. No additional research is justified until the types and quantities of materials used are known. At that time, additional research may be justified if the environmental effects appear excessive.

Sufficient information appears to be presently available to permit preparation of an environmental statement for the fusion reactor program. Development of the reactors has progressed far enough that the general design and environmental effects of the initial commercial reactors can be estimated. Sufficient information appears to be available to permit estimation of all environmental effects except those due to magnetic forces and large lithium releases. In those two cases the impacts on the general public probably can be reduced to insignificance by use of large exclusion areas and by proper plant design.

A document presenting the results of this analysis is in the final stages of publication.

Controlled Thermonuclear Reactor Environmental Analysis

JR Young

An environmental analysis of the CTR development program has been initiated. The purpose of the analysis is to prepare a draft document that estimates the environmental effects of CTR power plants early in the 21st century after CTRs have been accepted for commercial use. Only an initial program statement can be written because the CTR development program has not progressed far enough for the basic design of the commerical CTR to be known. Each of the CTR design concepts currently being developed will be described and analyzed to determine the general types of interactions with the environment. Then, based on that information, a typical or representative design for the first commerical CTRs will be selected as a basis for illustrating the environmental impacts that may occur at commercial CTR power plants.

Because of the early stage of the development effort, only order-of-magnitude environmental effects can be estimated. A conservative approach will be used. The effects will be estimated by use of current technology. Since commercial CTRs are not expected to operate until about the year 2010, future technical advances should result in lower impacts than estimated in this analysis.

The primary activities to date have been organization of the effort and initiation of the technical analysis. A synoptic outline of the final report has been prepared, and appropriate sections have been assigned to a work team of twenty principal investigators.

The effort is based on a seventeen month schedule with issue of draft report early in December, 1975 (Figure 15). One section of the report, the description of the CTR development program, has been completed. Revisions will occur early in FY 1976, as required by program changes in the meantime.

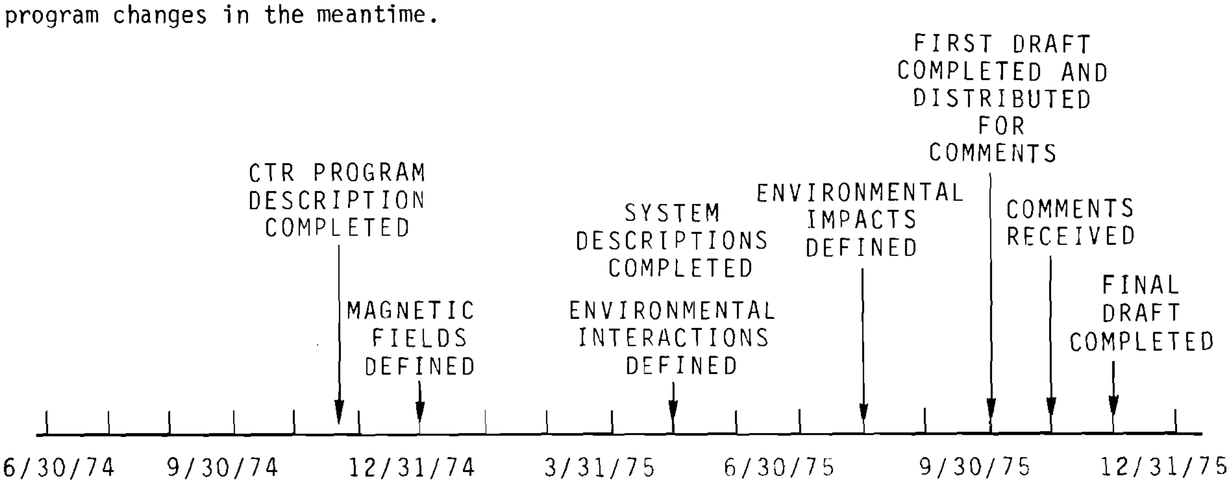

FIGURE 15. CTR Environmental Analysis Program Time Schedules

Preparation of the descriptions of the CTR concepts currently being developed has been completed. Prior to completion of that portion of the document, these descriptions will be reviewed by the laboratories developing these concepts.

An envelope of design concepts for CTR power plants is being developed and the general nature of the magnetic fields has been estimated.

A survey of fission reactor radwaste systems has been completed. A summary description of treatment systems and source terms has been developed for HTGR's, LMFBR's and LWR's, and a preliminary description of radwaste source terms for various CTR designs has been developed.

Most of the other portions of the anlaysis are in the initial period activities of making literature surveys and development of analysis procedures. This conforms to the program schedule which calls for completion of most of the technical analysis early in FY 1976. (Figure 15).

Magnetic Field Effects

DD Mahlum

Since extremely strong magnetic fields will be used in fusion reactors, consideration must be given to the possible consequences of exposure of people, other animals or plant material to these fields. A first step in this consideration is to determine what information is presently available concerning the biological effects of magnetic fields and then to determine what additional data are necessary. PNL is critically reviewing the literature to make recommendations concerning needs in this area.

A conference was held on February 27, 1974, at AEC in Germantown, Maryland, to explore several aspects of the problem of biomagnetic effects. Contemplated and current uses of magnetic fields, results of biological studies and future needs were discussed. The information presented summarized some of the data being produced in the area of biomagnetic effects as well as some of the misinformation which has been disseminated. Minutes of this meeting were prepared as part of this task.

A number of bibliographies and reprints of published work have been obtained and categorized. A search for additional references and sources of information is continuing in an attempt to obtain as much pertinent data as possible for inclusion in the evaluation process. The material currently available is being summarized and additional material included as it becomes available.

The information obtained thus far indicates that biological effects occur under certain conditions of exposure to magnetic fields. Nevertheless, there is a great deal of inconsistency between experiments. Part of this is perhaps a result of poor experimental design in many cases as well as a lack of theory concerning the interaction of magnetic fields with biological materials. The reported results are being critically evaluated in an attempt to define those areas where substantive agreement exists and to determine areas where additional study is necessary.

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