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**Pacific Northwest Laboratory
Report on Fusion Energy Research
January 1977 - March 1977**

April 1977

**Prepared for the Energy Research
and Development Administration
under Contract EY-76-C-06-1830**

Battelle
Pacific Northwest Laboratories

BNWL-1939-7

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PACIFIC NORTHWEST LABORATORY
REPORT ON FUSION ENERGY RESEARCH
JANUARY 1977 - MARCH 1977

April 1977

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Richland, Washington 99352

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PREFACE

This quarterly report on Fusion Energy Research consists of progress summaries of research conducted by the staff of Pacific Northwest Laboratories (PNL). This reporting period includes progress made from January 1, 1976 through March 31, 1977. The ERDA Division of Magnetic Fusion Energy is a major sponsor of the work. However, fusion-related work sponsored by others is also included as appropriate.

The summaries are presented in four major sections:

- Fusion Systems Engineering
- Materials Research and Radiation Environment Simulation
- Safety Analysis and Environmental Effects of Fusion Concepts
- Manpower Development

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.

D. A. Dingee
Fusion Programs Manager

Other Reports in the Series:

- Annual Controlled Thermonuclear Reactor Technology Report-1971, BNWL-17604.
- Annual Report on Controlled Thermonuclear Reactor Technology-1972, BNWL-1685.
- Annual Report for 1973 on Controlled Thermonuclear Reactor Technology, BNWL-1823.
- Annual Report for 1974 on Controlled Thermonuclear Reactor Technology, BNWL-1890.
- PNL Report on Controlled Thermonuclear Reactor Technology, January through September 1975, BNWL-1939-1.
- PNL Report on Controlled Thermonuclear Reactor Technology, October through December 1975, BNWL-1939-2.
- PNL Report on Controlled Thermonuclear Reactor Technology, January through March 1976, BNWL-1939-3.
- PNL Report on Controlled Thermonuclear Reactor Technology, April through June 1976, BNWL-1939-4.
- PNL Report on Fusion Energy Research, July through September 1976, BNWL-1939-5.
- PNL Report on Fusion Energy Research, October through December 1976, BNWL-1939-6.

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PACIFIC NORTHWEST LABORATORY
FUSION ENERGY RESEARCH
JANUARY THROUGH MARCH 1977
ERDA RESEARCH AND DEVELOPMENT REPORT

SUMMARY

The control of thermonuclear reactions is looked upon as the ultimate source of high-temperature energy. Success in the containment of fusion plasma has encouraged scientists to look beyond the confirmation of scientific feasibility to the identification of fusion reactor engineering problems that need solutions before demonstration plants can be built. Some fundamental and applied scientific problems could limit thermonuclear reactor development even after scientific feasibility is attained. Research at PNL continues to emphasize research in these development and technology areas.

The fissile fuel producing capability of an outside blanket for the Tokamak Engineering Test Reactor (TETR) was evaluated in a cooperative study with the University of Wisconsin. Analyses of the modular, helium-cooled blanket design indicate that the TETR fusion driver system could demonstrate sensible plutonium production rates with off-the-shelf UO_2 fuels. The cost of the blanket was estimated to be \$22,450,000 (1976 dollars), excluding the cost of the natural UO_2 needed to fuel the blanket assembly.

Analysis of the economic role of the fusion-fission hybrid continued with an evaluation of the effects of rising fossil fuel costs. Modeled as providing a \$10 billion benefit, the hybrid would nearly eliminate the role of the LMFBR by hybrid production of plutonium to fuel the LWR. Results suggest that parametric hybrid design studies should assume a plutonium value of ~\$50/g in the year 2000, rising to \$90/g in 2020 and stabilizing.

In studies being performed to develop heat transfer and fluid flow design tools for fusion reactor blankets, a survey of computer codes found no code capable of describing the necessary heat transfer and fluid flow information. A detailed thermal hydraulic analysis of the Argonne EPR design concept is underway to determine the capabilities needed in a general design code.

In surface effects research, samples exposed to plasma cleaning processes and to normal plasma operations in UCLA's Microtor Tokamak were examined using Auger Electron Spectroscopy (AES) and sputter profiling. Materials deposited in the port areas of Microtor showed a decreasing oxygen content as machine processing continued, and samples modified by the plasma in the plasma chamber itself showed a similar composition to the materials deposited in the port areas. A vacuum/vacuum transfer chamber was designed to minimize environmental changes while pre-analyzed samples are transported for exposure to Tokamak processes and returned to PNL for analysis. In work examining the blistering effects of polyenergetic helium ions, a blistering work chamber to allow sample heating during ion bombardment was completed. The report from the Round Robin Neutron and Proton Sputtering Experiments was completed.

In studies of the effects of radiation on mechanical properties, wire samples of nickel, niobium, and 316 stainless steel were irradiated with 16 MeV protons to check thermometry and handling of multiple specimens in light ion experiments. Multiple sample packets of test

specimens (Ni, Nb, SS) are being irradiated at the Lawrence Livermore Laboratory (D,T) neutron source and constructed for irradiation at the University of California, Davis (D,Be) neutron source.

In materials development studies, length and density measurements were made of two-dimensional graphite cloths irradiated to a neutron fluence of $1 \times 10^{22} \text{ cm}^{-2}$ EFF at 470°C. Three of four types of cloth maintained integrity despite large changes in density and axial fiber length. It appears that if graphite cloths are to be used as liner material between the plasma and reactor first wall, they will have to be installed to accomodate large dimensional changes. Preliminary design work on a capsule for irradiation of graphite at temperatures up to 2000° in HFIR at ORNL has been completed. In work evaluating electrical insulators for fusion reactor applications, a vacuum system was prepared for dielectric breakdown measurements and the development of acoustic emission techniques to describe prebreakdown behavior. Methods of fabricating insulators with controlled porosity are being developed for experiments to determine the effects of helium buildup on the dielectric breakdown strength of insulators.

Discussions were held with DMFE to define PNL's plan for providing information input and guidance to DMFE's developing program plan for safety-related fusion research. A preliminary assessment of all possible safety hazards is underway.

PNL staff have been teaching a fusion technology course sequence as a part of the University of Washington's graduate study program in nuclear engineering. The current quarter's work has emphasized blanket technology and environmental aspects.

FUSION SYSTEMS ENGINEERING

PNL's effort in Fusion System Engineering comprises research and development aimed at solving some of the major technological problems of bringing fusion power systems on-line to fulfill U.S. energy needs. It is necessary that attention be given early enough in the fusion program to properly identify and solve these engineering problems so that they do not limit the development and implementation of fusion reactors in commercial power production. Work reported this quarter includes the end of two design and development studies for fusion-fission hybrid reactors based upon the Two Component Torus (TCT), an economic analysis of the hybrids' potential role in a situation of rising fossil fuel costs, and an evaluation of computer codes that would help describe heat transfer and fluid flow in reactor blanket cooling studies.

In a recently completed cooperative study with the University of Wisconsin, PNL evaluated the fissile fuel producing capability of an outside blanket for the Tokamak Engineering Test Reactor (TETR) based upon near-term technological developments that could be implemented in the middle or late 1980's. A modular, UO_2 -fueled blanket assembly was adapted to the TETR for maximum production of ^{239}Pu . Thermal hydraulic and structural analyses were performed for the double-wall, helium-cooled blanket design. The study indicates that the TETR fusion driver system could demonstrate sensible plutonium production rates with the off-the-shelf UO_2 fuels. The cost to the TETR was estimated to be \$22,450,000 (1976), including the cost for fuel assembly fabrication but not including the cost of the 215,400 kg of natural UO_2 needed, which would be an additional \$12,570,000 to \$25,140,000.

A study on the conceptual design of a TCT-based hybrid reactor has been completed in a PNL effort coordinated with the Princeton Plasma Physics Laboratory. A draft of the final report is expected by August 1977.

In continuing analyses of the economic role of the fusion-fission hybrid, the effects of rising fossil fuel costs were examined. Hybrid electric efficiency ($40 \frac{\text{MWe}}{\text{MWt}}$) and fissile production rate (1.0 kg/MWt-yr) were considered in a situation using the ERDA (mid) electric demand forecast and an estimated coal price rise of 1%/yr. If the hybrid is modeled as providing a \$10 billion benefit, the role of the LMFBR would be nearly eliminated by the hybrid production of plutonium to fuel the LWR. Important results of the model are the forecasted U_3O_8 price and the forecasted plutonium value. Results suggest that parametric hybrid design studies should assume a plutonium value of $\sim \$50/\text{g}$ in the year 2000, rising to \$90/g in 2020 and stabilizing.

In blanket shield and engineering studies, experimental power reactor (EPR) blanket cooling studies are being performed to develop heat transfer and fluid flow design tools for fusion reactor blankets. A survey of currently available codes was conducted, and no single code was found capable of describing the heat transfer and fluid flow information necessary to the study. A concept combining computer codes to derive the necessary information is being evaluated as work progresses. To determine the needed capabilities of a general code, a detailed thermal hydraulic analysis of the Argonne EPR design concept is being attempted. Work has begun on calculations proposed to determine the upper capability limits of helium cooling with respect to flow rate versus coolant channel length and diameter in several geometries.

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SYSTEMS STUDIES

System studies focus on providing key technical and economic information for planning and analyzing the fusion power program. One concept under study in the national fusion program is the fusion-fission (hybrid) system, which offers a possible near-term application of the fusion principle. The PNL studies of the hybrid are continuing to provide a rational basis for evaluating the merits of this concept in the nation's energy economy. A cooperative effort between Princeton Plasma Physics Laboratory (PPPL) and PNL has been completed on the conceptual design of a hybrid based upon a Two Component Torus (TCT).⁽¹⁾ This concept utilizes intense neutral deuterium beams to drive a tritium Tokamak plasma, and it provides the basis upon which the Tokamak Fusion Test Reactor (TFTR)⁽²⁾ has been designed. Another cooperative effort with the University of Wisconsin has been completed. This study has evaluated the plutonium breeding capability of perhaps a more near-term but yet smaller device--the Tokamak Engineering Test Reactor (TETR)⁽³⁾ which is also based upon a TCT.

Standard economic analyses procedures and data bases are being established to develop optimum-cost time estimates for the introduction of advanced fusion power plants and to assure compatibility of cost estimates between fusion plants and competitive systems. Studies of the energy marketplace are being made to estimate the competitiveness of fusion power systems and establish target costs for fusion reactors.

TETR-HYBRID BLANKET DESIGN

V. L. Teofilo, R. T. Perry, B. R. Leonard, Jr., M. A. McKinnon,
M. C. C. Bampton, C. H. Bloomster, L. L. Fassbender, U. P. Jenquin

PNL in cooperation with the University of Wisconsin (UW) has recently completed a study to evaluate the fissile fuel producing capability of the Tokamak Engineering Test Reactor (TETR) which had been designed by UW to produce a high neutron flux for engineering and materials testing. A list of characteristic TETR design parameters are provided in Table 1.

TABLE 1. TETR Design Parameters

R	3.24 m
a	0.6 m
A	5.4
Elongation	2.0
I _p	2.5 MA
B _T	4.2 T
q (a)	2.4
$\bar{n}_e$	$7.7 \times 10^{13} \text{ cm}^{-3}$
$\bar{T}_e$	11.3 keV
β_p	3.8
$\bar{n}_e \tau_E$	$8 \times 10^{12} \text{ cm}^{-3}\text{-sec}$
Q	1.78
Beam Power	150 MW
Neutron Power	214 MW
Pulse Duration	60 sec
Duty Factor	0.83

The blanket design prepared by PNL is based upon near term fission reactor technological developments which could be implemented in the mid to late 1980s. In order to fulfill this objective, and with a view toward maintainability, a natural UO_2 -fueled outside blanket was adapted for the maximum production of ^{239}Pu . The modular blanket assembly layout adapted to TETR is shown in Figure 1. The vacuum wall and its coolant system is designed to be 316 SS and operated at 250°C in order to operate in a regime where the first wall need not be replaced because of radiation damage during the lifetime of the reactor.

FIGURE 1. Vertical Cross Section of TETR Showing Hybrid Blanket Modular Layout

The design assumes a 65-cm blanket the thickness of which is restricted by a minimum of a 10-cm proximity to any normal conducting water-cooled shaping or vertical field coil (Figure 2). The 316 SS modular assemblies are separated by both the shaping field coil on the horizontal midplane, as well as the beam ports whose angle of penetration prohibit any uniform modular design between adjacent ports. They are interchangeable with respect to the horizontal midplane. There are 32 such separated modular assembly slices, two per TF coil, which allow for assembly and disassembly.

FIGURE 2. Schematic of Blanket Assembly

A cross section of a module and plenum chamber (Figure 3) shows its double wall construction to allow cooling of the structural walls by the incoming helium fluid $T_{in} = 600^{\circ}\text{F}$. The pins are 61 cm long and 1 cm in diameter. The thermal hydraulic analysis using the COBRA-IV-I code⁽⁴⁾ for the power density distribution determined by neutronic computations gives an outlet temperature of 1200°F at 700 psia with a 2.5 psi pressure drop through the fuel rods. The return flow of heated fluid is contained in a flow channel which is surrounded by the incoming flow of cold, helium coolant. The structure of the inner flow channel will see cyclic temperature variations but should have very small pressure loadings. However, the outer wall, which is subjected to very high pressure loadings, will be cooled by the cold incoming fluid so that it should see little, if any, cyclic temperature variation.

The structural analysis performed with the BORSOR4 2-D code⁽⁵⁾ indicates (Figure 4) that a minimum of 2.25 in. (5.27 cm) of 316 SS is needed for structural integrity of this outer wall unless some external structural support is provided. The inner wall, which is supported by the coolant delivery ducting, would not require this large thickness. Nevertheless, this wall should be made as thin as possible (e.g., by using double layered walls) in order to improve the neutron economy.

FIGURE 3. Module and Plenum Cross Section

FIGURE 4. Variation of Maximum Stress in Blanket Module with Thickness

The blanket schematic and volume fractions used for the neutronics computations of the UO_2 -fueled blanket design is shown in Figure 5. Using this configuration, together with the inner shield and plasma fusion neutron power distribution for TETR, the ANISN neutron transport code was run for 1-D vertical cylindrical geometry. The results (Table 2) were corrected for isotopic buildup and burnup as computed by the ORIGIN code.

The TETR fusion driver system could demonstrate sensible Pu production rates with the off-the-shelf UO_2 fuels. The added cost to the TETR system (excluding thermal pumps, heat exchangers, and remote disassembly facilities) for the fabricated modular blanket structure would be \$22,450,000 U.S. (1976), or \$100,250 per module (including the cost for fuel assembly fabrication). To this must be added the nuclear material cost of \$50 to \$100 kg of natural UO_2 for the 251,400 kg which are needed, or \$12,570,000 to \$25,140,000. Based upon these costs, the associated cash schedule and discounted cost yield net credits of \$21 to \$91/kg for the fabricated blanket alone.

FIGURE 5. Blanket Schematic - UO_2 Fueled

TABLE 2. TETR Hybrid Blanket Parameters

Fuel	316-Stainless Steel Clad Pins UO_2 Pellets
Fissions Per Fusion	0.35
Pu Atoms Produced Per Fusion	1.15
Pu Production in 1 Year	740 kg
Average Blanket Power, Density	15.6 W/cm ³
Helium Coolant:	
Inlet Temperature	600°F
Outlet Temperature	1200°F
Pressure	700 psi(a)
Initial Blanket Power	915 MWt
Blanket Power After 1 Year	1175 MWt

TCT HYBRID CONCEPTUAL DESIGN STUDIES

D. T. Aase, R. T. Perry, M. C. C. Bampton, R. A. McCann

The work for this study was completed the end of December 1976, and a final report is being prepared. It is expected that a draft of the report will be issued by the end of August 1977.

ECONOMIC REGIMES

R. L. Engel, D. E. Deonigi

Recent review of the cost of alternate energy systems has indicated that depletion of high-quality coal will result in higher coal costs. This has been confirmed by the FEA National Coal Model used in the PIES model. The potential role of Fusion-Fission (Hybrid) Reactors was examined under the rising fossil cost situation. This is an extension of the parametric analysis reported in previous quarterly reports and in a PNL report recently published.⁽⁶⁾ Coal prices were estimated to rise at 1%/yr. Only one set of hybrid performance characteristics was considered: electric efficiency of 40 $\frac{\text{MWe}}{\text{MWt}}$ and annual fissile plutonium production rate of 1.0 kg/MWt. The ERDA (mid) electric demand forecast was considered.

The allowable capitalized cost for the hybrid is shown in Table 3 for both cases with the LMFBR and without the LMFBR. The zero benefit cost is the cost at which the hybrid is barely economically feasible. The \$10 billion benefit assumes a 7.5% discount rate.

TABLE 3. Allowable Hybrid Capitalized Cost Expressed as a Multiple of LWR Cost (LWR Equivalent)

<u>Benefit (\$ Billion)</u>	<u>With LMFBR</u>	<u>Without LMFBR</u>
0	1.8	2.3
10	1.4	2.2

The building schedule for various technologies is the primary solution of the model. The installed capacities for various technologies are shown in Figure 6 and Figure 7 for the case with the LMFBR. The area between the curves represents the installed capacity for each technology. Figure 6 is the installed capacity without the hybrid and Figure 7 is the installed capacity when the hybrid provides a \$10 billion benefit. The role of the LMFBR is nearly eliminated by the hybrid production of plutonium to fuel the LWR. The hybrid at 1.4 times the LWR cost reduces the capacity of LMFBRs installed in the year 2030 from 2040 GWe to 175 GWe, while the LWR increased capacity from 1160 GWe to 2450 GWe. There will be 610 1-GWe hybrids installed in the year 2030.

Another important result of the model is the forecasted U_3O_8 price and the forecasted fissile plutonium value. These forecasted prices are displayed in Figure 8 for the case with the LMFBR and in Figure 9 without the LMFBR. The shaded area represents the changes caused by the availability of the hybrid. The U_3O_8 price reaches \$125/lb without the LMFBR and is only reduced \$5 by the hybrid. With the LMFBR, the U_3O_8 price reaches \$100/lb and the hybrid reduces it by \$13. Without the LMFBR, plutonium value is constantly rising to \$88/g, and the hybrid causes a 10% reduction to \$80/g. In the case with the LMFBR, the plutonium value peaks at \$92/g in 2020 as the rapid building of LMFBRs creates a high demand for plutonium and then

FIGURE 6. Forecasted Operating Electrical Installed Capacity (GWe) for Low Demand Growth without Hybrid

FIGURE 7. Forecasted Operating Electrical Installed Capacity (GWe) for Low Demand Growth with Hybrid (\$10 Billion Benefit)

FIGURE 8. Forecasted U_3O_8 Prices (\$/lb) and Plutonium Values (\$/g) with Low Demand Growth and with LMFBR

FIGURE 9. Forecasted U_3O_8 Prices (\$/lb) and Plutonium Values (\$/g) with Low Demand Growth and No LMFBR

falls rapidly as the LMFBR produces excess plutonium. In this case, the hybrid stops further increase in the value of plutonium at \$48/g immediately after its availability in the year 2000. The plutonium value then remains at \$40/g since both the LMFBR and the hybrid produce plutonium, causing an abundance.

Based on these results, it is suggested that parametric hybrid design studies assume a plutonium value of \$50/g in the year 2000, rising to \$90/g in 2020 and stabilizing. The comparable value of kWh would be 30.8 mills in the year 2000, rising to 32.2 in 2010, and then declining linearly to 26.3 in 2040. The decline in power cost is the result of the success of the advanced technologies.

PRELIMINARY ECONOMIC ANALYSIS

J. R. Young

Design studies for fusion power plants are expected to be issued frequently during the development program. These studies will provide the basis for economic and environmental analyses which determine research needed to produce the most attractive designs. Use of past design studies for analyses has been difficult because of lack of information or inconsistent presentation of information.

Economic and environmental information requirements have been provided in a document prepared for the Division of Magnetic Fusion Energy. This document describes the related types of design, operation, and economic information that should be included in each design study and establishes certain basic general rules and assumptions that should be used in the development of that information. Provisions of such information in design studies will assure that the necessary data are available and that satisfactory comparisons of design concepts can be made.

A draft of the information requirements document was prepared and forwarded to the Division of Magnetic Fusion Energy for comments. As the result of those comments, the section on environmental information requirements has been removed and an example of costing for a typical power plant has been added. The revised draft document has been forwarded to the Division of Magnetic Fusion Energy for distribution prior to a workshop on the information requirements.

BLANKET AND SHIELD ENGINEERING

Blanket and Shield Engineering studies focus on developing valid technical bases for reactor design and transferring the technology to industry capability. Experimental Power Reactor (EPR) blanket cooling studies are being performed to develop heat transfer and fluid flow design tools for fusion reactor blankets. PNL is also participating in the nuclear data community to define nuclear data needs and develop best-estimate nuclear data files for DMFE technology programs.

HEAT TRANSFER AND FLUID FLOW

D. T. Aase, C. W. Stewart

The objective of these studies is to develop design analysis tools and to scope experimental needs to investigate, define, and assess the heat transfer and fluid flow development requirements for the base technology of fusion reactors.

EPR blanket cooling studies are being performed to develop heat transfer and fluid flow design tools for fusion reactor blankets. The study consists of two parts: in the long term, to develop thermal hydraulic design tools and, in the short term, to evaluate the maximum capabilities of helium cooling.

A brief survey of currently available codes was conducted with the general conclusion that nothing off the shelf is going to do the job. It appears that the best codes have the following general capabilities:

- Complete and detailed conduction solution coupled with very crude thermal hydraulics. (TAP, HEATING4)
- Complete and detailed thermal hydraulic solution with insufficiently general or detailed conduction model. (RELAP, COBRA)
- Complete combined thermal hydraulic and conduction models for specific one-dimensional geometries.

Generality is the key word in considering a possible design code. The toroidal geometry and modular design of most controlled thermonuclear reactor concepts produces a wide spectrum of non-symmetric, multi-dimensional, non-homogeneous, and discontinuous applications requiring equal treatment of conduction and thermohydraulics. The building block approach of the TRUTH and TAP codes seems ideal here. At this stage, it looks like a combination of TAP with the bare hydrodynamic solution from COBRA would be sufficient for any fusion reactor design work. This concept will continue to be evaluated as work progresses.

In order to determine just what capabilities will be needed, an attempt is being made to perform a detailed thermal hydraulic analysis of the Argonne EPR design concept. The TRUTH code is being set up initially and the TAP code is being activated and converted to CDC system for comparison.

Although helium has been considered in many conceptual fusion reactor designs, it has generally been in the context of a comparison with another coolant or constrained to a particular geometry. This task proposes to let the calculations "run free" and find out just how far helium cooling can be pushed.

There are only two basic limits which determine the maximum heat transfer rate for helium: a maximum surface temperature and a maximum permissible pumping power. Given a length and hydraulic diameter of the coolant channel, these two limits fix the flow rate. If a coolant volume fraction is known, the required flow rate can be translated into a maximum volumetric heating rate.

It is proposed that a series of calculations be performed to define a "limit surface" of flow rate versus length and diameter for several geometries. This would be the direct inverse of the normal procedure of fixing the flow rate and dimensions and then checking whether the limits have been exceeded.

No calculations have been performed yet, but a tentative system of equations has been derived to incorporate the pumping power and temperature limits in a closed form solution.

NUCLEAR DATA STANDARDS

B. R. Leonard, Jr.

The objective of this task is to participate in the development of current best-estimate nuclear data files for DMFE Technology programs. During this quarter a special meeting on the ENDF/B-V Neutron Dosimetry File was attended at the National Bureau of Standards. Individual responsibilities for Phase I review of these files were fixed during the meeting. An International Specialists Symposium on Neutron Standards, for which the principal investigator served on the technical program committee, was also attended at NBS.

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MATERIALS RESEARCH AND RADIATION ENVIRONMENT SIMULATION

The various materials and components of a fusion reactor will be subjected to extreme physical environments, intense fluxes of high energy neutrons, alpha particles, and ionic species as well as internally generated products of neutron-induced reactions, such as helium. PNL is examining the effects of these environments on candidate fusion reactor materials, using reactor irradiations and simulation techniques to: 1) determine in-reactor material transfer during normal plasma operation and different discharge cleaning modes, 2) correlate changes in mechanical properties produced by light ions and fusion neutrons, 3) evaluate carbons and graphites for possible use as first-wall liner materials, and 4) determine the effects of porosity and helium buildup on the dielectric breakdown strength of electrical insulators.

The PNL Surface Science group has been working on techniques for measuring plasma surface interactions on present day plasma machines as well as studying the effects of surface preparation and impurities on the operation of Tokamaks. Samples exposed to normal plasma operations and plasma cleaning processes in UCLA's Microtor Tokamak were examined at PNL using Auger Electron Spectroscopy (AES) and sputter profiling. Material deposited in the port areas of Microtor showed a decreasing oxygen content as machine processing continued, and samples modified by the plasma in the plasma chamber itself showed a similar composition to the materials deposited in the port areas. Surface stable graphite and stainless steel samples were prepared and are being inserted in both the Macrotror and Microtor Tokamaks at UCLA. They will provide a stable baseline to which changes on samples exposed to discharge cleaning and normal plasma discharge can be compared. A vacuum/vacuum transfer chamber was designed to minimize environmental changes while allowing pre-analyzed samples to be exposed to Tokamak processes and then returned to PNL for analysis. In other surface science work this quarter a blistering work chamber was completed for a study examining the blistering effects of polyenergetic helium ions; the chamber allows sample heating during ion bombardment. The report resulting from the Round Robin Neutron and Proton Sputtering Experiments was completed and is being distributed.

Mechanical properties studies at PNL are continuing with experiments to determine the correlation between the properties of materials irradiated with light, penetrating ions and those irradiated with neutrons. Twelve wire samples of nickel, niobium, and 316 stainless steel were irradiated with 16 MeV protons at 32°C and a dose rate of $\sim 10^{12} \text{ p}^+/\text{cm}^2\text{-sec}$ to check thermometry and handling of multiple specimens. Calibration of the ion current monitoring system is in progress for the light ion irradiation apparatus at the University of Washington Nuclear Physics Laboratory. Preliminary beam diagnostics were made monitoring the current in vacuum and 760 torr helium gas. Room temperature fusion neutron irradiations of multiple sample arrays are continuing at the Lawrence Livermore Laboratory (D,T) neutron source. The sample packet containing wire tensile test specimens (Ni, Nb, SS) and nickel and niobium foils for microstructural studies will be irradiated to an accumulated dose near $5 \times 10^{17} \text{ n/cm}^2$. Similar sample packets were constructed for neutron irradiations at the University of California, Davis (D,Be) source. The sample capsule for elevated temperature (200°C) fusion neutron irradiations was designed and fabricated.

Materials studies have continued to evaluate radiation damage, sputtering, and outgassing for carbons and graphites to determine if these are suitable for use as low-Z liner materials between the reactor first wall and the plasma. Length and density measurements were taken for two-dimensional graphite cloths irradiated at 470°C. Despite large changes in density and axial length of fibers, three of the four types of cloth maintained their integrity very well. Thornel-50 cloth deteriorated significantly, and two types of three-dimensional cloth samples irradiated at the same time literally fell apart. Though the fluences (to $1 \times 10^{22} \text{ cm}^{-2}$ equivalent fission fluence) in this irradiation were ten times higher than for any previous tests on graphite cloths or fibers, irradiations to higher fluences and temperatures are needed to evaluate possible cloth behavior in a fusion reactor environment. It appears that if graphite cloths are to be used in fusion reactors, they will have to be installed to accommodate large dimensional changes. Preliminary design work on a capsule for irradiation of graphite at temperatures up to 2000° in HFIR at ORNL has been completed. Two of three irradiations have been completed in the task to correlate relative atomic displacement rates in graphite irradiated with neutrons of different energies. The third irradiation is scheduled to be redone in the Brookhaven Medical Research Reactor because of inadequate temperature control in the original irradiation.

Work has continued in the development of acoustic emission techniques for determining prebreakdown behavior and failure mechanisms in electrical insulators with potential use in fusion reactor applications. Major efforts this reporting period were in preparation of a vacuum system for dielectric breakdown measurements and the fabrication of samples for experiments to determine the effects of helium buildup on the dielectric breakdown strength of insulators. Instrumental calibration of the vacuum system, which is designed to operate at $1.3 \times 10^{-4} \text{ Pa}$, has been completed. The system will allow high-voltage dielectric breakdown measurements ($\sim 10^6 \text{ V}$ at $1.3 \times 10^{-4} \text{ Pa}$) at temperatures to 300°C. Fabrication methods are being developed to introduce controlled porosity into MgAl_2O_4 , one of the insulators selected for the helium buildup studies. Some polycrystalline samples of MgAl_2O_4 were prepared for testing.

SURFACE SCIENCE RESEARCH RELATED TO FUSION TECHNOLOGY

This work is examining the effects of the fusion reactor environment on surface and near-surface regions of reactor structures and the influence of plasma-wall interactions on the plasma. Ions, neutral particles, neutrons, photons and electrons from the plasma will interact with exposed surfaces to produce effects such as physical and chemical sputtering, desorption, blistering, and re-emission. Material which leaves the wall due to these effects may then enter the plasma and cause loss of plasma energy through bremsstrahlung or line radiation. Erosion will weaken structural integrity and may adversely influence surfaces exposed to coolant in the cooling channels.

SURFACE EFFECTS

M. T. Thomas, D. L. Styris, D. R. Baer

The PNL Surface Science Group has been working on techniques for measuring plasma surface interactions on present day plasma machines as well as studying the effects of surface preparation and impurities on the operation of Tokamaks. Some of these studies are being done in cooperation with the UCLA Tokamak Group. As part of this program samples were exposed to plasma cleaning processes and to normal plasma operations and then were sent from UCLA to PNL for analysis. Auger Electron Spectroscopy (AES) and sputter profiling were the techniques used to examine these samples. The samples studied were from the small Microtor Tokamak during different stages of cleaning and plasma operation. Others were from the larger Macrotor Tokamak while it was first being cleaned. Measurements of material collected in the port areas of Microtor show a decreasing oxygen deposit as machine processing continues, which correlates well with spectroscopy measurements made on the plasma. Samples in the plasma chamber itself are modified by the plasma and show a composition similar to the materials deposited on the walls in the port area. Materials exposed to glow discharge cleaning in Macrotor are significantly altered by the process; although the relative concentrations of the elements are altered, oxygen is the primary element being deposited on the port areas samples. We have recently prepared graphite and stainless steel samples which are being inserted in both Macrotor and Microtor. The surfaces of these samples have been analyzed and found to be stable during the vacuum loading and unloading handling. Therefore, we have a good baseline to which changes on samples exposed to discharge cleaning processes or normal plasma discharge can be compared.

A vacuum/vacuum transfer chamber has been designed which will allow pre-analyzed samples to be exposed to Tokamak processes and then returned to PNL for analysis. The samples will be maintained in carefully controlled vacuum conditions to minimize environmental changes.

A Secondary Ion Mass Spectroscopy (SIMS) probe has been received and installed on our AES system. This surface probe will give us additional capability to observe surface composition and surface changes on the samples that have been exposed to plasma machines.

A new blistering work chamber which will allow sample heating during ion bombardment has been completed and is presently connected to PNL's 2 MeV Van de Graaf accelerator. Bake out and calibration procedures are currently underway.

The report resulting from the Round Robin Neutron and Proton Sputtering Experiments is complete and is being distributed.

BULK RADIATION EFFECTS AND SIMULATION

Structural materials in magnetic fusion reactors will be subjected to bulk radiation damage from energetic neutrons coupled with cyclic temperatures and stresses. Irradiation-induced creep, fatigue, and embrittlement of materials are serious limitations on the design life of many fusion reactor components. An understanding of the specific effects of irradiation on creep, fatigue, and crack growth rates is necessary for the development and selection of alloys for these components; however, materials studies for fusion reactor applications are severely hampered by the lack of a high-flux, high-energy, large-volume irradiation facility. Charged particle irradiations of materials can help satisfy the flux and energy requirements for irradiation damage studies while lowering the necessary radiation levels (relative to neutronic irradiations) and offering the potential to control radiation parameters. A balanced program of light ion, fusion neutron, and fission neutron irradiation studies is underway to gain an understanding of the effects of a fusion neutron environment on the mechanical properties of materials.

RADIATION ENVIRONMENT SIMULATION - MECHANICAL PROPERTIES

R. H. Jones, D. L. Styris

The purpose of the present experiments being performed under this program is to determine the correlation between the properties of materials irradiated with light, penetrating ions and those irradiated with neutrons. Wire samples are being used for the postirradiation tensile testing, and foil samples are being used for microstructural study. The irradiations use fusion and fission neutrons and 16 MeV protons.

Program elements which have been initiated during this quarter are ion irradiation of multiple samples, additional fusion neutron irradiation of multiple samples, ion beam diagnostics, and thermometry.

ION IRRADIATION

The ion beam is provided by the University of Washington tandem Van de Graaf accelerator, which is isolated from a high-vacuum irradiation chamber by a 6350- μm Havar window. A Faraday cup monitors the ion current inside the vacuum chamber, and a lithium-doped silicon particle detection system monitors ions backscattered from the window. Calibration of this system is in progress. Once calibrated, the backscattering technique will provide a means of monitoring ion current when the irradiation chamber is backfilled with helium cooling gas. Backscattering data is stored on magnetic tape and processed at the University of Washington Nuclear Physics Laboratory computer facilities. Beam profiles are monitored by a beam scanner 0.5 m upstream from the Havar window. Preliminary beam diagnostics have been made monitoring the current in vacuum and 760 torr helium gas.

Twelve wire samples (4 - 316 SS, 4 - Ni, 4 - Nb) have been irradiated with 16 MeV protons (p^+) at 32°C and a dose rate of approximately $10^{12} p^+/cm^2\text{-sec}$. This is comparable to the neutron dose rates achieved on sample packets at the Rotating Target Neutron Source. The ion fluence was $10^{16} p^+/cm^2$. This preliminary irradiation was made to check specimen handling and thermometry. Temperature was monitored by two, butt-welded, chromel-alumel thermocouples mounted as dummy samples within the multiple sample array. Sample cooling was accomplished with static (760 torr) helium gas in the irradiation chamber.

NEUTRON IRRADIATION

Room temperature fusion neutron irradiations of multiple sample arrays are continuing at the Lawrence Livermore Laboratory (D,T) neutron source. Accumulated dose will be near $5 \times 10^{17} n/cm^2$. The sample packet contains a planar array of forty wire tensile test specimens (Ni, Nb, SS) and a nickel and niobium foil along with niobium dosimeters.

Similar sample packets have also been constructed for neutron irradiations at the University of California, Davis (D,Be) source. Two irradiations to fluences of 10^{17} and $10^{18} n/cm^2$ will be made at this facility. This will provide data needed to make comparisons between damage from (D,T) neutrons, (D,Be) neutrons, and 16 MeV protons.

The sample capsule for elevated temperature (200°C) fusion neutron irradiations has been designed and fabricated. Irradiation to a fluence of $6 \times 10^{16} n/cm^2$ is scheduled at the Lawrence Livermore Laboratory facility.

MATERIALS TECHNOLOGY

Programs to identify and solve materials problems are essential to successful commercial fusion power development. The PNL materials technology program addresses areas where information will be needed for experimental and commercial reactors including the behavior of graphites and electrical insulators in fusion reactor environments. New experiments in the electrical insulator studies are examining the effects of porosity and helium buildup on the dielectric breakdown strength of insulators.

EVALUATION OF CARBONS AND GRAPHITES FOR FUSION REACTOR APPLICATIONS

W. J. Gray, W. C. Morgan, G. L. Tingey

Some fusion reactor concepts have proposed the use of a low-Z liner between the plasma and the first structural wall. A number of materials are being considered for the liner, including graphite or graphite cloth. Among the more important data required before the choice of liner materials can be narrowed are: 1) radiation damage, 2) neutron, ion, and chemical sputtering, and 3) degassing.

To provide the necessary information on graphite, the following areas are being investigated:

- Radiation Damage in Graphite Cloth - evaluation of radiation damage effects in cloths and fibers since there are no previous data for these materials above about 1×10^{21} cm⁻² EFF. (a)
- Radiation Damage at Very High Temperatures - evaluation of radiation damage effects at temperatures up to 2000°C since no data exist above ~1400°C.
- Fission-Fusion Correlation - experimental testing of the correlation between damage production rates in fission and fusion reactor spectra. This will allow more accurate projections of graphite performance in fusion reactors based on the extensive graphite data from fission reactor irradiations.
- Sputtering - measurement of neutron, ion, and chemical sputtering rates for carbon. This work is being done in conjunction with other surface science research presented in this report.
- Degassing - measurement of degassing properties of large graphite samples under conditions appropriate to fusion reactor applications.

RADIATION DAMAGE IN GRAPHITE CLOTH

Figures 10 and 11 show length and density changes, respectively, as a function of fluence for two-dimensional cloths irradiated at 470°C in EBR-II. The length changes were determined

(a) All fluences are quoted in terms of "Equivalent Fission Fluence for Damage in Graphite." (7)

FIGURE 10. Radiation Induced Length Changes of Graphite Cloth at 470°C

FIGURE 11. Radiation Induced Density Changes of Graphite Cloth at 470°C

by measuring the diameter changes of cloth discs originally ~1.2 cm in diameter. They closely approximate, therefore, axial length changes of the graphite fibers. Densities were measured by immersion in toluene. Despite the large changes (~6 times larger than for typical nuclear grade graphites irradiated under the same conditions), three of the four types of cloth maintained their integrity very well. Thornel-50 cloth did deteriorate significantly, however.⁽⁸⁾ Also, the two types of three-dimensional cloth samples included in this irradiation literally fell apart. The samples were too small to be truly representative, but it appears likely that a type of three-dimensional weave more able to accommodate large changes in fiber dimensions must be developed if these types of materials are to be used in fusion reactors.

The small density change of Thornel-50 relative to the other cloths and relative to the length changes implies that these fibers have swelled in the radial direction. Measurements to confirm this are in progress.

The fluences achieved in this irradiation are nearly ten times higher than for any previous known tests on graphite cloths or fibers. Thus, they represent the first irradiations of any real interest for fusion reactor applications. Nevertheless, irradiations to still higher fluences and at higher temperatures are needed. Our speculations are that length changes at higher temperatures are unlikely to be grossly different from those shown here. We are not willing to speculate, however, about how well cloths would maintain their integrity at higher temperatures nor about their behavior at higher fluences. It appears that if graphite cloths are to be used in fusion reactors, they will have to be installed in a manner that will accommodate large dimensional changes. This task, including publication of results, will be completed next quarter.

RADIATION DAMAGE AT VERY HIGH TEMPERATURES

Preliminary design work on a capsule for irradiation of graphite at temperatures up to 2000°C in HFIR at ORNL has been completed. The high temperatures will be achieved by placing a tungsten heater shell around the graphite. The capsule will be non-instrumented; thus, temperatures must be calculated. This requires that the total emissivities of the tungsten and cladding (aluminum) surfaces be known quite well. Experiments to measure these values in the laboratory are being planned.

The present schedule calls for the first capsule to be charged into the reactor late in FY-1977. There is some possibility, however, that irradiations will be done in ORR at ORNL. If so, the first capsule will not be charged into the reactor until sometime in FY-1978.

FISSION-FUSION CORRELATION

Relative atomic displacement rates in graphite irradiated with neutrons of different energies are being determined by measuring relative changes in the elastic modulus of highly oriented pyrolytic graphite samples irradiated in three different neutron sources. The three sources are:

- 1) The RTNS at LLL with neutron energies of ~ 15 MeV
- 2) The (D,Be) source at Davis, California, operated at an average neutron energy of ~ 5 MeV
- 3) A thermal fission reactor where average neutron energies are ~ 1 MeV.

Two of the three irradiations have been successfully completed. Inadequate temperature control (the samples ran too warm) during the third irradiation, which was done in the Hanford N-Reactor, requires that it be redone. The re-irradiation is scheduled to take place toward the beginning of next quarter in the Brookhaven Medical Research Reactor. If no further problems develop, this task will be completed near the end of next quarter.

SPUTTERING

A glassy carbon sample labeled with ^{14}C has been prepared for use in neutron and other sputtering experiments. A proportional counter prepared especially for measuring the quantity of sputtered ^{14}C is currently being debugged. A series of blank runs with the labeled sample will then be made before actual sputtering measurements are started.

DEGASSING

Degassing measurements using the apparatus described last quarter⁽⁸⁾ are in progress. Results obtained to date are still very preliminary in nature and will not be reported until next quarter.

ELECTRICAL INSULATORS FOR FUSION REACTOR APPLICATIONS

J. L. Bates, J. E. Garnier

Current technology calls for the use of electrical insulators in various locations in controlled thermonuclear reactors. These insulators will be subjected to a variety of cyclic conditions and must function in the reactor's complex radiation environment. One of the more stringent requirements is the need for insulators which are electrically, thermally, and mechanically stable to intense neutron, particle, gamma, and beta radiation, e.g., $\sim 10^{16}$ nv and $\sim 10^{23}$ nvt. In some cases the insulator will be required to withstand very high electrical potentials (~ 100 kV/cm) and cyclic conditions at high temperatures.

One of the major goals of this program is to evaluate the dielectric breakdown behavior of ceramic insulators using acoustic emission (AE) techniques. These studies emphasize:

- 1) The development and understanding of acoustic emission information generated during the occurrence of electrical breakdown and the relationship of these data to material properties.
- 2) The effects of helium and hydrogen buildup on the dielectric strength and electrical resistivity of insulators.

Progress in these areas is described below with our major efforts devoted to the preparation of a vacuum system for dielectric breakdown measurements and the fabrication of samples for helium effect studies.

PHYSICS OF ACOUSTIC EMISSION IN INSULATORS

Fundamental to the use of AE techniques for determining prebreakdown behavior and failure mechanisms in electrical insulators is the development of an understanding of what AE information can reveal. This requires an evaluation of the AE breakdown technique relative to insulator structure and breakdown characteristics under high electrical stress. Measurements are being conducted to determine the informational content of the recorded AE data and reproducibility of results. AE count, AE energy count, AE event energy, frequency distribution of an AE event, and the instantaneous current flow to ground through the insulator are being evaluated. The information is being correlated with well characterized single crystal and polycrystalline samples of Al_2O_3 and MgAl_2O_4 as a means of evaluating AE breakdown data as related to various microstructural and temperature effects.

Previous breakdown studies at this laboratory have been made in an oil bath to prevent current breakdown outside the sample. However, in order to evaluate the acoustic emission method for studying dielectric breakdown at very high voltages and at higher temperatures, measurements should be made in either a high vacuum (1.3×10^{-4} Pa) or at high gas pressures. The measurements in vacuum were selected primarily because a vacuum is more representative of the low pressure environment in controlled thermonuclear reactors than the very high gas pressures used by many investigators for dielectric breakdown strength. In addition, the vacuum acoustic emission technique is more readily adapted to high voltage and high temperature measurements and to in situ radiation studies. The cost and time for installation of equipment is also substantially less.

Efforts this quarter have been devoted toward the completion of a vacuum test chamber and internal dielectric breakdown testing apparatus. The system is designed to operate at 1.3×10^{-4} Pa using an ion pumping system to eliminate organic vapors. Dielectric breakdown measurements can be made to 300°C (minor modifications can raise this to 1000°C). The high vacuum will also allow high voltage measurements at these temperatures, i.e., $\sim 10^6$ V at 1.3×10^{-4} Pa and 8×10^4 V at 1.3 Pa for a 1-cm electrode separation. Samples are being tested under dc voltages up to 60 kV steady-state or ramp and pulse voltages to 26 kV. Instrumental calibration of the system has been completed.

EFFECTS OF HELIUM AND HYDROGEN BUILDUP

The effects of helium and hydrogen buildup on the electrical resistivity and dielectric breakdown strength are being evaluated. The study will initially investigate two methods of introducing these gases into the insulator: 1) gases introduced into a controlled microstructure, e.g., pores, during the fabrication process, and 2) gases injected at high temperature and high pressure. Both of these techniques are dependent upon the ability to fabricate samples with a desired microstructure.

Al_2O_3 and MgAl_2O_4 insulator materials have been selected for these studies. Both can be obtained as single crystals, and Al_2O_3 can be obtained as high-density polycrystalline bodies. Presently, we are developing fabrication methods to introduce controlled porosity

into MgAl_2O_4 , i.e., controlling pore volume and size distribution. Once this is accomplished, fabrication in varied He or H_2 gases at different pressures will allow a controlled gas pressure in the pores. Some polycrystalline samples of MgAl_2O_4 have been prepared for preliminary testing. The high-pressure, high-temperature furnace has been modified to evaluate the introduction of helium into insulators.

SAFETY ANALYSIS AND ENVIRONMENTAL EFFECTS OF FUSION CONCEPTS

In safety analysis and the environmental assessment of fusion concepts, PNL is working to assure the timely availability of information which will be required when safety analysis reports and environmental impact statements are prepared for DMFE's fusion power development program. The identification of fusion safety and environmental impacts early enough in the development program would permit maximum emphasis on research and reactor designs to minimize those impacts.

During the past quarter discussions were held with DMFE to define PNL's plan for providing information input and guidance to DMFE's developing program plan for safety-related fusion research. As a first step in developing a safety research program plan, a preliminary assessment of all possible safety hazards is proceeding at PNL in four areas: 1) identification of direct hazards to human safety and their possible sources, 2) investigation of specific accidents with potentially severe consequences, 3) investigation of procedures and equipment for continuous operational safety, and 4) analysis of failure modes and effects for reactor system components.

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FUSION SYSTEMS SAFETY ANALYSIS

The development program leading toward practical fusion power will involve construction and operation of a number of experimental reactors. This will require an understanding of the normal and accident behavior of the reactor systems in order to assure the safety of the operating staff and the public. Potential safety concerns in such facilities and in the support activities (e.g., transportation, waste disposal, etc.) need to be addressed early in the program to allow timely development of necessary data and predictive methods. PNL's fusion systems safety analysis task is identifying the key needs and providing a plan for obtaining this information. This task will also review existing and evolving safety criteria which may be applicable to fusion reactors.

FUSION SAFETY PROGRAM PLANNING

H. J. Willenberg, S. H. Bush

A program plan is being developed to guide DMFE's safety-related fusion research. The plan will be keyed to the Fusion Power R&D Program Projections and to the Fusion Systems Engineering Program Plan. Potential safety concerns will be identified for each major experimental program device. Information and analysis tools needed to analyze these concerns will be identified. The program plan will provide for assessment of the adequacy of existing data and analysis tools and will provide a work plan and schedule for developing additional needed information. This effort will utilize information from reactor safety studies. It will, in turn, provide schedules for development of safety analysis methods and data, safety criteria, detection and control systems, and verification experiments.

During the past quarter discussions were held with DMFE for the purpose of defining PNL's plan for accomplishing these goals. Agreement was reached on a schedule for providing DMFE with guidance for a fusion safety research program plan on a timely basis. Discussions were held with all U.S. laboratories engaged in safety-related magnetic fusion research.

The first step in developing a safety research program plan is to make a preliminary assessment of all possible safety hazards. When a plan is being prepared for a technology which is in such a conceptual stage as magnetic fusion power, extra precautions should be made to ensure that a maximum range of potential issues are addressed. It is essential for providing this broad base that the methodology for performing the preliminary safety assessment be flexible. This assessment is proceeding at PNL in four areas:

- direct hazards to human safety and their possible sources are being identified,
- specific accidents with potentially severe consequences are being investigated,
- procedures and equipment for assuring continuous operational safety are being explored, and
- a Failure Modes and Effects Analysis is being performed on reactor system components.

Safety hazards to plant personnel and the public are classified in Table 4. This is an outline of all potential hazards to humans, organized with emphasis on the hazards rather

than specific reactor components or accident scenarios. The most significant hazards appear to be those associated with radiation, pressure, and missiles. This analysis has revealed that, in addition to the radiation hazards due to blanket activation, tritium, and corrosion product transport, attention must be paid to direct radiation streaming to the reactor building, activation of reactor building atmosphere, and volatilization of blanket activation products. The pressure and thermal disturbances due to stored energy in graphite and cryogenic organic insulators, liquid-liquid contact (such as lithium-water or air-helium), and hydrogen explosions must be investigated. The potential for missile generation due to magnet or vacuum failure should be carefully investigated.

TABLE 4. Safety Hazard Classification

<p>A. Radiation</p> <ol style="list-style-type: none"> 1. Exposure from Reactor Vessel 2. Air Activation <ol style="list-style-type: none"> a. Plasma In-leakage b. Reactor Building Atmosphere 3. Tritium 4. Corrosion Products 5. Volatile Structure Materials <p>B. Heat</p> <ol style="list-style-type: none"> 1. Liquid Metal Fires 2. Electrical Fires 3. Loss of Heat Transfer 4. Component Overheating <p>C. Cold</p> <ol style="list-style-type: none"> 1. Magnet Coolant 2. Cryogenic Distillator 3. Vacuum Coolant <p>D. Pressure</p> <ol style="list-style-type: none"> 1. Stored Energy 2. Liquid-liquid Contact 3. Explosions 	<p>E. Electromagnetic Fields</p> <ol style="list-style-type: none"> 1. Arcing 2. Magnetic Fields 3. Radiofrequency and Microwave Radiation <p>F. Stable Toxic Materials</p> <ol style="list-style-type: none"> 1. Lithium Compounds 2. Beryllium 3. Coolants 4. Others <p>G. Internal Missiles</p> <ol style="list-style-type: none"> 1. Structural Failure <ol style="list-style-type: none"> a. Magnet Failure b. Magnetic Forces on Conductors c. Vacuum Failure d. Structural Collapse e. Coolant Overpressurization 2. Explosions <ol style="list-style-type: none"> a. Hydrogen b. Magnet Coolant <p>H. Suffocation</p>
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While many general safety hazards can be analyzed independently of their sources, the potential consequences of certain accident scenarios are sufficiently severe and unique to warrant careful investigation of specific accident pathways. These accidents are listed in Table 5. Fault tree construction and analysis of specific accident scenarios should be performed for these accident initiators. The emphasis in liquid metal spills is on lithium, but sodium and potassium spills should also be addressed. The analysis should consider the potential for fires, the damage to reactor systems due to heat, pressure and release of chemically reactive species, toxicity, and means for prevention and control. The fuel spill issue should include the consequences of tritium release, hydrogen explosions, and chemical reactions. Loss of heat transfer includes a variety of coolant perturbations, such as loss

of coolant, loss of flow, and power oscillations. The consequences of natural disasters such as earthquakes, floods, and tornadoes will be investigated. Table 6 lists topics which do not relate to specific accident analysis, but which have an impact on safe operation and maintenance of reactor systems. Early consideration of the safety impact of these topics in reactor design will greatly decrease the reactor down time due to safety-related problems and backfitting.

TABLE 5. Accident Analysis

- A. Liquid Metal
- B. Fuel Spill
- C. Loss of Heat Transfer
 - 1. Magnet
 - 2. Primary Coolant
- D. Natural Disasters
- E. Sabotage

TABLE 6. Operational Safety Assurance

- A. Remote Handling
- B. Instrumentation
- C. Dosimetry
- D. Waste Disposal and Recycle
- E. Material Degradation

Finally, a Failure Modes and Effects Analysis (FMEA) has been initiated. This analysis addresses the possible modes in which reactor components can operate, and identifies the possible effects on other reactor systems. Specific component failure modes are outlined in Table 7. It should be emphasized that this analysis, as well as all the others in this preliminary safety hazards assessment, is performed in order to identify potential safety concerns, and does not address economic issues related to component failure.

TABLE 7. Failure Modes and Effects Analysis

A. Burning Plasma 1. Power Fluctuations 2. Plasma Discharge to First Wall 3. Rapid Radiative Loss 4. Fluctuations in Particle Number 5. Impurity Level Fluctuations 6. D-T Imbalance B. Plasma Heating 1. Radiofrequency Waveguide Electric Discharge 2. RF Heater Interference with Electric Circuits 3. Neutral Beam Transmission Through Target Plasma 4. Waveguide Vacuum Failure 5. Injector Cryopanel Failure C. Fuel System 1. Neutral Beam/Pellet Over-Injection 2. Tritium Storage System Permeation 3. Tritium Permeation Through Feeder Lines 4. Hydrogen Spill 5. Spent Fuel Separator Failure 6. Hydrogen Explosion 7. Fuel Injector Electrical Discharge D. Vacuum and Impurity Control 1. Primary Vacuum Pump Failure 2. Primary Vacuum Containment Failure 3. Saturation of Cryopanel 4. Divertor Failure 5. Plasma Focusing in Divertor Slots 6. Collector Plate Loss of Coolant 7. Divertor Film Overheating 8. Divertor Heat Exchanger Leak 9. Loss of Divertor Coolant Flow E. First Wall 1. First Wall Failure 2. Issec Failure 3. Carbon Curtain Failure 4. First Wall Sputtering 5. Hydrogen Reactions F. Blanket 1. Loss of Inner Blanket Coolant Flow 2. Loss of Blanket Decay Heat Removal 3. Blanket Cracking 4. Blanket Melting 5. Helium Deposition 6. Corrosion of Niobium Window 7. Failure of Niobium Window 8. Hydrogen Embrittlement 9. Helium Permeation G. Primary Heat Transfer 1. Piping Failure 2. Pump Failure 3. Flow Coastdown 4. Coolant Plugging 5. Corrosion Product Activation 6. Helium Bubble Formation 7. Tritium Permeation 8. Tritium Buildup 9. Pressure Pulse from Magnetic Field Transient	H. Shielding 1. Neutron Dose 2. Gamma Dose 3. Tritium Release 4. Breach of Containment 5. Shielding Coolant Activation 6. Shielding Coolant System Failure I. Structure 1. Natural Disasters 2. Magnetic Structural Failure 3. Structure Irradiation Damage 4. Insulator Irradiation Damage 5. Bellows Irradiation Damage 6. Electrical Currents in Structures 7. Containment Integrity 8. Magnetic Forces J. Magnets 1. Loss of Magnet Coolant 2. Loss of Magnet Coolant Flow 3. Magnet Coolant Flow Blockage 4. Excess Heating 5. Shorting and Arcing 6. Air Leakage Into Magnet Dewar 7. Magnet Dewar Failure 8. Magnet Power Supply Shorting and Arcing 9. Helium Permeation 10. Superconductor Break K. Secondary Heat Transfer 1. Loss of Coolant Flow 2. Loss of Coolant 3. Heat Exchanger Leak 4. Loss of Working Fluid Flow 5. Corrosion Buildup L. Electricity 1. Loss of Offsite Power 2. Generator Malfunctions 3. Electric Discharge from Transformers 4. Energy Storage, System Malfunction 5. Switching Malfunctions M. Waste Disposal Accidents N. Instrumentation and Controls 1. Protective Instrumentation Malfunction 2. Functional Instrumentation Malfunction 3. Diagnostic Instrumentation Malfunction 4. Electromagnetic Interference O. Remote Maintenance and Repair P. Mining and Transportation 1. Lithium Mining and Extraction 2. Lithium Transportation 3. Radwaste Transportation 4. Tritium/Deuterium Transportation

ENVIRONMENTAL EFFECTS ANALYSES

A correlation exists between the acceptance of a new technology and the environmental effects incurred by that technology. Therefore, the extrapolation of known data plus the generation of data from existing technologies as they could apply to fusion reactor concepts are useful in guiding the engineering of systems for future fusion reactor designs.

FUSION REACTOR ENVIRONMENTAL EFFECTS

J. R. Young

The environmental analysis for the fusion reactor development program and the supporting reference documents have been published and distributed.

The analysis was made to determine the possible environmental effects of the first commercial fusion power plants, the research necessary to assure adequate ability to describe those effects, and the power plant designs that would result in minimum adverse environmental impacts. The fusion reactor design concepts being developed were analyzed to determine the interactions with the environment and the resultant impacts. A conservative (or pessimistic) approach was used by assuming that only current technology would be used for designing the plant subsystems that control releases to the environment. Yet, the analysis still shows that fusion reactors have potential for substantially lower total environmental effects than current fossil or fission power plants or than the estimated effects of fast breeder reactors.

The environmental analysis summarizes information developed and gathered at PNL, which includes much previously unsummarized background on fusion concepts and technical analyses. The details, the unique data and individual analyses used during preparation of the larger environmental analysis, are presented in the supporting series of reference topical reports.

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MANPOWER DEVELOPMENT

To promote a better understanding of fusion technology, PNL participates in educational programs offered through the Joint Center for Graduate Study in Richland, Washington. PNL staff are currently teaching a course sequence for the three quarters of the 1976-77 academic year as a part of the University of Washington's graduate study program in Nuclear Engineering. The first quarter included an introduction to plasma engineering and its application to controlled thermonuclear fusion. The second quarter, now ending, has emphasized blanket technology and environmental aspects.

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EDUCATIONAL DEVELOPMENT

B. F. Gore, A. B. Johnson, Jr., H. B. Liemohn, F. P. Hungate

A graduate course sequence on fusion technology with a balanced emphases of plasma and engineering aspects was approved for the Nuclear Engineering Curriculum of the University of Washington at the Joint Center for Graduate Study.^(a) The Center is operated in Richland, Washington, by Oregon State University, Washington State University and the University of Washington.

The sequence is being given routinely in three quarters starting with the 1976-77 academic year. The first quarter covered fusion plasma engineering and included an introduction to plasma theory and its application to controlled thermonuclear fusion. In the second quarter, currently ending, the emphasis has been on blanket technology and environmental aspects. The last quarter will cover candidate materials problems of principal fusion reactor conceptual designs such as first walls, coolant systems, blankets, shields, magnets, and heat exchangers.

^(a) Effort funded as a part of the Joint Center for Graduate Study Program

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