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**Pacific Northwest Laboratory
Report on Fusion Energy Research
October 1976 - December 1976**

January 1977

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

Battelle
Pacific Northwest Laboratories

BNWL-1939-6

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PACIFIC NORTHWEST LABORATORY
REPORT ON FUSION ENERGY RESEARCH
OCTOBER 1976 - DECEMBER 1976

January 1977

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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 October 1, 1976 through December 31, 1976. 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-1604.

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.

Report on Controlled Thermonuclear Reactor Technology, January through September 1975, BNWL-1939-1.

Report on Controlled Thermonuclear Reactor Technology, October through December 1975, BNWL-1939-2.

Report on Controlled Thermonuclear Reactor Technology, January through March 1976, BNWL-1939-3.

Report on Controlled Thermonuclear Reactor Technology, April through June 1976, BNWL-1939-4.

Report on Fusion Energy Research, July through September 1976, BNWL-1939-5.

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PACIFIC NORTHWEST LABORATORY
FUSION ENERGY RESEARCH
OCTOBER THROUGH DECEMBER 1976
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 capability of a conceptual, UO_2 -fueled, outside blanket to produce power, plutonium, and tritium in the Tokamak Engineering Test Reactor (TETR) were determined in a fusion-fission hybrid development study being made in cooperation with the University of Wisconsin. Plutonium production rates and tritium breeding performance were also evaluated for the conceptual TETR hybrid blanket using various technologically advanced fuels. In a PNL-PPPL study to develop a hybrid reactor using the Two Component Torus (TCT), structural and thermal stresses were analyzed for a double-wall design of the reactor first wall. Results indicate that cycling thermal stresses would be within acceptable limits with the use of room-temperature cooling water. Other efforts in the TCT design study are identifying the controlling failure mode of the first wall and evaluating the integrity of the fuel slugs under thermal cycling conditions.

The levels of actinide contamination were calculated for fissile fuel produced by fusion neutrons. The preliminary evaluation is that ^{233}U produced with a blanket containing ThO_2 would have a significantly greater magnitude of actinide contamination than ^{239}Pu produced with a blanket containing UO_2 . This difference and ^{239}Pu 's high fissile content appear to make ^{239}Pu the preferable product for breeding in hybrid blankets.

The present worth benefits for a fusion power plant and the present worth costs of fusion research and development were estimated as functions of the date when fusion power plants would become commercially available. Preliminary analysis indicates an optimum commercial introduction date (maximum benefit-to-cost ratio) before the year 2000.

A task was initiated to scope experimental needs and to develop design analysis tools for heat transfer and fluid flow in fusion reactor technology.

In the area of surface effects research, preliminary fluence data in (D,Be) neutron sputtering experiments at the University of California-Davis have indicated a sputtering yield of $\sim 8 \times 10^{-6}$ Atoms/n for gold sputtering targets. Equipment is being assembled to study the blistering effects of multiple-angle, helium ion irradiations using quasi-continuous energy spectra. In a study characterizing material transfer during different discharge

cleaning modes of UCLA's Microtor, graphite collector samples have been prepared and inserted in Microtor. Sputter depth profiles of oxygen, carbon, and iron found on the surface of a glass collector placed in Microtor showed that relative concentrations varied with depth in a way that could be related to Microtor's operation history.

In studies of the effect of radiation on mechanical properties, the radiation hardening threshold for MRC Marz grade nickel wire irradiated with (D,T) neutrons at 20°C was determined to be $\sim 2 \times 10^{16}$ n/cm². No hardening was observed for 316 stainless steel wire irradiated at a fluence of 6×10^{16} n/cm². Neutron radiation hardening data for PNL niobium wire showed two yield points, one similar to LLL niobium data, and another, higher yield point. Experiments are being planned to determine the reasons for this yield point discrepancy.

In materials development studies, some graphite cloths irradiated to a neutron fluence of 1×10^{22} cm⁻² EFF at 470°C accommodated a shrinkage approaching 20% in the axial direction of the fibers with no apparent deterioration. The third of three irradiations in a task to correlate damage production for graphites in fission and fusion reactor spectra has been completed. Methods of cleaning the surface of a ¹⁴C-labeled carbon sample are being developed prior to neutron sputtering measurements. The carbon and graphite studies are aimed at the development of a liner material for placement between the fusion reactor plasma and the first structural wall. In evaluation of electrical insulators for fusion reactor applications, work has continued in the development of acoustic emission techniques for determining prebreakdown behavior and failure mechanisms in dielectric materials. Research objectives were considered for experiments to determine the effects of porosity and helium buildup on the dielectric breakdown strength of electrical insulators. Sample and equipment preparations are in progress.

The available fusion reactor design concepts have been analyzed to determine the probable interactions with the environment and the resultant environmental effects. The environmental analysis prepared by PNL for the fusion program has been published along with eighteen supporting reference documents. PNL has begun an effort this quarter to define the scope of a fusion reactor safety program plan and to identify the basic technology necessary to prepare a comprehensive safety analysis report on a fusion power plant. Contacts were initiated with other laboratories engaged in fusion safety research, and an investigation was started to examine the relevance of existing fusion reactor safety criteria to the fusion program.

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 present quarter includes an introduction to plasma engineering and its application to controlled thermonuclear fusion.

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 these engineering problems be given attention early enough in the fusion program to properly identify and solve them so that they do not limit the development and implementation of fusion reactors in commercial power production. Work reported this quarter includes continued design and development studies for the fusion-fission hybrid reactors, an economic analysis to develop information needed to establish an optimum R&D schedule for magnetically confined fusion systems, a new task on the development requirements of heat transfer and fluid flow, and a task on the development of nuclear data standards for DMFE programs.

In cooperation with the University of Wisconsin PNL is defining and evaluating a fissile breeding 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. With these constraints, uranium dioxide (UO_2) has been identified as the fertile breeding material for production of plutonium, and the capability of a conceptual, UO_2 -fueled, outer blanket to produce power, plutonium, and tritium has been determined. Plutonium production rates and tritium breeding performance were also assessed in calculations using the conceptual TETR hybrid blanket with various technologically advanced fuels - UMo , UC , and U_3Si .

A coordinated effort between PNL and the Princeton Plasma Physics Laboratory is being completed on the conceptual design of a fission-fusion hybrid reactor based on a Two Component Torus (TCT). Preliminary analyses indicate that the use of room-temperature cooling water may be possible in a double wall concept introduced by PNL as a firstwall design. Low temperature differentials through the wall ($\sim 25^\circ\text{C}$) and between the inner wall surface and mean fluid temperature ($\sim 9^\circ\text{C}$) in this design are within acceptable limits of cycling thermal stresses. Steady-state analysis of the converter region using a modified computer modeling code indicates that satisfactory cooling of the converter region (which consists of 316 SS tube, UMo annular slugs, and helium coolant) can be accomplished for lengths of one-sixth of the major circumference. Other modeling programs have been developed to identify the controlling failure mode of the first wall and to evaluate the integrity of the fuel slugs under thermal cycling conditions.

PNL has made a preliminary evaluation of the quality of fissile fuel produced by fusion neutrons, based on the degree to which the fuel is contaminated with undesirable isotopes. Calculations were made for a blanket containing either ThO_2 or UO_2 to produce ^{233}U or ^{239}Pu , respectively. The magnitude of actinide contamination appears to make ^{233}U production economically unattractive, while high fissile content and the low quantity of waste products appear to make plutonium the preferable product for breeding in hybrid blankets.

DMFE comments were received on a revised draft of the PNL document describing the design, operation, and economic information that should be included in fusion system design studies. A section on environmental information requirements is being removed, and an example of costing for a typical power plant is being added as a result of those comments.

Economic analysis by Vashon and Nichols of the University of Texas indicates that accelerated R&D expenditures through the use of parallel programs could bring about faster achievement of the technological milestones necessary for introducing magnetically confined fusion technologies. This would mean increased R&D costs but earlier introduction of potentially cost-saving fusion power plants. Conversely, an extended R&D period would have greater cost efficiency in the research process but would delay the possible cost benefits from the fusion power plants. The present worth benefits and R&D costs were estimated for various dates of power plant introduction. Preliminary analysis indicates an optimum introduction date (maximum benefit-to-cost ratio) before the year 2000, regardless of the discount rate used to estimate the present worth benefits and R&D costs.

Studies were initiated this quarter to develop design analysis tools and to scope experimental needs for the definition and assessment of heat transfer and fluid flow development requirements for the base technology of fusion reactors. Current documentation on The Next Step/Experimental Power Reactor designs have been identified.

In a task to participate in the development of current best-estimate nuclear data files for DMFE technology programs, a series of meetings were attended to discuss data needs.

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 is being 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 is underway to evaluate 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. These studies are nearing completion with the issuance of draft topical reports targeted for the next quarter.

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 SCOPING STUDY

V. L. Teofilo, R. T. Perry, B. R. Leonard, Jr., M. A. McKinnon

The TETR design study of Tokamak to produce a high neutron flux for engineering testing has recently been completed.⁽⁴⁾ The objective of the TETR-Hybrid study, being done in cooperation with the University of Wisconsin, is to define and evaluate a fissile breeding outside blanket for TETR (see Figure 1) based upon near-term technological developments that could be implemented in the middle to late 1980's. With this principal objective in mind, the fertile breeding material identified for the maximum production of plutonium (Pu) is uranium oxide (UO_2).

In addition to the evaluation of UO_2 , the relative breeding performance of technologically advanced fuels (e.g., metals, carbides, and silicides) has been evaluated for the defined outer TETR blanket. In order to estimate the requirements for upgrading TETR for possible commercial Pu and power production, the capability of the defined UO_2 -fueled outer blanket to breed tritium (T) and to produce power has been determined.

COMPUTATIONAL METHODOLOGY

The ANISN code has been run in a one-dimensional, vertical cylindrical geometry. Thirty neutron groups in an S_8 - P_3 calculation was employed. The neutron source was space

FIGURE 1. Cross Section View of TETR⁽⁴⁾

and energy dependent. The blanket schematic used for the calculations is shown in Figure 2. Because of cross section accessibility, stainless steel was substituted from the initial TETR design for insulation and the coils. The boundary conditions used were reflective for inner side of the torus and vacuum for the outer side of the torus.

FIGURE 2. Blanket Schematic

The first wall and coolant channels on the outside blanket were estimated to be 5 cm thick containing 55% SS. In each case, a volume fraction of 10% SS was used as an estimate of the cladding and structural material requirements.

Several different fuels, UO_2 , UMo , UC , and U_3Si , were used in various calculations, the object of which was to determine the maximum ^{239}Pu production per source neutron through a volume fraction parameterization of fuel, structure, and coolant. Natural uranium was employed in each of the fuels. The ^{235}U and ^{238}U cross sections were spatial self-shielded using a geometry from previous calculations.

BLANKET LAYOUT

A preliminary layout for a practicable modular blanket design for containing UO_2 fuel pins was identified. For the outer blanket region of TETR this concept assumes a 65-cm blanket the thickness of which is restricted by a minimum 10-cm proximity to any normal poloidal field coil (see Figure 3). The modules are 3.5 cm from the 316 SS vacuum wall in order to permit adequate volume for helium cooling the first wall. They are divided into modular sections which contain 316 SS clad, helium-cooled, UO_2 fuel pins and which are separated by both the poloidal field coil on the horizontal midplane, as well as the beam ports whose angle of penetration prohibit any uniform modular design between adjacent ports except in only one 45° sector (see Figure 4).

FIGURE 3. Vertical Cross Section of TETR Showing Hybrid Blanket Modular Layout

FIGURE 4. Top View of the TETR⁽⁴⁾

UO₂-FUELED BLANKET

For the geometry of the 65-cm conceptual modular blanket design, the ANISN code was run for the volume fraction parameterization to determine the maximum Pu production rate using UO₂ fuel. The resulting Pu production rates, which neglect depletion, are tabulated in Table 1.

TABLE 1. UO₂-Fueled TETR-Hybrid Blanket-Pu Production

vol% UO ₂	vol% 316 SS	vol% Void	Fissions per Fusion	Pu Atoms Produced per Fusion	Pu Production Rate, kg/yr	Average Blanket Power, MWt
0.2	0.1	0.7	0.1784	0.700	394	434
0.3	0.1	0.6	0.2209	0.923	520	542
0.5	0.1	0.4	0.2668	1.160	563	637
0.7	0.1	0.2	0.2879	1.260	709	680
0.9	0.1	0	0.2993	1.307	736	702

In computing these production rates, the effective neutron source strength of the TETR plasma was reduced from the 214 MW instantaneous neutron power (9.51×10^{19} n/sec) to

to 7.78×10^{19} n/sec due to losses through the divertor channels and beam ports as well as the lack of fertile-fueled submodules in the regions of the poloidal field coil on the horizontal midplan and beam ducts. In addition, the power duty cycle (0.8275) and plant factor (0.7) were taken into account. The blanket power, also averaged over a duty cycle, is due to fissions in the fuel and 60% of the effective 14-MeV source neutrons which are assumed to lose all of their energy in the blanket. Gamma radiation due to activation is neglected. We note the performance of the 70 vol% UO_2 -fueled blanket as producing about 700 kg Pu/yr while generating about 700 MWt of power in the blanket alone for an effective multiplication of ~ 7 for the 100 MW of neutron power which enter it.

ADVANCED FUELS

In order to assess the relative Pu production rates of fertile fuel, which could be available with the advances in development and technology expected by the 1990s, a volume fraction parameterization was performed with the ANISN code and the geometry of the conceptual TETR hybrid blanket for various advanced fuels (see Table 2).

TABLE 2. Advanced Fueled TETR-Hybrid Blanket-Pu Production

Fuel	Volume Fractions			Fissions per Fusion	Pu Atoms Produced per Fusion	Pu Production Rate, kg/yr	Average Blanket Power, MWt
	Fuel	316 SS	Void				
UMo	0.9	0.1	-	0.3519	1.383	779	812
	0.7	0.1	0.2	0.3420	1.348	759	791
	0.5	0.1	0.4	0.3237	1.271	716	754
	0.3	0.1	0.6	0.2816	1.061	597	666
	0.2	0.1	0.7	0.238	0.844	475	575
UC	0.9	0.1	-	0.3621	1.525	859	833
	0.7	0.1	0.2	0.3504	1.480	833	808
	0.5	0.1	0.4	0.3286	1.385	780	764
	0.3	0.1	0.6	0.2790	1.138	641	662
	0.2	0.1	0.7	0.2293	0.890	501	559
U_3Si	0.9	0.1	-	0.3907	1.591	895	893
	0.7	0.1	0.2	0.3780	1.542	869	866
	0.5	0.1	0.4	0.3550	1.438	810	818
	0.3	0.1	0.6	0.3031	1.173	660	710
	0.2	0.1	0.7	0.2512	0.9168	516	625

As compared to the UO_2 -fueled base case, the Pu and power producing rates of the 70 vol% advanced fueled blanket cases are approximately 10% greater for UMo fuel, 20% greater for UC fuel, and 25% greater for U_3Si fuel.

TRITIUM BREEDING

The tritium breeding performance of the defined TETR-Hybrid blanket fueled with UO_2 or the best advanced fuel (U_3Si) was computed for various T breeding volumes outside the fertile to fissile convertor region but within the blanket modules. The breeder consists of 25% Li_2O having a natural (7%) 6Li isotopic content, and 45 vol% carbon with 10 vol% 316 SS and 20 vol% void for helium coolant. The convertor contained the base case 70 vol% of UO_2 fuel, with 10 vol% 316 SS and 20 vol% void. The results are tabulated in Table 3.

TABLE 3. TETR-Hybrid Blanket-Pu Production/T Breeding

<u>Fuel</u>	<u>Pu Breeder</u>	<u>T Breeder</u>	<u>T Atoms Produced per Fusion</u>	<u>Fissions per Fusion</u>	<u>Pu Atoms Produced per Fusion</u>	<u>Pu Production Rate, kg/yr</u>	<u>Average Blanket Power, Mwt</u>
UO_2	39	26	0.1145	0.2731	1.1244	633	657
	26	39	0.2852	0.2472	0.9201	518	608
	13	52	0.5713	0.1822	0.5416	305	490
U_3Si	39	26	0.1090	0.3658	1.412	795	846
	26	39	0.3099	0.3409	1.199	675	804
	13	52	0.6002	0.2672	0.7651	431	657

The maximum T breeding ratios obtained, about 0.6, correspond to more than a 50% reduction in the Pu production rate, as expected. It is possible to increase this ratio to about 1.0 by both enriching the 6Li isotopic content of the Li_2O and running natural liquid lithium through the inner blanket. Whether this can be accomplished and at the same time produce Pu at a rate of 450 kg/yr, corresponding to one PWR charge fueled with pure, first cycle Pu, will be evaluated during the next phase of this study.

TCT HYBRID CONCEPTUAL DESIGN STUDIES

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

The objective of this study is to develop a conceptual design of a hybrid based on current Two Component Torus (TCT) technology. If the concept appears promising, a reference design will be developed. The ground rules are: 1) to optimize on production of fissile material, consistent with the requirement that a tritium breeding ratio in excess of one is obtained, 2) to constrain the technology to that available to construct a machine in the early 1990s, and 3) to minimize the capital costs.

The Tokamak Fusion Test Reactor (TFTR)⁽²⁾ will be the first fusion device using magnetic plasma containment to yield neutrons from the fusion process. The TFTR uses the TCT concept for plasma heating and containment. While the concept is expected to yield a near Lawson Plasma, it is not expected to extrapolate to a net exporter of electrical power. One potentially feasible way to commercially utilize this technology, however, is to design a hybrid

device which uses the neutrons from the TCT device to breed fissile material for use in fission reactors and to produce some power directly. A coordinated effort between scientists and engineers from PPPL and PNL is underway to evaluate the potential of this concept and develop a conceptual design.

A double-walled configuration of 316 SS tube was introduced by PNL and identified as a primary candidate for the first wall of the toroidal plasma chamber. It was assumed to contain a maximum of about 10 vol% SS. Preliminary structural and thermal hydraulics analysis have been performed. Computer analyses used the computer code BOSOR4 for buckling under external pressure and AXISOL⁽⁵⁾ for internal pressure between the double walls. The analyses indicate that the use of room-temperature cooling water may be possible. The cooling water temperature rise is 3.5°C for a mean velocity of 5.1 m/sec. The maximum temperature difference through the wall is approximately 25°C and the temperature difference between the inner wall surface and mean fluid temperature is about 9°C. These temperature differentials are within acceptable limits of cycling thermal stresses.

In order to identify the controlling failure mode of the first wall, an ANSYS program model⁽⁶⁾ has been set up with the following capabilities to represent a segment of the toroidal chamber:

- two walls connected by stiffeners with 4.0-in. spacing
- different pressures at each surface
- meridionally varying temperature on both walls.

This model differs from the AXISOL model by virtue of being a shell representation of a toroid structure. With the inclusion of radiation swelling effects, it will provide stress levels to assess the significance of:

- fatigue
- crack propagation
- creep rupture
- embrittlement.

The fabrication aspects of the first wall have been explored and a draft report is being prepared.

The PNL COBRA⁽⁷⁾ code has been modified to model the converter region. This region is modeled as a single 316 stainless steel tube (SST), uranium molybdenum annular slugs, and helium coolant. Thermophysical properties and a range of possible configurations have been identified. Steady-state analysis of the converter region using the modified COBRA code indicates satisfactory cooling can be accomplished for lengths of one-sixth of the major circumference. Preparation of the COBRA code for transient calculations is currently underway in order to determine the change in pressure tube and fuel temperatures during the shutdown portion of the power cycle.

Two axisymmetric stress models have been developed to evaluate the integrity of the fuel slugs under thermal cycling conditions. One model investigates a section in the middle of the slug where conditions are axially uniform, and the second model investigates the ends of the slug where conditions are not axially uniform. A steady-state thermal condition is being applied.

Thermophysical properties for the blanket region have been collected. Several heat transfer codes have been examined for suitability and one has been selected which will account for change of phase of the molten salt. Preliminary steady-state calculations suggest the need for several coolant paths inside the blanket to maintain acceptably low temperatures.

HYBRID STUDIES - THE QUALITY OF FISSILE FUEL BRED IN A FUSION REACTOR BLANKET⁽⁸⁾

B. R. Leonard, Jr., U. P. Jenquin

A preliminary evaluation has been made of the quality of fissile fuel produced using fusion neutrons. The quality of the product is expressed as the degree to which the product is contaminated with undesirable isotopes. Calculations were made for a blanket containing either ThO_2 or UO_2 to produce ^{233}U or ^{239}Pu , respectively. The rates of production of the product and by-product isotopes were tracked as a function of the irradiation time. For perspective on the quality of the fuel produced, the U and Pu generated in the hybrid reactor are compared to the same products produced in fission reactors; namely plutonium produced in LWRs and LMFBRs, and ^{233}U produced in High Temperature Gas Cooled Reactors (HTGRs). The by-products were examined to determine the quantities and characteristics of radioactive waste associated with hybrid fissile fuel production.

The hybrid reactor with a ThO_2 blanket produces approximately 1.4 kg of $^{233}\text{U}/\text{MT}$ of thorium per 100 days of irradiation. The fissile content of the bred uranium is nearly 100%, whereas HTGR fuel on an equilibrium cycle is ~70% fissile.

The ^{233}U fuel produced in a hybrid thorium blanket contains substantially higher concentrations of ^{232}U compared to ^{233}U produced in HTGRs. One of the isotopes in the ^{232}U decay chain is ^{208}Tl which emits a high-energy gamma ray. Shielding of this gamma ray is required to protect personnel involved in postreprocessing operations such as fabrication and transportation of the recovered uranium. In order to lower the ^{232}U concentration to maximum levels expected in the HTGR cycle (0.12% ^{232}U), the cycle length would have to be less than 70 days. However, under these conditions a higher fuel-cycle cost would result from increased blanket fabrication and reprocessing through-put per unit of recovered product. Although it was not within the scope of this study to quantify the significance of high ^{232}U concentrations in the fission fuel cycle, this may be a significant drawback to breeding ^{233}U in hybrid reactors.

Another consideration in the Th- ^{233}U cycle is the production of actinide by-products and the implications of these on management of radioactive residues. The high-energy neutrons generate approximately 0.4 kg of these isotopes/kg of product ^{233}U . The characteristics of these isotopes are such that they will require long-term waste management similar to that required for high-level waste from reprocessing fission reactor fuels.

The hybrid reactor with a UO_2 blanket produces 1.4 kg of ^{239}Pu /MT of uranium/100 days of irradiation. The fissile content of the bred plutonium is greater than 95%. This is much higher than the 69% fissile plutonium obtained from the LWR uranium cycle or the 75% fissile plutonium obtained from the equilibrium LMFBR cycle. Since the high fissile plutonium would have to be used in a different manner than that proposed for LWR-produced plutonium, a study of the complete fuel cycle is necessary to ascertain the true economics of high fissile plutonium.

A major constraint in plutonium fuel fabrication is due to neutron emissions from (α, n) reactions with oxygen and light element impurities. The alpha activity results primarily from ^{238}Pu decays; therefore, ^{238}Pu is an undesirable contaminant. Hybrid-produced plutonium contains less ^{238}Pu than LMFBR-produced plutonium or LWR-produced plutonium; hence, the hybrid plutonium is more valuable with respect to ^{238}Pu content.

Plutonium-236 is an isotopic contaminant with a half-life of 2.8 years and rapidly builds up the ^{232}U decay chain with its source of high-energy gamma rays. As with ^{232}U in the Th- ^{233}U cycle, this places additional personnel shielding requirements on fuel handling operations subsequent to fuel recovery. The ^{236}Pu concentration in hybrid-produced plutonium is three orders of magnitude larger than in LMFBR- and LWR-produced plutonium.

Approximately 0.14 kg of by-product actinides are produced per kg of plutonium recovered from a UO_2 hybrid blanket. Essentially all of this by-product waste consists of ^{237}Np . There is a factor of three more ^{237}Np produced in a hybrid than that generated in an LWR and a factor of four more than that generated in an LMFBR for equivalent masses of plutonium. On the other hand, there is virtually no americium or curium produced in a hybrid, whereas both the LWR and the LMFBR produce 0.07 kg of these elements for each kg of plutonium produced. As with the Th- ^{233}U cycle, the characteristics of the actinide waste are such that they will require long-term waste management similar to that required for high-level waste from fission reactor fuels.

The results of these calculations indicate that plutonium breeding in hybrid blankets may be preferable to ^{233}U breeding. The contaminants of ^{233}U appear to be of a magnitude to make the product economically unattractive while the high fissile content and the low quantity of waste by-products appear to make plutonium attractive. However, in order for the hybrid breeder to be competitive in the economic market place, the value of the fissile fuel it produces must offset its capital and operating costs. The entire fuel cycles require a considerably more detailed study to determine the synergistic relationships with specific fuel cycles, net energy balance, and radioactive waste management requirements.

The conclusion that ^{233}U breeding in a hybrid blanket is less attractive, is predicated on the assumption that a molten salt processing system is not developed.

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 draft 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. Provision 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 revised 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 is being removed, and an example of costing for a typical power plant is being added.

ECONOMIC REGIMES

D. E. Deonigi, R. L. Engel

During the past year, Vanston and Nichols⁽⁹⁾ of the University of Texas have estimated the costs of achieving important technological milestones for magnetically confined fusion energy systems. Of particular interest is that these estimates also include the impact of rate of expenditure. In general, it is estimated that accelerated expenditures through the use of parallel programs can produce earlier achievement of the technology milestones. This would lead to the possible introduction of magnetically confined fusion technologies at an earlier date; however, greater efficiency may be achieved in the research process by extending the research and development period, which would reduce the total expenditure at the expense of later introduction of the potentially cost-saving advanced fusion power plants.

Figure 5 shows the cumulative present worth cost of research and development estimated by Nichols as a function of availability date at a 7.5%/annum and a 10%/annum discount rate. The two discount rates allow direct comparison with calculations made using the linear programming model of the U.S. electrical generating system, which estimates the benefits of fusion power plants as a function of availability date.

The benefits of fusion power are estimated by using a scenario for the development of competing energy sources consistent with that presented in the recent LMFBR cost-benefit

FIGURE 5: Cumulative Present Worth Cost of R&D as a Function of Fusion Power Plant Availability Date

study (ERDA 38).⁽¹⁰⁾ However, some modifications include higher LMFBR cost (LWR plus \$100/kWe) and a 12-year doubling time consistent with the utility opinion collected in a survey of the Electric Power Research Institute fusion committee. This scenario is, therefore, generally consistent with that presented in previous quarterly reports dealing with the hybrid power plant.

Figure 6 illustrates the calculated present worth benefits for a fusion power plant having costs equal to the LMFBR plus \$30/kWe. This figure includes cases for introduction at various dates with discount rates of 7.5% and 10% consistent with the cost estimates in Figure 5. The benefit/cost ratios (estimated cost benefits/estimated R&D costs) are shown in Table 4 for the same discount rates in the same years. The R&D costs in Figure 5 are extrapolated beyond the year 2000 data supplied by Nichols, based on the assumption that at some point earlier than 2000 (~1990) the R&D costs would be extremely large due to the need for many parallel efforts. Benefit growths are not as fast from 1990 to 2000 as from 2000 to 2010 because the fusion plant does not compete as well in the early years due to the low cost fuel still available from coal and nuclear plants. These results indicate that the optimum time of introduction is before year 2000 and that the point of maximum benefit-cost ratio seems to be independent of the discount rate used. Benefits increase at uniform percentage rates for every ten years of earlier availability. The network analysis performed by Nichols as well as the benefit analysis has some uncertainties. Nichols indicates that an uncertainty of $\pm \$1$ billion is typical for R&D costs, while benefit simulation indicates a $\pm 50\%$ uncertainty.

FIGURE 6. Calculated Present Worth Benefits for a Fusion Power Plant as a Function of Plant Availability Date

TABLE 4. Present Worth Benefit/Cost (B/C) as a Function of Availability Date

Availability Date	7.5% Discount Rate			10% Discount Rate		
	R&D Costs, \$ Billions	Benefits, \$ Billions	B/C	R&D Costs, \$ Billions	Benefits, \$ Billions	B/C
1990 (Estimate)	13	33.0	2.6	8.5	12.0	1.4
2000	4.85	19.0	3.9	3.62	5.9	1.6
2010	1.96	4.4	2.2	1.43	1.4	1.0
2020	0.88	0.9	1.0	0.56	0.4	0.7

The very preliminary analysis presented above is an attempt to develop information to establish an optimum rate at which R&D for magnetically confined fusion systems should be pursued. This analysis indicates that introduction of the fusion power plants prior to the year 2000 would produce the greatest benefits, especially when compared with introduction in the year 2020. Of particular significance is the fact that this conclusion seems to be independent of the discount rate used in the analysis.

During the next quarter, a more detailed and refined analysis of this particular situation will be pursued.

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. Work was initiated on developing design analysis tools for heat transfer and fluid flow and to scope experimental needs. 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

A. M. Sutey, D. T. Aase

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.

This task was initiated this quarter. For FY77, the scope of work will emphasize heat transfer and fluid flow methods development and implementation in support of Experimental Power Reactor (EPR) and The Next Step (TNS) designs. To date, current documentation on TNS/EPR designs are being reviewed and the appropriate contacts in each design organization have been identified.

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 series of meetings were attended to discuss data needs. A meeting of the Cross Section Evaluation Working Group's (CSEWG) Nuclear Model Codes Subcommittee was held at the Hanford Engineering Development Laboratory, October 13-14 to compare nuclear model calculations by five laboratories on ^{59}Co cross sections. These comparisons are relevant to the estimation of p and α production to 40 MeV.

A specialist's meeting on Nubar (number of neutrons per fission) was held at BNL October 26 as a Standards Subcommittee meeting of CSEWG to arrive at recommended values for fissile isotopes for ERDA's Evaluated Nuclear Data File/B-V. This was followed by the CSEWG and Standards Subcommittee meetings at BNL October 27 and 28. A presentation was made to the Data Testing Subcommittee on PNL-evaluated thorium data for ENDF/B-V. A presentation was also made to CSEWG on long term radioactive dose from retired fission plants and the resultant need for a special file of decay data for LWRs, LMFBRs, and fusion reactors. The Standards Subcommittee reviewed the proposed agenda for the Symposium on Nuclear Data needs to 40 MeV with M. Bhat, of BNL, proposing changes in the agenda suggesting invited speakers in certain areas.

At the Washington, DC ANS meeting November 14-19, an attempt was made to influence reversal of the position of the ANS Publications Committee on adoption of an ASTM standard on SI units. This is a problem for communication in nuclear physics. A reversal was effected, at least in large part. Also at the meeting a discussion on the status of thorium data was held with Bohn and Carpenter of ANL, who are interested in both fission and hybrid application.

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 parameters for sputtering and blistering at materials surfaces, 2) correlate changes in mechanical properties produced by light ions and fusion neutrons, 3) evaluate carbons and graphites for use as possible first-wall liner materials, and 4) determine the effects of porosity and helium buildup on the dielectric breakdown strength of electrical insulators.

PNL surface science work this quarter has included studies of fast-neutron sputtering, blistering effects of polyenergetic helium ions, and in-reactor material transfer during different discharge cleaning modes. Nuclear counting of the dosimeter foils used during the (D,Be) neutron sputtering experiments at the University of California-Davis has furnished preliminary fluence data which imply a sputtering yield of about 8×10^{-6} atoms/n from gold sputtering targets. A new target which will allow a significant increase in useful neutron flux at the University of California's cyclotron has been purchased from LLL and is slated for use early in 1977. In a study of the blistering effects of continuous or quasi-continuous helium ion energy spectra, a programmable energy degrader and equipment allowing simultaneous multiple-angle irradiations have been designed and are being assembled. Graphite collector samples for the discharge cleaning program have been prepared and are being inserted in Microtor at UCLA to study material transfer during different discharge cleaning modes. Preliminary examination of a glass collector that was in Microtor during many cycles and cleaning modes showed large amounts of oxygen, carbon, and iron on its surface; sputter depth profiling showed that relative concentrations varied with depth in a way that could be related to Microtor's operation history.

Mechanical properties studies continue at PNL to correlate the microstructures and radiation hardening produced by light ions and fusion neutrons. The light ion irradiation apparatus, helium flow system, and data readout equipment have been transported to the University of Washington Nuclear Physics Laboratory where beam diagnostics are planned for January of 1977 with proton irradiations for the $p^+/(D,T)$ neutron correlation experiment to follow. The radiation hardening threshold for MRC Marz grade nickel wire irradiated with (D,T) neutrons at 20°C is approximately 2×10^{16} n/cm². Neutron radiation hardening data for PNL niobium wire showed a lower yield point, similar to LLL niobium data, but also an upper, enhanced yield point for PNL niobium. Further experiments are being planned to clarify the reasons for this upper yield point discrepancy. A cyclic irradiation creep experiment is being planned to follow the light ion/fusion neutron correlation experiment. The experiment will study the creep behavior of fusion reactor candidate materials under simulated operating conditions.

Studies have continued to evaluate radiation damage, sputtering, and outgassing for carbons and graphites which could possibly be useful in preventing structural damage by shielding the first wall from the plasma. Preliminary measurements of thin graphite cloth samples irradiated to a neutron fluence of $1 \times 10^{22} \text{ cm}^{-2}$ at 470°C indicate shrinkages approaching 20% have occurred in the axial direction of the fibers. Nevertheless, some types of cloth accommodated this shrinkage with no visual deterioration. All three irradiations in the task to correlate damage production for graphites in fission and fusion reactor spectra have been completed, but temperature excursions during the third irradiation may mean that it will have to be redone. Methods of cleaning the surface of a carbon sample labeled with ^{14}C are being developed prior to the start of neutron sputtering measurements. Equipment problems with the graphite degassing apparatus have been eliminated and measurements are now in progress.

Work has continued in the development of acoustic emission techniques for determining prebreakdown behavior and failure mechanisms in electrical insulators for fusion reactor applications. Research objectives were considered for experiments to determine the effects of porosity and helium buildup on the dielectric breakdown strength of insulators. Some specific insulators and a helium introduction method were chosen for initial investigation. Sample preparation and setting up of the experimental system are in progress.

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

D. L. Styris, M. T. Thomas

Progress is reported this quarter for three programs being conducted by PNL surface science staff: (D,Be) neutron irradiations are being used in determining neutron sputtering data; the blistering effects of polyenergetic He^+ irradiations of niobium samples are being studied; and material transfer during different discharge cleaning modes is being examined in Microtor at UCLA.

Nuclear counting of the dosimeter foils used during the neutron sputtering experiments at the (D,Be) neutron source of the University of California-Davis is still in progress. Data analysis techniques used to determine neutron fluences at (D,Be) sources are those developed by Dr. M. Saltmarsh of ORNL. Preliminary results indicate that for each of the three experiments the gold sputtering target was irradiated to a fluence (E_n greater than ~ 2 MeV) of $\sim 1.3 \times 10^{17}$, $\sim 1.7 \times 10^{17}$, and $\sim 2.1 \times 10^{17}$ n/cm², respectively. These preliminary fluence data imply a forward sputtering yield for (D,Be) neutrons (40-MeV deuterons) of about 8×10^{-6} atoms/n. The useful neutron flux at the University of California-Davis's cyclotron could be increased significantly if the targets were closer to the Be and a higher deuteron current were used. The new water-cooled target purchased from LLL allows both objectives to be accomplished. This new target should be ready for use sometime early in 1977.

Our helium blistering program is designed specifically to study the effects of continuous or quasi-continuous helium ion energy spectra. Since the success of the preliminary experiments with a simple degrader foil, a programmable energy degrader has been designed, and the assembly is near completion. In addition, a faraday cup-sample configuration which allows simultaneous multiple-angle irradiations at temperatures up to 1000°C has been designed, the assembly of which is also near completion.

The emphasis in the discharge cleaning program has been in preparing collector samples for insertion in Microtor at UCLA. Surfaces of pyrolytic graphite have been characterized and have been found to be very stable in air if handled correctly. This stability makes

pyrolytic graphite an ideal collector for discharge cleaning experiments where the analysis must be performed at a facility not connected to the Tokamak. A series of graphite samples have been mounted in stainless steel holders and are being inserted in Microtor to study material transfer during different discharge cleaning modes.

A glass collector that has been in Microtor during many cycles and various cleaning modes has been examined using Auger spectroscopy. Preliminary data showed large amounts of oxygen, carbon and iron on the surface. Sputter depth profiling showed that the relative concentration varied with depth in a way which could be related to Microtor's operation history.

A representative from PNL's Surface Science group was represented at the Plasma Wall Interaction Symposium held in Julich, Germany, in October. Visits were made to the laboratories at Julich and Garching, and the Harwell and Culham laboratories in England.

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 neutron 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

It is desirable to assess the ability of light ions to simulate fusion neutron irradiation effects on mechanical properties before the advantages of charged particle irradiations are pursued in irradiation damage studies. Of the large number of charged particles available for such damage studies, 16 MeV protons (p^+) appear the most attractive because: 1) there is some evidence that the damage structure produced with 16 MeV p^+ is similar to 14 MeV neutron damage; 2) displacement rate effects can be minimized by real-time experiments while accelerated tests can be performed at 20 to 50X that of a fusion reactor first wall; 3) 16 MeV p^+ can penetrate a reasonably thick (~ 0.25 mm) specimen, thereby minimizing contamination from the bombarding ion and the chemical environment; 4) contamination from the accelerator is eliminated by isolating the experimental chamber from the accelerator beam line by a thin metal window.

The near term objective of this program is to correlate the microstructures and radiation hardening produced by light ions and fusion neutrons. To accomplish this objective the microstructure and yield strength change of Ni, 316 SS, and Nb irradiated with 16 MeV p^+ and (D,T) and (D,Be) neutrons are being evaluated. The second objective of this program is to study the creep behavior of fusion candidate materials under simulated fusion reactor operating conditions.

LIGHT ION/FUSION NEUTRON CORRELATION EXPERIMENT

The light ion/fusion neutron correlation experiment is expected to produce data which will allow a comparison between the damage efficiencies of 16 MeV p^+ and (D,T) and (D,Be) neutrons. Data for correlating (D,T) and (D,Be) neutron damage will also be produced.

Light Ion Irradiations

The light ion irradiation apparatus, helium flow system, and data readout equipment has been transported to the University of Washington Nuclear Physics Laboratory. Preirradiation set-up is in progress with beam diagnostic tests planned for January 1977. Proton irradiation

for the $p^+/(D,T)$ neutron correlation experiment (see Table 5) will begin following satisfactory beam diagnostic experiments. These include calibration of the faraday cup in a 760 torr helium atmosphere which is used for sample cooling. The experiments listed in Table 5 are planned for completion by the end of this fiscal year.

TABLE 5. Summary of Light Ion/Fusion Neutron Correlation Experiment

Experiment	Particles	Flux	Proton Fluence, p^+/cm^2	Neutron Fluence, n/cm^2	Temp., °C	Material	Proton Beam time, hr
$p^+/(D,T)$	14.8-MeV	$10^{12}/\text{cm}^2\text{-sec}$	3×10^{16}		20, 200	Ni, 316 SS, Nb	2×8.5
	Neutrons		6×10^{16}	6×10^{16}	20, 200		2×17
	and		1×10^{17}	1×10^{17}	20, 200		2×28
	16-MeV p^+	$(0.2 \text{ } \mu\text{a}/\text{cm}^2\text{-}p^+)$		2.5×10^{17}	20		
				5.0×10^{17}	20		
$p^+/(D,Be)$	(D,Be) n	$10^{13}/\text{cm}^2\text{-sec}$	1×10^{17}	1×10^{17}	20-50	Ni, 316 SS, Nb	2
	and		5×10^{17}	5×10^{17}	20-50		10
	16-MeV p^+	$(2 \text{ } \mu\text{a}/\text{cm}^2\text{-}p^+)$	1×10^{18}	1×10^{18}	20-50		20
High Flux	16-MeV p^+	$4 \times 10^{13}/\text{cm}^2\text{-sec}$	5×10^{17}	---	200, 500	Ni, 316 SS, Nb	2.5
			1×10^{18}	---	200, 500		5.0
		$(8 \text{ } \mu\text{a}/\text{cm}^2)$	4×10^{18}	---	200, 500		2.0
				---	200, 500		

Fusion Neutron Irradiations

Radiation hardening data for Ni and 316 SS is shown in Figure 7 where it can be seen that the nickel-wire data (PNL-nickel) is similar to the Lawrence Livermore Laboratory (LLL) copper data, while the hardening threshold for PNL-316 SS wire irradiated with (D,T) neutrons at 20°C is above $6 \times 10^{16} \text{ n/cm}^2$. Radiation hardening data for niobium is shown in Figure 8 where it can be seen that the lower yield point of the PNL niobium wire data is similar to the LLL niobium data.

FIGURE 7. Comparison of $\Delta\sigma_{ys}$ versus Neutron Fluence for Fusion and Fission Neutron-Irradiated Nickel, Copper, and 316 SS

FIGURE 8. Comparison of $\Delta\sigma_{ys}$ for Niobium Wire (PNL) and Sheet (LLL) Irradiated with Fusion Neutrons at 20°C

An enhanced yield point has been found for the PNL niobium while LLL has not observed any yield point. This difference also exists for the unirradiated niobium. Irradiation-enhanced yield points have been observed for fission neutron-irradiated niobium by Reed et al.⁽¹⁷⁾ and Loomis and Gerber⁽¹⁸⁾ at fluences of $1-8 \times 10^{17} \text{ n/cm}^2$ and $5 \times 10^{17} \text{ n/cm}^2$, respectively; however, the magnitudes of $\sigma_{uyp} - \sigma_{lyp}$ (difference between upper and lower yield points) were smaller than that observed in the present results. Further experiments are being planned to elucidate the observed enhanced yield point. The light ion/fusion neutron correlation in niobium will be based on the lower yield point because it is a better measure of bulk flow properties. The upper yield point is very sensitive to surface damage or defects which can initiate local flow and hence reduce the applied stress to cause inhomogeneous flow.

The tensile properties of the nickel, 316 SS, and niobium irradiated and tested to date are given in Table 6.

TABLE 6. Tensile Properties (20°C) of Nickel, 316 SS, and Niobium Wire (0.18 mm) Irradiated with (D,T) Neutrons

Material	Fluence, n/cm ²	0.2% YS, MPa	(dσ/dc)1% MPa	UTS, MPa	CT, %	Number Samples Tested	Anneal	
MRC	0	76.2	2122	363	29	8	1 hr at 550°C.	
HARZ	~ 6 x 10 ¹⁶	105.4	2270	336	33	8	1 x 10 ⁻⁶ torr	
Nickel	2.5 x 10 ¹⁷	177.6	2327	375	21	6		
	0	236	2258	506	31	1	20 min at	
316 SS	~ 6 x 10 ¹⁶	232	2556	540	40	3	1050°C, 10 ⁻⁵ torr	
MRC	0	120 ^(a)	395	161	2.7 ^(b)	10.3	5	1 hr at 1050°C.
HARZ	~ 6 x 10 ¹⁶	158 ^(a)	334	160	4.9 ^(b)	13.2	6	2 x 10 ⁻⁸ torr
Niobium								

(a) Lower yield point

(b) Discontinuous strain

FUSION REACTOR SIMULATED CREEP EXPERIMENT

The flux, temperature, and stress of a Tokamak fusion reactor first wall will cycle with a period of a few thousand seconds. The cyclic creep behavior of materials is a problem which the fission reactor designer has not had to concern himself with while the fusion reactor designer will. Penetrating light ions are ideally suited for this type of study because of the ability to chop the ion beam and adjust the temperature and stress of the sample to simulate a fusion reactor cycle. A cyclic irradiation creep experiment is planned to follow the correlation experiment with construction of a creep module which fits the existing ion irradiation chamber. It is expected that the cyclic creep experiment will begin in FY 1978.

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) 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 $1 \times 10^{21} \text{ cm}^{-2} \text{ EFF.}^{(a)}$
- 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

Disassembly of the second and third irradiation capsules, which were discharged from EBR-II in July, was delayed until September. Fluences for these irradiations were 7.3 and $10 \times 10^{21} \text{ cm}^{-2} \text{ EFF}$, respectively; temperatures in both cases were 470°C .

(a) All fluences are quoted in terms of "Equivalent Fission Fluence for Damage in Graphite."⁽¹⁹⁾

Preliminary postirradiation examination shows that the high rate of shrinkage has continued in all four types of cloth and, at the highest fluence, is now two to three times the ~8% reported earlier at $3.5 \times 10^{21} \text{ cm}^{-2} \text{ EFF.}$ (20-22) Nevertheless, three of the four types are still in good shape, typically as shown in Figure 9. Figure 10 shows the fourth type, which has deteriorated significantly.

This task, including publication of results, is expected to be completed next quarter.

FIGURE 9. Carborundum Graphite Cloth, CSGC-2

FIGURE 10. Thornel-50 Graphite Cloth

Fission-Fusion Correlation

Elastic modulus changes of the second set of samples, which were irradiated at the University of California-Davis (D,Be) source with an average neutron energy of about 5 MeV, were about as expected. The third and final set of samples were irradiated in the Hanford N-Reactor in October. Problems with sample temperature control during irradiation, however, will likely mean that this irradiation will have to be redone. This will delay task completion until next quarter.

Sputtering

Glassy carbon samples prepared in the same manner as the one labeled with ^{14}C were found to be contaminated with sodium. Therefore, it has been assumed that the labeled sample is similarly contaminated. Heating unlabeled samples at 2400°C for 2 hr totally removed the sodium as well as oxygen and other minor surface impurities. Following some additional testing of techniques with unlabeled samples, the labeled sample will be heat treated. This will be followed by two or more types of blank runs before neutron sputtering measurements are started.

Degassing

Figure 11 is a schematic diagram of the degassing system that has been assembled. It is based on the pressure-rise method of measuring gas content and degassing rates. After evacuating the sample chamber, the pneumatic valve is closed, and the pressure rise measured by the capacitance manometer is recorded. When the pressure reaches a predetermined value, the valve opens and the chamber pressure is quickly reduced to a lower predetermined value where the valve closes again. By making the pressure-rise time long compared to the pump-down time, total gas content as well as degassing rate can be determined. Degassing rates as low as 1×10^{-11} torr-liter/sec/gram of sample can be determined.

Equipment problems primarily with the valve and the manometer, have been eliminated and degassing measurements are now in progress.

FIGURE 11. Schematic Diagram of Degassing Apparatus

ELECTRICAL INSULATORS FOR FUSION REACTOR APPLICATIONS

J. L. Bates, J. E. Garnier

Current thermonuclear reactor technology calls for the use of electrical insulators in various critical locations in each of the Theta-Pinch, Mirror, and Tokamak designs. The insulator requirements vary within each reactor concept. The Theta-Pinch requires electrical insulators for the first wall, the inner blanket, the implosion coil, and the compression coil. Mirror reactors require inorganic electrical insulators for the baffles of the lithium coolant channels in the blanket, the high energy particle injectors, the particle collectors, and the superconducting magnet. The Tokamak design requires electrical insulators for the lithium coolant baffles of the blanket, neutral particle injectors, superconducting magnets, and radio-frequency heaters, if used. All designs will require adequate insulation for power leads and instrumentation.

These various insulators will be subjected to a variety of cyclic electrical, thermal, and mechanical conditions and must function in the reactor's complex radiation environment. One of the more stringent requirements is the need for a high-temperature (1100 to 1500°K) insulator which is electrically, thermally, and mechanically stable to the 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 (~ 10 kV/mm) under cyclic thermal and irradiation conditions at these high temperatures.

The current objective of this program is the development of a fundamental understanding of the use of acoustic emission (AE) techniques for determining prebreakdown behavior and failure mechanisms in electrical insulators. Work in this quarter has been directed towards the establishment of experimental objectives and the initiation of experiments designed to determine the effect of porosity and helium buildup on the dielectric breakdown strength of insulators using AE techniques.

EXPERIMENTAL OBJECTIVES

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. Studies will be conducted to determine the informational content of the recorded AE data and reproducibility of results. These studies will allow definition of the usefulness of the AE technique in studying effects of helium environment, electrical potential variations, temperature, irradiation, etc. on potential fusion reactor candidate insulator materials.

Since this area of research is new and no studies of this type have been made by other investigators (to the author's knowledge) the above objectives will be carried out on three selected potential fusion reactor electrical insulators. Two have been selected (α - Al_2O_3 and MgAl_2O_4) with a third to be chosen later.

$\alpha\text{-Al}_2\text{O}_3$

Aluminum oxide is a common electrical insulator which can be supplied in high purity forms and can be fabricated by a variety of techniques. Although the radiation stability is not the best, $\alpha\text{-Al}_2\text{O}_3$ will be a reference oxide since other property data relative to irradiation are available and others in the fusion reactor insulator program are studying $\alpha\text{-Al}_2\text{O}_3$.

MgAl_2O_4

This oxide spinel appears to have a high potential for electrical insulator use in fusion reactor applications. The spinel is isotropic and has a relatively open lattice structure made up of low-Z elements. It can be fabricated by a variety of techniques, and the sintering characteristics make it suitable for preparing controlled microstructures with helium-containing pores. The pure spinel (applied as a thick, dense coating to metals by plasma spraying) has potential use as a first wall insulator or protective layer. An evaluation of this material as a coating and bulk insulator is in progress.

Helium Effects on Electrical Properties of Insulators

Helium will be generated in electrical insulators exposed to neutron irradiation in fusion reactors. The presence of helium coupled with the other synergistic effects of particle and neutron radiation, such as lattice damage and impurities from transmutation, may significantly alter the electrical properties of the insulator. At the present time, there is no data on the effects of helium on the electrical resistivity or dielectric strength of insulators and little data relating to helium diffusion and solubility in insulators.

The available data do, however, suggest that helium generated in electric insulators during irradiation, either in the lattice or in pores, will potentially alter the electrical properties, in particular, the dielectric strength. Related studies with actinide oxides indicate that substantial helium still remains at $\sim 1600^\circ\text{C}$, and below 1000°C helium diffusion rates are low. These studies suggest that the solubility in the electrical insulators may be low, resulting in the formation of helium-containing pores or defects. The formation of pores in itself is known to reduce the insulation capacities of ceramics. The effects of gas in the lattice of oxides on the electrical properties has also received little attention. Alpha decay in PuO_2 cause a saturation of point defects, and high helium inventories result in a change in electrical properties.

One of the major problems in evaluating the helium effects on the electrical properties is the introduction of helium into the insulators. Irradiation in an environment similar to that expected in a fusion reactor is the ideal method since it combines the synergistic lattice effects resulting from irradiation. However, because it does introduce effects of lattice damage and transmutation, it would be difficult to isolate initially the potential effects resulting only from helium buildup. Cost and time suggests that other means be considered for a more rapid and controlled helium introduction.

A study is to be made on the feasibility of introducing helium into the lattice of an insulator material. A number of potential methods being considered are:

- tritium trick
- irradiation with an alpha emitter
- doping with an alpha emitter
- ion or particle irradiation
- dissolution at high pressures and temperatures, and
- entrapment during fabrication.

Introduction of helium during fabrication by sintering will be investigated initially. Sintering at different helium pressures to high density with closed porosity will provide variable pressures of helium in the pores. Variations in the number, size, and distribution of pores can be obtained by controlling the composition and sintering conditions and by the use of pore-formers. Helium will reside predominantly in the pores.

The second method to be investigated will involve the dissolution of helium by heating the insulators in a high pressure helium environment at high temperatures. Use of single crystals and polycrystalline insulators will permit an evaluation of the effects of lattice-dissolved helium in contrast to that found in pores.

INITIATION OF EXPERIMENTS

To achieve the experimental objectives discussed, effort this quarter was directed towards the following:

- Experimental system setup. An experimental system is being assembled as shown in Figure 12. In a given experimental run, the following information can be obtained:
a) total AE counts and energy; b) AE event energy; c) one or more individual AE event signals, which can be analyzed for frequency content; and d) the number and magnitude of discharge pulses.
- Specimen fabrication. Preparation of samples was initiated for polycrystalline Al_2O_3 and spinel with varying percentages (0 to 15%) of porosity, sintered in vacuum and helium atmosphere. These samples are to be used in a detailed study of porosity and helium buildup effects on the dielectric strength of insulators.

FIGURE 12. Circuit Schematic for Detection and Measurements of Acoustic Emissions and Discharge Pulses in Insulator Materials

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.

The available fusion reactor design concepts have been analyzed to determine the probable interactions with the environment and the resultant environmental effects. Although results based on current technology indicate that the fusion reactor should have a lower total environmental impact than fission power plants, adverse impacts may result from an increase in construction material usage and an increase in the radiation dose to employees. Since commercial power plants are not expected to operate for 40 to 50 years, future technical advances should lower these impacts. A document describing the analysis and eighteen supporting reference documents have been published.

PNL has begun an effort this fiscal year to define the scope of a fusion reactor safety program plan and to identify the basic technology necessary to prepare a comprehensive safety analysis report on a fusion power plant. Formulation of the plan emphasizes planning and implementation activities and the identification of safety criteria applicable to fusion reactors. This quarter, contacts were initiated with other laboratories engaged in fission safety research and an investigation was started to examine the relevance of existing fission reactor safety criteria to the fusion program.

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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.

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.

FUSION SYSTEM SAFETY ANALYSIS

The development program leading toward practical fusion power will involve the construction and operation of a number of experimental reactors. This will require an understanding of the normal and transient 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.

FUSION SAFETY PROGRAM PLANNING

H. J. Willenberg, S. H. Bush

PNL's fusion system safety analysis task is identifying the key needs and providing a plan for obtaining the above information. This task will also review existing and evolving safety criteria which may be applicable to fusion reactors.

PLANNING AND IMPLEMENTATION

A program plan is being developed to guide DMFE, 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 a full-time task leader was designated to conduct and coordinate the two safety-related activities which were initiated this quarter. A guide to the planning activity was prepared in preparation for discussions with DMFE. In addition, contacts were initiated with all laboratories engaged in fusion safety research.

SAFETY CRITERIA

The technical bases for the establishment of design criteria directed to the safety performance of fusion reactors are being identified. Available information is being reviewed for adequacy, and recommendations will be made for the provision of needed additional information. Philosophical considerations supporting specification of design criteria will be developed and documented. An examination of existing fission reactor criteria for relevance to the fusion power program was begun this quarter to initiate this component of the safety analysis task.

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 present quarter includes an introduction to plasma engineering and its application to controlled thermonuclear fusion. The second and third quarters will cover blanket technology and materials problems in fusion reactor conceptual designs.

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

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