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UTILIZATION OF FUSION NEUTRONS IN
THE TOKAMAK FUSION TEST REACTOR
FOR PLASMA PERFORMANCE TESTING
AND OTHER RESEARCH ENGINEERING
EXPERIMENTS

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UTILIZATION OF FUSION NEUTRONS IN THE TOKAMAK
FUSION TEST REACTOR FOR BLANKET PERFORMANCE TESTING
AND OTHER NUCLEAR ENGINEERING EXPERIMENTS

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UTILIZATION OF FUSION NEUTRONS IN THE TOKAMAK FUSION TEST REACTOR FOR BLANKET PERFORMANCE TESTING AND OTHER NUCLEAR ENGINEERING EXPERIMENTS

ABSTRACT

Neutron utilization experiments in the Tokamak Fusion Test Reactor (TFTR) can be designed to obtain engineering data on tritium breeding and fissile breeding in blanket modules that are representative of practical reactor blanket modules, on neutron streaming in special duct configurations, and on radiation damage testing of arrays of insulators. This experimental program follows naturally from accelerator-based physics experiments that have provided important integral cross section data, but which have limited benefit to fusion reactor development because of their very small sample volumes, point neutron sources, and (in most cases) a neutron spectrum dissimilar to that incident on a toroidal fusion reactor blanket.

Without experiments on an actual fusion machine, it is unlikely that the breeding performance of a finite heterogeneous blanket in a given region of a toroidal reactor can be predicted at low risk to better than a factor of 1.35 to 1.5. The TFTR neutronics experiments can provide quantitative information concerning tritium and fissile breeding ratios in a realistic geometry with an extended neutron source. By benchmarking blanket design codes with the TFTR engineering data, the uncertainty in predicting breeding performance in practical fusion and hybrid reactors with restricted breeding regions would be greatly reduced, so that the minimum blanket coverage factors to be achieved could be specified. By combining neutron and gamma-ray dosimetry data, power distribution benchmarks would also be obtained.

In addition to developing a set of reacting-plasma/blanket-neutronics benchmark data, the TFTR fusion application experiments would provide operational experience with fast-neutron dosimetry and the remote handling of blanket modules in a tokamak reactor environment; neutron streaming and hot-spot information invaluable for the optimal design of penetrations in future fusion reactors; and the identification of the most damage-resistant insulators for a variety of fusion-reactor components.

The total cost of this fusion applications program is estimated to be about 1% of the total cost of the TFTR facility and operations. Thus at modest incremental cost, the TFTR can be exploited as a testbed for certain important aspects of fusion reactor blanket engineering and materials testing. This report begins to address the feasibility of implementing neutron utilization experiments.

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I. INTRODUCTION AND MOTIVATION

I.1 EXPLOITATION OF THE TFTR FACILITY

The Tokamak Fusion Test Reactor (TFTR) [1] will offer an opportunity for practical experiments using 14-MeV neutrons that will be unique in the near future. For the first time, important neutronics experiments, including the breeding of tritium and uranium-233, can be carried out in a radiation environment and blanket module configuration that are nearly prototypical of those of a practical tokamak fusion reactor. In order that the TFTR facility can serve this vital nuclear engineering role, as well as its well-known plasma physics and plasma engineering roles, it is necessary that planning and implementation of neutron utilization experiments be initiated now. Such experiments would be consistent with the stated policy of the U.S. Dept. of Energy to make maximum use of existing and planned tokamak facilities — in this case the TFTR. This report discusses those experiments with TFTR fusion neutrons that we feel will serve as an essential component of the U.S. toroidal reactor development program, and will insure fuller exploitation of the TFTR's unique DT-burning capability.

At the present time it seems likely that both the injected beam power and the heating pulse length in the TFTR will eventually be increased substantially above the initially prescribed values, in order to reach conditions of higher fusion energy gain in D-T, and to study those reacting-plasma conditions for significant periods. In that case, the neutron fluence per pulse would be many times larger than assumed in this study, so that there would be even greater incentive to utilize the fusion-neutron production. Thus the TFTR can be much more than an instrument for studying hot plasmas. The TFTR blanket experiments proposed herein would be the first on a practical fusion machine, and would provide a large step forward from the many conceptual design studies of blankets that are currently underway and planned for the next several years.

The TFTR neutron utilization experiments are referred to collectively as FAX, for Fusion Application Experiments. We emphasize that the tritium breeding experiments proposed for the TFTR-FAX program, as well as materials irradiation and neutron streaming experiments, are intended to support directly the U.S. development program for "pure fusion" reactors. Only the U-233 breeding experiments are concerned specifically with hybrid reactor development.

1.2 NEED FOR TFTR INTEGRAL NEUTRONICS EXPERIMENTS

Performance of the integral blanket experiments during TFTR D-T operation, which is scheduled to start in 1983, will exploit a unique opportunity in the U.S. magnetic fusion energy development program. This experimental window is expected to close after the TFTR D-T operating period, not to reappear until about 1990 when a more powerful fusion device becomes operational. Careful planning, adequate conceptual design, and reservation of space now on the TFTR will enable important breeding demonstration measurements to be executed during the only such experimental window within reach in the next decade. Industrial, utility, and ultimately government support and involvement in extending the domestic resource base for electric power production via the pure fusion and fusion-hybrid routes will depend upon success in reaching tangible milestones at frequent intervals. The integral experiments of the proposed FAX program, described in this report, will provide a direct measurement of two important fuel production parameters which are vitally relevant to the economic feasibility of fusion-based energy systems. The production of tritium and U-233, as measured by sensitive radiochemical techniques, will be compared directly to the incident 14 MeV neutron fluence (i.e., the fusion neutron wall loading). This experimentally measured parameter is already the focal point of an extensive diagnostics program in the TFTR project [2].

D-T operation in the next large U.S. tokamak facility, such as the ETF (Engineering Test Facility), is expected to begin not before 1988-90, and blanket experiments might not commence until one year or so later. This timescale is sufficiently long so that the neutronics performance of prototypical ETF modules for tritium and fissile breeding can be tested on the TFTR. Early confirmation of adequate breeding will enable the ETF blankets to be designed at a higher confidence level.

Thus at an estimated total cost for the entire FAX program of the order of 1% of the cost of the TFTR facility and operations, the usefulness of the TFTR to the development of practical toroidal fusion reactors can be augmented greatly.

(The experimental program discussed in this report was first outlined in a PPPL memorandum entitled, "On Tritium and Fissile Breeding Experiments with TFTR Fusion Neutrons," by D. L. Jassby, issued on 22 May 1978, and revised version issued on 16 June 1978 [3].

A feasibility study of the experimental program discussed herein has been proposed in an Internal Report [4] entitled, "Feasibility Assessment of the Use of TFTR Fusion Neutrons," issued on 14 August 1978 by the Princeton Plasma Physics Laboratory and The Babcock & Wilcox Company, Lynchburg Research Center. That report formed part of a proposal submitted to agencies that fund energy research and development.)

2. EXPERIMENTAL OBJECTIVES

The proposed Fusion Application Experiments (FAX) are listed here in order of priority:

- (1) Demonstration of a tritium breeding ratio greater than unity, in a prototypical blanket module.
- (2) Determination of U-233 breeding ratios in prototypical blanket modules.
- (3) Development of dosimetry techniques for fusion reactor blankets.
- (4) Accumulation of bench-mark data to calibrate 2D and 3D neutronics codes, in the areas of transport, thermalization, streaming and leakage of neutrons.
- (5) Radiation damage testing of insulators of fusion reactor relevance.

Here the tritium breeding ratio, TBR, is defined as the number of tritium atoms bred in the reference sector of a blanket module, divided by the number of 14-MeV neutrons incident on that sector. The uranium-233 breeding ratio, UBR, is defined similarly.

(Although not a part of current planning because of nonproliferation concerns, plutonium breeding in a uranium blanket could be added at a later date as a hybrid benchmark experiment.)

In addition to providing invaluable neutronics information, the TFTR blanket experiments would provide unique experience in important fusion reactor systems areas: (1) operational experience with blanket dosimetry in a tokamak reactor environment, (2) remote handling of blanket modules in the tokamak environment, and (3) verification that the electromagnetic interaction of the blankets with the tokamak plasma can be made sufficiently small. This systems experience, which is vital to the confident design of blanket modules for an engineering test reactor, can be obtained in no other way.

Table 1 indicates a possible program for developing fusion reactor blanket modules. The first two items are discussed in Section 5 of this report. The TFTR blanket experiments will play a key role in linking present and planned accelerated-based neutronics experiments to the demands of practical blankets on engineering test reactors, fusion breeders, and electrical power reactors.

Table 1.

PROGRAM FOR DEVELOPING FUSION REACTOR BLANKET MODULES

● PRESENT ESTIMATES OF BLANKET PERFORMANCE (Cross sections and conceptual design studies)	1978
● ACCELERATOR-BASED BLANKET SIMULATION EXPERIMENTS	1975-81
● TFTR BLANKET MODULE EXPERIMENTS	1982-87
● ETF BLANKET MODULE DESIGN	1983-88
● ETF PROTOTYPE BLANKET MODULE TESTING (Thermal hydraulics, neutronics, fuel extraction)	1985-90
● ETF BLANKET MODULE OPERATION	1989 +

[ETF $\equiv$ Engineering Test Facility]

3. SUMMARY OF PROPOSED EXPERIMENTS

This section provides an overview of the neutron utilization experiments that we have considered. The principal fusion and fission reactions of interest are listed in Appendix A.

In the base-line D-T operation of the TFTR [1], pulse lengths will range up to 1 s, with 14-MeV neutron intensities as large as 10^{19} n/s. In the following it is assumed that the average 14-MeV neutron yield is 3×10^{18} n/pulse. With the prescribed total of 4000 full-power D-T pulses, the total 14-MeV neutron production would be 1×10^{22} neutrons, or a total uncollided fluence at the first wall of approximately 1×10^{20} n/m².

(In 100,000 beam-injected deuterium pulses, the total 2.45-MeV neutron fluence from $D(D,n)^3\text{He}$ would be approximately 3×10^{19} n/m², with an accompanying 14-MeV neutron fluence of the order of 10^{18} n/m² resulting from partial burn-up of the triton reaction product. (The 14-MeV flux may be enhanced by triton recycling.)

3.1 BLANKET MODULE LOCATIONS

It is proposed that one of the ten TFTR segments be devoted to fusion application experiments. Relatively thin blankets, perhaps 20 to 30 cm in thickness, could be located between the vacuum vessel and the TF (toroidal-field) coils. However, a somewhat thicker blanket is needed to absorb essentially all the 14-MeV neutrons, and to serve as a prototype of a practical fusion reactor module. At least one blanket of arbitrary thickness and with a surface area of at least 0.5 m² (e.g., 0.6-m wide by 1.0-m tall) could be located between adjacent TF coils. Figure 1 depicts the placement of a module of this latter type. Nevertheless, thin test volumes located between the vacuum vessel and TF coils, as depicted in Fig. 2, can still be invaluable for such purposes as radiation damage experiments.

3.2 TRITIUM BREEDING

Any system of DT-based fusion reactors must have an overall tritium breeding ratio (TBR) of at least unity. Therefore, an important feasibility experiment for the TFTR is to demonstrate $\text{TBR} > 1$ in a quasi-realistic blanket geometry. (Here TBR is defined as the number of tritium atoms bred in the reference sector of a blanket module, divided by the number of 14-MeV neutrons incident on that sector.) In a practical D-T reactor, the minimum TBR (as defined here) will

786472
Fig. 1. Elevation view of the TFTR, showing the blanket module location at a port section between two adjacent toroidal-field (TF) coils. (Adopted from drawing of TFTR Diagnostics Group [2].)

786473
Fig. 2. A thin blanket module under a toroidal-field coil.

have to be about 1.2, as the blanket coverage factor is unlikely to exceed 0.85 to 0.90, and the tritium useful recovery factor may be only 0.95. Thus an important objective of the TFTR blanket experiments is to determine whether a TBR of 1.2 can be attained in blankets of realistic composition.

In 10 full-power D-T pulses in the TFTR, the atom concentration of tritium in a lithium-oxide lattice would build up to the range 2 to 6×10^{-11} . The corresponding beta activity of the order of $0.2 \mu\text{Ci/g}$ is easily measured. Attaining a high-performance breeding system will probably require significant neutron multiplication in materials such as lead or beryllium. The effect of the vacuum vessel wall in fusion-neutron attenuation and energy degradation will be assessed in these experiments. A preliminary blanket module concept, which could be used for measuring the TBR, is discussed in Section 4.

Tritium production will be evaluated both as a function of location and as an integrated value over the experimental module. The accuracy with which tritium and fissile breeding ratios can be measured is discussed in Sections 6 and 7.

3.3 URANIUM-233 BREEDING

For a number of reasons, the fissile nuclide U-233 is perhaps the most attractive fissile fuel for thermal converter reactors. U-233 can be bred from thorium, whose supply in the U.S. is comparable with that of uranium, and is probably several times greater on a world-wide basis [5]. In a once-through fuel cycle employing 3%-enriched fissile fuel, only 1/5 as much thorium ore need be mined, as compared with the uranium ore required to produce 3% U-235 by concentration in an isotope separation plant. U-233 fuel offers a much higher conversion ratio than either U-235 or Pu-239 fuel, which is advantageous if reprocessing is carried out. Denatured U-233 (i.e., diluted with U-238) is claimed to be much more "proliferation resistant" than Pu-239 [6].

One method of breeding U-233 is by the bombardment of thorium with fast neutrons. With a total fusion neutron fluence of $2 \times 10^{13} \text{ n/cm}^2$ obtained in 10 high-power shots in the TFTR, about 0.05 milligram of U-233 could be bred in a 25-cm thick thorium-bearing assembly of 0.5-m^2 area, resulting in a fissile atom concentration of the order of 1×10^{-9} . However, concentrations orders of magnitude smaller can be measured from the characteristic 310-keV gamma radiation

of the intermediate nuclide, ^{233}Pa , whose half-life is 27 days [7] (see Fig. 3). *A single full-power TFTR pulse could produce an activity of 3000 disintegrations per second per gram of thorium.*

In later experiments, both U-233 and tritium breeding could take place in the same assembly. A wide variety of blanket compositions can be considered, such as one with a depleted-uranium neutron-multiplying layer preceding the thorium and lithium regions. The U-233 breeding experiments could also make use of thorium-bearing pellets or particles suitable for refabrication and direct use in fission reactors, such as the LWR, LWBR, or HTGR, without chemical re-processing.

Blanket heating by neutron and gamma deposition will be negligible for standard TFTR operation with 1.0-s pulses, occurring once every 300 s. Nevertheless, a realistic coolant configuration would be used in the breeding modules, so that the neutronics would resemble that of reactor blankets. In the event of a "super-upgrade" of TFTR with high-power 10-s pulses, then blanket heating could be significant, especially if Q-values much exceeding unity were attained. In that case, the production per pulse of tritium or U-233 atoms would also be much higher than indicated above (although higher production rates are unnecessary for the neutronics experiments of interest).

3.4 BLANKET DOSIMETRY

Integral neutronics measurements would be performed in conjunction with all the blanket experiments. That is, neutron energy spectra would be measured as a function of position in the lattice by energy-sensitive neutron activation detectors, for example. The intensity and spectrum of the neutron leakage out of the blanket assemblies would also be determined. Thus integral neutronics experiments that are difficult to calculate accurately can be performed in relatively large blankets that are crudely prototypical of reactor blankets. These measurements will provide definitive data for comparison with the results of neutronics codes that are based on cross-section data derived from "clean" experiments with accelerator/target neutron sources (see Section 5).

A development and calibration program may be required for much of the dosimetry needed for the energy range 3 to 15 MeV, which is relatively uncharted. These dosimetric techniques will be essential for use in future fusion reactor blankets.

Figure 3. Protactinium-233 gamma-ray spectrum, from Ref. [7].
 The principal line of interest is at Channel No. 311.8.

3.5 NEUTRON STREAMING

Neutron streaming in complicated geometries is difficult or impossible to calculate accurately. Duct configurations that are of interest for future reactors can be placed in blanket modules and in the igloo shield of the TFTR. Real-time detectors and neutron cameras will measure neutron and gamma-ray leakage and locate neutron "hot spots". Integral neutronics and gamma-ray experiments will be performed at positions behind and within various shielding compositions.

These experiments, the first with an extended fusion-neutron source, will provide bench-mark data for comparison with neutron transport codes.

(It is expected that neutron streaming through the major vacuum penetrations will be monitored in any event for safety reasons.)

3.6 RADIATION DAMAGE TO INSULATORS

The radiation dose absorbed by insulators located near the first wall will be about 10^8 rads/yr. A very large number of specimens can be irradiated simultaneously in various test modules, each of which might have a volume of the order of 0.1 m^3 (see Fig. 2). The types of insulators that would be irradiated include organic insulators (dose limits = 10^8 to 10^{10} rads), inorganic insulators, "superinsulation," seals, and lubricants [8]. In situ non-destructive measurements of properties such as resistivity, dielectric strength, and dielectric constant would be made periodically. Post-irradiation testing would measure degradation of mechanical properties, such as tensile strength, flexural strength, shear strength, and modulus of elasticity, as well as chemical changes and the build-up of helium and hydrogen.

The annual total neutron fluence near the first wall of the TFTR ($\sim 1 \times 10^{16} \text{ n/cm}^2$ including backscattered neutrons) is comparable with the annual fluence expected at the superconducting coils in toroidal reactor designs [8], although the spectrum at the coils would be much softer. An additional important consideration for organic insulators used in superconducting coils is the effect of a cryogenic environment. The feasibility of long-term cryogenic testing in the TFTR needs to be determined.

(We note that radiation damage to structural materials in the TFTR is expected to be exceedingly small.)

At the Idaho National Engineering Laboratory, samples of an insulator material that has been proposed for use in the toroidal-field coils of the TFTR have been irradiated in a fission reactor with a specially shaped spectrum.

At doses of the order of 10^8 rad, small damage effects were observed [10]. If these experiments are confirmed, they would indicate that useful radiation damage experiments can be performed in the TFTR neutron field, and with at least hundreds of specimens simultaneously.

3.7 NUMBER OF PULSES REQUIRED

As shown in Table 2, each neutronics and breeding experiment with a given blanket module can be completed in a single day's run of full-power D-T shots (10 to 96 shots). While even a single pulse would be adequate for some neutronics experiments, several pulses may be desirable for improved statistics.

Table 2.

MINIMUM EXPOSURE TIMES IN TFTR

<u>Experiment</u>	<u>Number of Full-Power D-T Pulses per Assembly</u>
1. Tritium breeding.	1 to 10
2. U-233 breeding.	1 to 10
3. Miscellaneous blanket neutronics.	1 to 10
4. Neutron streaming.	3 to 10
5. Radiation damage to insulators.	≥ 1000

4. ILLUSTRATIVE BLANKET MODULE CONCEPT

The breeding test module concept must incorporate a design permitting appropriate experimental data collection in a configuration representative of a practical blanket design. The lithium or thorium fertile material can be arranged in a geometry that promises minimal effort in removal of selected portions for analysis. An illustrative fuel assembly, shown in Figs. 4 and 5, can be designed to provide (1) semi-permanent fuel rods whose handling, removal, and analysis will take place only during TFTR outages or maintenance shutdowns, and (2) rapid-access "rabbits," pneumatically or mechanically transferred, whose removal/analysis may be accomplished at any time. The rapid-access samples will not remain in the module sufficiently long to require remote handling.

A preliminary concept illustrating a possible module arrangement applicable to both tritium breeding and uranium-233 breeding is shown in Fig. 6. Descriptive information concerning the module is given in Table 3. This module concept consists of a support frame into which short clad fuel rods are stacked in a hexagonal array. Each tube will contain powder, microspheres, or pellets of Li_2O , Li_7Pb_2 or ThO_2 . The fuel array is sized to provide acceptable nuclear isolation around a central zone in which several rapid access "rabbit" sample tubes are located. Locations between fuel rods as well as locations adjacent to the fuel array are available for housing activation foils, pull wires, and other experimental materials as needed (see Fig. 5). The inherent flexibility and relative ease of access to components makes this approach highly desirable, since sample access requirements will be dependent on the operating schedule of the TFTR. This approach also provides the best approach to efficient utilization of TFTR operating schedules for exposure of experimental materials and of access to the module components during downtime periods. Provisions can be made for incorporating neutron multiplying components in or near the module if this is desirable. The ability to remove individual fuel rods from the array allows for interim examination of rods and substitution of alternate materials or compositions. In the case of U-233 breeding, it represents the type of system most likely to be commercially feasible for extraction of the fuel rods for use in fission reactors.

Illustrative calculations of tritium and U-233 breeding with oxide fuels have been performed with the NITAWL and XSDRN codes, and the results are given in Appendix B for water-moderated blankets. Experimental measurements would be taken under both dry and water-moderated conditions.

Fig. 4.

Scale: Approx. 2x

786471
Fig. 5. Axial view of fuel matrix.

786470
Fig. 6. Illustrative blanket module concept for TFTR.
The fuel rods fill the hexagonal shell.

Table 3. Parameters of Illustrative Small Blanket Modules

Basic Structural Material: Austenitic Stainless Steel

Fuel Clad Material: Zircaloy-4

Fuel Array: Hexagonal array of stacked fuel rods (see Fig. 6). Spacing of 0.5 mm achieved by oversized end caps or wire wrap. See Figs. 4 and 5.

Experimental Space Available:

* Thirteen 1.25-cm diameter rabbit tubes, seven in center zone at least 30 cm from nearest escape surface.

* Approximately 2,000 (~ 0.25-cm diameter) flux wire positions.

2,180 remotely removable rods (accessible only during maintenance periods) containing lithium compounds or ThO_2 .

Space available outside array.

Size of Structure: 90 cm tall $\times$ 80 cm wide $\times$ 40 cm deep **

Fuel Matrix: Lithium compound or ThO_2 materials either as pressed pellets or as packed powders and inserted in 1.2 cm O.D. Zircaloy-4 tubes to 30 cm length. Pellet size $\approx$ 1.0 cm diam. $\times$ 2.5 cm length.

	<u>Tritium Breeding Module</u>	<u>U-233 Breeding Module</u>
Fuel Material	Li_2O	ThO_2
Fuel Weight	130 kg	790 kg
Total Module Weight	330 kg	990 kg

Note: Other lithium compounds, such as Li_7Pb_2 or LiAlO_2 , may be preferred.

* Rapid-access systems, via pneumatic shuttle or pull-wire tubes.

**A deeper module (e.g., 60 cm) can easily be accommodated.

A detailed engineering design would be established based on the physical constraints of the TFTR layout, the requirements for nuclear isolation of the test region within the module, and the need to accommodate the supporting experimental equipment, such as rabbit tubes, activation foils, and other neutron dosimetry material samples. Module installation and residence time would be estimated based on neutronics calculations, and on what is currently known of TFTR scheduling.

A judgment would have to be made concerning the feasibility of using geometrically similar test modules, or the same module at different times for the tritium and U-233 breeding experiments. The possibility of using a dual-purpose tritium and U-233 breeding module design should also be addressed. Also to be evaluated are the vital additional elements involved in mating the blanket module with the TFTR installation, such as the need to displace shielding blocks, provision for structural support, compatibility with TFTR remote handling equipment, available space for rabbit connections and receivers and electrical and instrumentation requirements. (No modifications would be required to the TFTR vacuum vessel.)

Appendix C summarizes what we consider are the ground rules for the design and operation of TFTR blanket modules.

5. RELATION TO OTHER EXPERIMENTAL PROGRAMS

Experiments with high energy neutrons ($E_n \gtrsim 10$ MeV) can be performed with accelerator/target facilities, whose parameters are compared in Table 4. In fact, low-intensity D-T neutron sources have been used to provide energy-dependent cross-section data in "clean" experiments (i.e., with a simple lattice and geometry), while the high-intensity RTNS [11] and LAMPF facilities [12] have been used for radiation damage studies with small numbers of small specimens. The FMITF [13], a D-Li neutron source which is to become operational at about the same time as D-T operation in the TFTR, can subject small specimens to extremely large radiation intensities.

However, as explained in the following, the TFTR facility will be by far the most suitable test-bed available anywhere in the world in the near future for the proposed experiments, and for many purposes it will be unique. The small incremental cost that is expected to be required for those experiments deemed most feasible (see Section 10) should result in a substantial increase in the benefits to be gained from the TFTR project.

5.1 NEUTRON ENERGY SPECTRA

Figure 7 compares the neutron spectrum of a 10-keV thermonuclear (TN) plasma with that of a 10 keV beam-target (TCT) plasma driven by 120-keV beams [14]. It is expected that the neutron source of most future tokamak reactors, such as an ETF (Engineering Test Facility) will be predominately of the TN type, while the TFTR neutron source will be primarily of the TCT type. At 14 MeV, the FWHM of the TCT spectrum is 1.15 MeV, compared with 0.62 MeV for the TN spectrum. The difference in FWHM values is only 4% of the nominal neutron energy (14.1 MeV), and will be masked by neutron attenuation ($\gtrsim 5\%$) in the reactor liner and "first wall" and also by backscattered neutrons. Figure 3 shows the calculated TFTR spectrum just outside the first wall (1.25 cm of SS-304) in a region between TF coils. While the dominant features below 10 MeV are obviously determined by backscattering, the 14-MeV "spike" is also broadened by the first-wall attenuation and by backscattering. These features are expected to be similar in the TFTR and in the ETF, so that the TFTR blanket neutronics results should be representative of those to be expected in the ETF.

The energy spectra of neutrons in the FMITF [13] and LAMPF [12] facilities are shown in Fig. 9. These spectra are evidently far removed from a fusion spectrum, and therefore can give little useful information concerning the neutronics performance of fusion reactor blankets. Note in particular that the TFTR spectrum has a "spike" at 14 MeV [see Fig. 8], whereas the FMITF spectrum has an extremely broad peak in this region. The D-T neutron sources

Table 4
PARAMETERS OF FUSION-NEUTRON TEST FACILITIES

Experimental Tokamak Reactors	Operating Date for D-T	14-MeV Intensity n/s	14-MeV Flux n/cm ² /s	14-MeV Fluence n/cm ² /yr	Total n Fluence n/cm ² /yr	Comments
TFTR	1983	5×10^{18}	4×10^{12}	2×10^{15}	1×10^{16}	1000 pulses/yr
TFTR Upgraded	1985	1×10^{19}	8×10^{12}	1×10^{16}	5×10^{16}	1000 pulses/yr
JET	1986	1×10^{19}	5×10^{12}	1×10^{16}	5×10^{16}	500 pulses/yr
T-20	1988(?)	6×10^{19}	2×10^{13}	2×10^{18}	1×10^{19}	Machine not authorized
<u>Materials Test Devices</u>						
RTNS-I (D+T)	1974	6×10^{12}	$\leq 2 \times 10^{12}$	$\leq 4 \times 10^{19}$		
RTNS-II (D+T)	1978	4×10^{13}	$\leq 1 \times 10^{13}$	$\leq 2 \times 10^{20}$		
FMITF (D+Li)	1983	3×10^{14}	$10^{14} - 10^{15}$		$10^{21} - 10^{22}$	Non-fusion Spectrum
LAMPF (Spallation)	1977	$\sim 1 \times 10^{14} *$	$\sim 2 \times 10^{11} *$	$\sim 2 \times 10^{18} *$	$\sim 2 \times 10^{20}$	Non-fusion Spectrum
<u>Neutronics Experiments</u>						
JAERI	1974	3×10^{10}	3×10^8	6×10^{15}		Sample at r = 3 cm
FIEF	1978	1×10^{10}	3×10^7	6×10^{14}		r = 5 cm
Tokamak Engineer. Test Facility	1990	1×10^{20}	4×10^{13}	2×10^{20}	1×10^{21}	

*E_n ≥ 10 MeV

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793001
Fig. 7. Comparison of neutron energy spectra, for thermonuclear and TCT ($D \rightarrow T$) fusion neutron sources. (See Ref. [14])

793009
Fig. 8. Calculated TFTR neutron spectrum just outside the stainless steel vacuum vessel, between two toroidal-field coils. (Figure supplied by L. P. Ku)

Fig. 9a. Neutron yield at 0 deg from thick lithium target bombarded with 40-MeV deuterons. From Ref. [13], p. 390.

Fig. 9b. Neutron flux spectrum for LAMPF spallation target, at 135 deg. Solid line is calculated spectrum and dashed line is unfolded flux by foil activation dosimetry. From Ref. [12], p. 343.

(RTNS, JAERI, FIEF) provide the same *primary* spectrum as that from a reacting fusion plasma, but the complete neutron and gamma-ray spectra present in a tokamak reactor blanket due to multiple scattering (see Fig. 5) will not be reproduced in these accelerator-target systems. Furthermore, energy attenuation in the first wall of a magnetic fusion reactor is enhanced by the extended nature of the neutron source, because of oblique impingement on the wall by a substantial fraction of the fusion neutrons. This effect is difficult to reproduce with a point neutron source. Much of the lower energy flux in Fig. 8 is due to back-scattering of neutrons around the torus, another effect that is difficult to simulate with an accelerator/target system.

5.2 RELATION TO EXPERIMENTS USING ACCELERATOR-BASED NEUTRON SOURCES

Basic physics experiments using accelerator/target neutron sources and simple blanket configurations have served as logical precursors to the proposed TFTR-FAX experiments. These basic experiments include transport and spectral measurements, cross-section and code verification, and integral reaction measurements. The first experiment of this type was performed by Weale and his collaborators in Britain in 1960 [15]. During the last several years, a large number of other basic physics experiments have been performed in the U.S., Japan, and Western Europe, and many of these are listed in Appendix D. For example, experiments using large lithium spheres have been carried out and analyzed by Maekawa and Seki [D.2, D.3, & D.4], Bachmann, et al. [D.5] and Hansen [D.6]. Lee [D.7] has analyzed reaction rate measurements in a uranium cylinder and has reported agreement with predicted values to within 10%. Hansen [D.8] at Livermore has recently completed additional pulsed sphere measurements and analysis for ^{232}Th , ^{235}U , ^{238}U , and Pu in bare metal configurations. Improved agreement in calculated versus experimental values has been obtained using updated ENDF/B IV cross-section libraries. However, significant discrepancies are still reported to exist for lithium metal blankets [D.2 to D.5] and for bare thorium metal [D.8].

Further basic experiments and analytical matching will continue to be performed, and this activity will contribute to the network of knowledge needed for blanket design and analysis. The TFTR-FAX integral experiments can be designed and performed in parallel with the ongoing basic work. The order in which the basic and integral experiments are performed is not of primary importance. The key issue is to take advantage of the unique one-time opportunity provided by TFTR as a test bed for doing several

integral blanket tests as realistic benchmarks. The accelerator-based experiments have all made use of a point neutron source, a simple neutron spectrum, and a homogeneous nearly spherical blanket— conditions that will not hold in a toroidal fusion reactor. The proposed TFTR experiments will provide a *direct* measurement of tritium and U-233 breeding in an analyzable, yet prototypical reactor blanket module, using an extended fusion-neutron source which can be described by measured plasma indices.

At the Magnetic Fusion Energy Blanket and Shield Workshop [16] sponsored by ERDA in 1976, three categories of integral experiments were identified:

- Type I - To Evaluate and Improve the Understanding of Basic Cross-Sections and, to a Lesser Extent, Computational Methods.
(Clean, homogeneous, simple sphere or cylinder)
- Type II- To Test the Feasibility of Generic Design Concepts
(Geometrical or compositional complexities to test generic engineering design approaches.)
- Type III - To verify the Performance of Detailed Designs
(Includes all important features of proposed reactor.
Used as verification just prior to final design freeze.)

The Type I(Fundamental) experiments are presently being performed with the accelerator-based neutron sources. The proposed TFTR-FAX program satisfies most of the requirements of the Type II (Feasibility) experiments, and may also be able to address some of the requirements of the Type III (Mockup) experiments.

5.3 NEUTRON STREAMING AND SHIELDING EXPERIMENTS

An experimental program on neutron shielding is currently underway at ORNL in the Fusion Integral Experimental Facility (FIEF), which provides a low-intensity 14-MeV neutron source (see Table 4) [17]. The success of the basic shielding experiments now being performed on slab configurations depends heavily on accurate spectral and intensity measurements for various slab thicknesses, and especially on the neutron and gamma flux emitted from the back surface of the shielding array. Here the objective is to attenuate neutrons by factors of 10^5 to 10^{10} and also to attenuate the secondary radiation to appropriately low levels. The steep gradients, scattered radiation, reflected neutrons, and the basic design requirements are severe challenges for the experimentalist.

For many types of active detectors, and also for activation foils or pellets with short half-lives, one needs to have a minimum flux, and this flux cannot always be obtained behind significant shielding using a relatively weak neutron source. At a distance of 5 cm from the tritiated target in FIEF, the fusion-neutron fluence in 1 day of continuous operation is comparable with that obtained in a single full-power pulse at the TFTR vacuum vessel.

The classical shielding experiments, like the accelerator-based physics experiments, will provide important data for comparison with sophisticated computer codes. But the TFTR will provide uniquely practical duct configurations with an extended neutron source and realistic toroidal geometry — effects that cannot be simulated in a slab experiment. In the TFTR, a variety of ducts and slots can be configured out of the igloo shielding in order to mock up special penetrations that will be found in future fusion reactors. Furthermore, the thickness and configuration of the local shielding around the TFTR beam injectors and other vacuum penetrations can be arranged at will in a controlled program. It is expected that some of these arrangements will result in beneficial variations in neutron and gamma-ray leakage that could not have been predetermined from calculation or simple experiments.

5.4 BREEDING WITH SPALLATION NEUTRONS

Another proposed means of producing fissile fuel is by the accelerator-driven spallation neutron source [18], such as the one set up with the LAMPF accelerator. It is obvious that neutronics, radiation damage, and neutron streaming experiments with the LAMPF facility are then directly relevant to spallation-breeding systems, but they are only marginally (if at all) relevant to tokamak fissile breeding systems. Atomic Energy of Canada Ltd. has been carrying out U-233 breeding experiments with the TRIUMF spallation-neutron source [19]. These experiments, which are motivated by interest in eventually using spallation-bred U-233 fuel in CANDU reactors, are completely analogous to our proposed U-233 breeding experiments on the TFTR.

5.5 T-20 REACTOR

Blanket modules for breeding tritium and fissile nuclides have been planned for the proposed T-20 device in the Soviet Union [20]. The large neutron flux and pulse length (~ 15 s) of T-20 raise the possibility of actually heating the blanket with fusion neutrons. While the flux will be much higher than required for neutronics experiments, the very large number of D-T shots (100,000) would

allow many more types of blankets to be tested, provided that the modules could be replaced sufficiently often. The proposed T-20 blanket experiments are of sufficient interest so that EPRI has agreed to participate in the design of one module to be used for the breeding of Pu-239 in a uranium-oxide medium [21]. However, the T-20 machine has not yet been authorized, and in any event it would come on line at least 5 years after the TFTR.

5.6 RADIATION DAMAGE EXPERIMENTS

The RTNS, LAMPF, and FMIF facilities will be used to obtain radiation damage information on small specimens of structural materials as well as insulators. From these results, one can attempt to estimate the damage that would be expected by the ambient neutron spectrum of a tokamak reactor. These extrapolations can then be compared with the *measured* damage rate to insulators in the TFTR experiments, thereby providing invaluable information concerning the extrapolatability of damage rates incurred by neutrons with a nonfusion spectrum.

While the test volumes in the RTNS, LAMPF, and FMIF facilities are extremely small compared with that available in the TFTR, the orders of magnitude larger annual fluence (see Table 4) could enable a comparable number of small specimens to be irradiated. However, it seems unlikely that insulator testing will receive priority in these radiation damage facilities, which will probably concentrate on the subject of structural material damage under extremely severe conditions.

Insulators of fusion reactor relevance are currently being irradiated in the CFRMF fission reactor at INEL [10]. Samples of each batch of materials to be irradiated in the CFRMF will be retained for eventual placement in small test modules that will be located next to the TFTR vacuum vessel (see Fig. 2). A direct comparison can then be made between the effects of the neutron spectrum in the CFRMF and of that in a toroidal fusion reactor. This comparison should provide important insight into the value of a specially shaped fission spectrum in simulating radiation damage to materials of potential use in toroidal fusion reactors.

6. DOSIMETRY TECHNIQUES

6.1 DOSIMETRY NEEDS

Meaningful and precise neutron spectral determinations and integral dosimetry are vital aspects of the proposed TFTR FAX experiments. The *absolute* neutron and gamma spectra and fluence should be measured throughout the blanket assembly during each experiment. Incident fusion-neutron fluences are required to calculate the breeding ratio. The precision with which this fluence is measured by the TFTR Diagnostics Group [2] will be reflected in the precision with which the breeding ratio is determined.

Both active and passive techniques would be used in the blanket module. Active dosimetry provides a real-time history of neutron flux and is needed also for pulse-to-pulse normalization. Passive techniques integrate the dose over the time period during which the detectors are exposed.

- Examples of Active Detectors

- Neutrons: Fission Chambers, BF_3 Counters, Proton Recoil Counters
- Gammas: NaI , Ge(Li) , Ion Chambers, GM or Proportional Counters

- Examples of Passive Detectors

- Neutrons: Activation Foils, Track Recorders, Helium Accumulation Flux Monitors
- Gammas: Films, TLD's, Electrometers.

Assessment of dosimetry needs requires input from physics calculations — specifically order of magnitude estimates of the neutron and gammas fluxes at representative locations throughout the blanket volume. With this input data, each of the active and passive detectors can be evaluated to assess its sensitivity and spectral response.

The fusion-neutron flux is expected to be toroidally symmetric, but the backscattered flux — that is, fusion neutrons that enter the test volume after scattering and loss of energy in some other part of the machine — may well be toroidally asymmetric, because of the variety of equipment that is attached to the torus. Furthermore, the fusion-neutron flux will probably have a significant

poloidal variation. Hence it is important that the incident neutron flux be monitored at a number of locations in the vicinity of a test module.

A radiochemical laboratory would be used to analyze rabbit samples, as well as to carry out extensive post-test analyses. Possible locations of a radiochemical laboratory at the TFTR site include (1) the Hot Cell Observation Room on the east wall of the TFTR Test Cell, (2) the second floor of the Neutral Beam Power Conversion Building, on the south wall of the TFTR Test Cell, (3) in the diagnostic processing rooms at the east end of the Test Cell.

6.2 ACCURACY AND SPECTRAL UNFOLDING OF INTEGRAL DOSIMETRY MEASUREMENTS

The major components of the inaccuracy in neutron flux measurements are the uncertainties in activation cross sections and in the unfolding techniques. At the present time, two groups that are very active in the field of fast-neutron dosimetry ($E_n \gtrsim 3$ MeV) are located at Hanford-EDL [22] and ANL (Argonne) [23]. Both these groups are working with accelerator-produced point neutron sources. These groups are engaged in the selection of appropriate reactions, determination of accurate cross sections, and estimates of uncertainty in the integral measurements and spectra unfolding. Since this field is evolving rapidly, it is difficult to estimate the degree of advancement to be expected by about 1983. The ANL group has recently reported integral measurements in an accelerator-produced neutron field with accuracies of 10% or better [23]. The uncertainties in the TFTR experiments should be even less, because there one can avoid the thick foil stacks and large flux gradients which have to be faced in experiments with a localized neutron source.

Foil activation techniques are completely insensitive to dose rate. In order to interpret the integral reaction-rate measurements, it is necessary only to know the relative intensity of each pulse and its time-of-day. Thus for a series of N pulses, one has

$$A = \lambda N \cdot \int \sigma(E) \phi(E) dE \cdot \sum_{j=1}^N R_j e^{-\lambda t_j}$$

where

A	= activity at the end of irradiation
λ	= decay constant of the reaction product
N	= number of target atoms
$\sigma(E)$	= cross section for the reaction at energy E
$\phi(E)$	= fluence due to an arbitrarily-defined average pulse
R_j	= intensity of j -th pulse divided by the average pulse
t_j	= length of decay period from pulse j to shut-down

This equation can be solved for $\int \sigma(E)\phi(E)dE$, which can then be unfolded to yield the energy-dependent fluence of an average pulse. Examples of the unfolding of integral data are given in References [23] and [24].

Appendix E gives relevant properties of many of the most promising reactions for use in measuring neutron fluences in the energy range above 5 MeV, by gamma decay techniques. The most sensitive reactions appear to be $^{19}\text{F}(n,2n)^{18}\text{F}$ and $^{63}\text{Cu}(n,2n)^{62}\text{Cu}$.

Recently, an experimental program has been initiated at the Idaho National Engineering Laboratory to develop dosimetry techniques in a simulated blanket module assembly, with a size similar to that appropriate for a TFTR blanket module [25]. The 14-MeV neutron source is located just inside the test module. It is anticipated that many dosimetric techniques proved out in this program can be applied directly for use in the TFTR blanket module. A notable exception concerns certain short half-life activation samples which may not be feasible for use in accelerator-based simulators because of their relatively low neutron fluxes, but which would be attractive in the high-flux but short-pulse TFTR experiments for recording the blanket performance in each pulse.

7. EXPECTED ACCURACY IN DETERMINING BLANKET PERFORMANCE

7.1 PRESENT UNCERTAINTIES IN BREEDING RATIOS IN TOROIDAL REACTORS

In the present context, breeding ratio is defined as the number of tritium or U-233 atoms produced in the reference sector of a blanket module per incident fusion neutron. Today's methods of neutronics analysis are estimated to be capable of predicting tritium or U-233 breeding ratios with an accuracy of ± 10 -15% for simple homogeneous geometric configurations. The accuracy is estimated to be about ± 20 % for a realistic blanket configuration, but benchmarking under actual D-T plasma conditions is needed to provide confidence in the calculated values.

Given the total neutron production rate in a fusion plasma completely surrounded by a homogeneous blanket (with no penetrations) from which the entire tritium or fissile production is extracted, one can predict the gross rate of breeding for the entire blanket with an accuracy close to that of the integral cross sections (± 10 -15%). However, such a global analysis is seriously unrealistic, because of the imperfect blanket coverage in a practical reactor, as well as the heterogeneity and finite dimensions of practical blanket configurations. The global analysis is especially inadequate when fissile or tritium breeding is not performed on the inboard side of the plasma (as is the case in many conceptual reactor designs). In such cases one needs to know the *neutron source geometry*, in order that the neutron flux on a specific breeding region can be calculated, as well as the neutron leakage out of the breeding region. Specific information on *local* breeding performance is vitally important for the proposed "refresh cycles," where spent fuel pins from thermal fission reactors are to be placed in certain regions of the fusion breeder blanket for replenishment of the fissile content [26]. Because of the *additional* uncertainties (± 20 -25%) in modeling the plasma neutron-source geometry and its relation to a practical finite blanket configuration, the absolute production rate of fissile fuel (or tritium) in a specific set of fuel pins in a specific location cannot be predicted to better than a factor of 1.35 to 1.5, for a given total fusion-neutron generation rate.

7.2 DETERMINATION OF BREEDING RATIOS IN THE TFTR

The accuracy in the experimental measurements of tritium or fissile breeding ratio in the proposed TFTR-FAX blanket modules would include the error in measuring

the incident fusion-neutron fluence (spectrum-averaged) for a series of pulses, and the error in performing radiochemical determinations of the bred isotopes. There may also be a small error in integrating the results for the test region. The FAX feasibility study [4], combined with work currently underway at a number of laboratories [22-25], will provide a quantitative estimate of these errors. At the present time, we believe that the incident fluence can be measured ultimately to within $\pm 15\%$, and the radiochemical determinations to within $\pm 10\%$, yielding a combined error of about $\pm 20\%$.

In the TFTR experiments the neutron source geometry will be measured, and this information would be included with the incident neutron flux and breeding measurements to form the benchmark data.

On those occasions when a D-T run (10 to 100 pulses) is planned for the TFTR, satisfactory and reproducible plasma conditions would be established first in deuterium alone. Judging from present beam-injection experiments in PLT and other machines, the fusion-neutron output could be made reproducible to $\pm 15\%$ from shot-to-shot during the D-T run, provided that the beam parameters, programming of the plasma current, and the input gas control are kept constant. However, any shot-to-shot variation in fusion neutron production would actually have insignificant effect on the accuracy of determining the blanket performance, as the neutron flux and fluence incident on the blanket will be monitored during each shot.

From the engineering and economic standpoints, it is extremely important to verify the production rates and the breeding ratios in the lead TFTR-FAX blanket experiments so as to reduce the uncertainty in this area. The accuracies of the measured parameters and calculated performance will improve with time and experience, so that repeat experiments, or analysis of past experiments at a later time, may prove to be invaluable — but provision must be made for doing these experiments in the first place. Accurate knowledge of practical breeding ratios will enable one to stipulate a minimum blanket coverage factor (for either tritium or fissile breeding), which presently has a large uncertainty and a high leverage on the overall plant breeding performance and economic feasibility [27]. As discussed previously, the minimum TBR (as defined here) of a practical fusion reactor will have to be about 1.2. In order to support the growth of a system of pure fusion or hybrid reactors, the performance demand on TBR is even greater [28].

8. IMPACT OF THE PROPOSED FUSION APPLICATION EXPERIMENTS

8.1 TRITIUM BREEDING

The achievement of a tritium breeding ratio of a least unity in a prototypical fusion reactor assembly (i.e. the TFTR) is an essential component of a feasibility demonstration for DT-fueled fusion reactors. Of equal practical importance, the extent to which the breeding ratio can be shown to exceed unity will determine the minimum coverage factor for a breeding blanket. The present uncertainty in calculating breeding ratios in *practical* toroidal reactor blankets — uncertainties due to the geometry of the fusion-neutron source and of the blanket, and the effect of penetrations, as well as the uncertainties in integral cross sections and neutron streaming — will be significantly reduced by the engineering data on tritium breeding obtained in the proposed TFTR blanket module experiments.

8.2 FISSILE BREEDING

At the present time, the fissile breeding performance locally within a heterogeneous blanket on a toroidal fusion reactor cannot be predicted at low risk with an accuracy better than a factor of 1.35 to 1.5. As in the case of tritium breeding, the U-233 breeding experiments in the TFTR will provide practical information that will markedly reduce the uncertainties in breeding performance of toroidal fusion reactor blankets. This information will have direct bearing on the technical feasibility of large-scale U-233 breeding. A demonstration of U-233 breeding ratios of the order of unity would be especially noteworthy if carried out with thorium-oxide pellets that could be refabricated into fuel assemblies for direct use in fission reactors such as the PWR or LWBR, without chemical reprocessing.

Together with the measured plasma and geometrical parameters of the tokamak neutron source, the TFTR blanket breeding experiments will provide a set of neutronics benchmark data that will prove invaluable for comparison with two- and three-dimensional code calculations. In that respect, the TFTR neutronics experiments could be comparable in impact to the now-famous Weale experiment of 1960 which measured neutron-induced transformations in breeding lattices, using a solid-target neutron source [15]. Large deviations of the code predictions from the measured data would have to be reconciled, so that the codes could later be used with confidence to predict the performance of future fusion reactor blankets.

8.3 DOSIMETRY DEVELOPMENT

Special neutron dosimetry will be developed for the blanket experiments, especially for the energy range 3 to 16 MeV. Operational experience with this dosimetry in a tokamak test reactor environment will provide essential engineering data for use in evaluating the relative accuracy and practicality of candidate techniques for monitoring the blanket performance of future test reactors, fusion breeders, and electrical power reactors.

8.4 DESIGN OF NEUTRON SHIELDING

Information gained on fast-neutron streaming and the location of neutron "hot spots" in ducts of various configurations will be invaluable for the design of penetrations in future fusion reactors. Relatively inexpensive experiments on the TFTR could take the place of extremely expensive three-dimensional calculations, and obviate expensive overdesign of shielding in future fusion machines. If more sophisticated neutron transport codes are developed for neutron shielding purposes, the results of the TFTR experiments can serve as benchmark data.

8.5 RADIATION DAMAGE TO INSULATORS

In small test modules on the TFTR, a very large number of insulator specimens can be subjected to neutron doses past damage thresholds. Radiation damage measurements, obtained with a neutron spectrum characteristic of a toroidal fusion reactor, will throw light on the accuracy of extrapolating damage effects obtained by irradiation of similar insulators in fission reactors and with accelerator-based neutron sources. The TFTR radiation damage data will also help determine the best choice of insulator for a given application in future fusion reactors.

8.6 IMPACT ON FOLLOW-ON FUSION REACTORS

It is well recognized that the TFTR plasma experiments will provide an invaluable data base for the plasma operations of the succeeding magnetic fusion reactor, such as an ETF (Engineering Test Facility). Similarly, TFTR blanket experiments could provide invaluable and unique engineering information for the design of the ETF blankets, in the areas of (1) neutronics, (2) dosimetry operations, (3) shielding, (4) activation, (5) remote maintenance, and (6) electromagnetic interactions with the plasma.

Successful operation of the TFTR plasma and of the proposed fuel breeding experiments would help speed the development of a fusion applications pilot plant,

which could have as one of its primary objectives the breeding of multi-kilogram quantities of U-233. The blanket design for such a test reactor would draw heavily on the neutronics, dosimetry, activation, and remote handling information gained from the proposed TFTR experiments, but in addition these blankets would have to demonstrate the capability of essentially continuous heat removal from the neutron power deposition. Table 5 compares important parameters of the TFTR with those of an illustrative fusion applications pilot plant. While the feasibility of the required reactor plasma will be demonstrated in the TFTR, the pilot plant would have a much higher neutron power loading, and more than one order of magnitude greater pulse length and duty factor [29].

8.7 UNFORSEEN RESULTS

A unique and broad experimental program such as the one suggested herein could have unforeseen results with important consequences. An example of a subtle neutronics phenomenon that led to disaster because it had not been discovered in an experimental prototype was the hard γ -ray leakage in the Japanese nuclear-powered ship MUTSU, which compelled mothballing of the ship. This gamma radiation came from ^{16}N , formed in the reaction $^{16}\text{O}(n,p)^{16}\text{N}$ by the high-energy tail ($E_n \geq 9 \text{ MeV}$) of the fission spectrum, which the reactor designers had thought to have no significance. Similarly, there could arise unexpected effects in fusion reactor blankets caused by neutrons of 13 to 16 MeV energy, a range where many cross sections are rising rapidly.

Table 5
Parameters of the TFTR and a Fusion Applications Pilot Plant

	TFTR (1983)	Fusion Applications Pilot Plant (1990)
Major radius (m)	2.5	4.5
Plasma current (MA)	2.5	3.0
Beam energy (keV)	120 (D ^o)	100 keV (D ^o), 150 keV (T ^o)
Beam power	20 MW	50 MW (D ^o), 50 MW (T ^o)
$\langle n_e \rangle \tau_E$ (cm ⁻³ s)	1.5×10^{13}	$\geq 1 \times 10^{13}$
$T_e(0)$ (keV)	10	10
$\bar{E}_{ion}(0)$ (keV)	35	60 (D ⁺), 75 (T ⁺)
Neutrons/s	$\sim 7 \times 10^{18}$	3.5×10^{19}
Neutron wall loading (MW/m ²)	0.1	0.4
Q _p	1.0	≥ 1
Pulse length (s)	1.0	≥ 50 s
Duty factor	0.003	≥ 0.8

9. SCHEDULE FOR IMPLEMENTING FUSION APPLICATION EXPERIMENTS

A planning schedule for implementating the FAX program is shown in Fig. 10. Development and testing of blanket dosimetry would be carried out using fast neutrons from accelerator/target systems, fission reactors, and smaller tokamaks. The first blanket modules would be installed on the TFTR during the shutdown for neutral-beam installation in the first half of 1982. During the presently planned one year of beam-injected deuterium operation, checkout of the blanket experiments, including sample removal for radiochemical analyses, would take place. During deuterium operation, the 14-MeV fusion-neutron flux from burn-up of the triton reaction product is expected to be 2 to 6% of the D-D 2.45-MeV neutron flux. Practice would be gained also with remote handling of the blanket modules during this period. The FAX measurement program discussed in this report would get underway with the beginning of full D-T operation in mid-1983. The state-of-the-art in dosimetry measurement and analysis will continue to improve during the interim period, so that the planning and execution of the TFTR-FAX experiments on the proposed schedule is considered to be prudent and technically sound.

(Actually, the blanket experiments with D-D neutrons are of interest in their own right, as fusion reactors based on catalyzed-D plasmas are preferred to D-T reactors, provided that the neutron wall loading can be brought into the appropriate range, and using a high-gain blanket if necessary [30].)

In order that the FAX program can be implemented on the TFTR on the required time scale (Fig. 10), timely preparations must be made for the assignment of space at the machine for the blanket module, for the provision of physical support for the module, for the supply of utilities, and for remote handling. It must be made convenient to install on-line instrumentation (e.g., rabbit tubes) without interference with various other equipment. In order to determine the mechanical and electrical interfaces of the blanket experiments to the TFTR device, it is necessary to complete a preliminary conceptual design of the blanket module, including significant neutronics analyses.

In addition to the large breeding module, small modules containing many samples of insulators to be irradiated could be placed between the TF coils and vacuum vessel, and left for periods of 1 to 4 years. This space is currently being assigned for heating and cooling ducts, cabling, and other paraphernalia. Design input is needed promptly in order to retain space for these smaller modules, and to insure that adequate instrumentation can be accommodated.

Fig. 10

10. COST ESTIMATE

The cost of performing a particular blanket experiment on the TFTR should be no greater than the cost of a corresponding experiment on an accelerator-based facility. In fact, the cost of the TFTR experiment could be less, inasmuch as neither the capital cost nor the operating cost of the neutron source (i.e., the TFTR plasma) need be shared by the blanket experiments. Remote handling of the blanket modules will be carried out by the TFTR operating crew, just as for diagnostics and other equipment on the machine, and will not incur a special charge. Fully equipped radiochemical laboratories on-site and off-site will be available for use [4].

The cost of the tritium and fissile breeding experiments will comprise (1) neutronics calculations and blanket module design, (2) fabrication of the (common) module, structural support, and utility interfaces, (3) procurement and installation of the blanket materials (a large number of options are available), (4) the blanket dosimetry, (5) module pretesting, (6) operation of real-time dosimetry, (7) post-test analyses, and (8) comparison of experimental results with code predictions.

It is assumed that thorium oxide, in either pellet or powder form, can be obtained from other U.S. Dept. of Energy programs for simply the cost of packaging and transport. Pure lithium oxide can be purchased, but other lithium compounds should be considered as cheaper substitutes. For one set of tritium-breeding experiments (i.e., a single module composition) and one set of U-233 breeding experiments, the total cost of the program is estimated to be roughly \$1.7 million.

Supplementary blanket tests (i.e., with a different blanket composition but using the same module framework and interfaces) as well as neutron streaming experiments would incur additional cost (as would radiation damage testing). The total cost of a FAX program that includes 1 set of tritium and fissile breeding experiments, and 5 additional blanket test series, plus the neutron streaming experiments, is estimated to fall in the range of \$3.5 to \$4 million. This cost is about 1% of the total cost of the TFTR project, including the baseline facility (\$240 million), expected improvements (\$60 million), and operating costs for 5 years (\$160 million).

11. SUMMARY

The proposed TFTR Fusion Applications Experiments (FAX) will provide determinations of practical tritium and fissile breeding ratios in a prototypical reactor blanket module, with a realistic fusion reactor neutron source. The FAX program will also provide neutronics benchmarking, operational experience with blanket dosimetry in a real tokamak environment, investigations of neutron streaming problems, and the study of radiation effects on insulators of fusion reactor relevance. The primary blanket module will be located between adjacent toroidal-field coils, and will be equipped with a replaceable internal composition, so that it can be used for a variety of blanket experiments. While active cooling of the blanket will be unnecessary, a blanket coolant will be incorporated for realistic prototyping of reactor blanket neutronics. Only fertile materials (lithium and thorium) will be used in the blankets and the build-up of tritium or U-233 will be negligible. Smaller test modules for insulator irradiation will be located in "dead spots" between the vacuum vessel and the TF coils.

Accelerator/target neutron sources can provide important energy-dependent cross-section data that can be used in codes for comparison with the neutronic performance of heterogeneous reactor blankets on the TFTR. The RTNS, FMIT, and certain fission reactors can provide radiation damage information that can be correlated with measured damage to arrays of insulator specimens on the TFTR. But the proposed TFTR breeding and other neutronic experiments cannot themselves be done in the accelerator/target systems because of their very small sample volume, unrealistic geometries using point neutron sources, and (in many cases) nonfusion spectra. Only the TFTR device offers the following critical features:

- An extended neutron source characteristic of a magnetic fusion reactor.
- A neutron spectrum (including backscattered neutrons) characteristic of a practical fusion reactor.
- Sufficient test volume for a prototypical reactor blanket, and for irradiation of an arbitrarily large number of insulators of fusion reactor interest.
- Operational experience with blanket dosimetry and with remote handling of prototypical blanket modules in a fusion reactor geometry and environment.

As well as being a testbed for toroidal reactor plasmas, at small incremental cost the TFTR facility can thus serve as the most suitable testbed available for the first fusion reactor blanket modules. None of the components of the TFTR device and systems, such as the vacuum vessel, poloidal-field coils, and neutral-beam injectors, will remain unchanged in design approach as the ultimate toroidal reactor is developed. But neither will there be quantum leaps in design for most of these components in proceeding to the *next* generation of toroidal fusion machines. Similarly, while the TFTR blanket modules will be only crudely prototypical of those *ultimately* used in toroidal reactors, they will provide vital design information for the modules used in the next stage of toroidal reactor development, such as an engineering test reactor.

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Appendix A. PRINCIPAL NUCLEAR REACTIONS

Fusion Reactions

1. $D + T \rightarrow n (14.1 \text{ MeV}) + {}^4\text{He} (3.5 \text{ MeV})$
- 2a. $D + D \rightarrow n (2.45 \text{ MeV}) + {}^3\text{He} (0.82 \text{ MeV})$
- 2b. $D + D \rightarrow p (3.02 \text{ MeV}) + T (1.01 \text{ MeV})$

These two D-D reactions have nearly equal probabilities.

Tritium Breeding

1. $n + {}^6\text{Li} \rightarrow T + {}^4\text{He} + 4.8 \text{ MeV}$
2. $n + {}^7\text{Li} \rightarrow T + {}^4\text{He} + n - 2.8 \text{ MeV}$

Fissile Breeding

1. $n + {}^{232}\text{Th} \rightarrow {}^{233}\text{Th} \xrightarrow[22\text{m}]{\beta^-} {}^{233}\text{Pa} \xrightarrow[27\text{d}]{\beta^-} {}^{233}\text{U}$
2. $n + {}^{238}\text{U} \rightarrow {}^{239}\text{U} \xrightarrow[24\text{m}]{\beta^-} {}^{239}\text{Np} \xrightarrow[2.4\text{d}]{\beta^-} {}^{239}\text{Pu}$

APPENDIX B

ILLUSTRATIVE BLANKET CALCULATIONS FOR FUSION BREEDER SYSTEM

Calculations have been performed to show the rate at which ThO_2 or Li_2O would be converted by neutron absorptions in the blanket surrounding a fusion device. Both fuels were considered to be in the form of zirconium tubes filled with pellets of the appropriate material and placed in contact on a triangular pitch. In this arrangement the fuel volume fraction in the blanket is 0.714.

The system analyzed is shown in Figure B-1. The system was assumed to be radially symmetric and was approximated by a cylinder, but the dimensions are appropriate for the TFTR. The characteristics of the fuel pins are summarized in Table B-1. Water is used as a moderator.

TABLE B-1.

Pin Characteristics

Clad Material	Zr
Clad Od	8.128 mm
Clad ID	7.315 mm
Fuel OD	7.2136 mm
Pin Pitch	8.1280 mm
ThO_2 Density	95% of theoretical
Li_2O Density	90% of theoretical

The 123 group XSDRN library was used in all calculations. The calculations proceeded as follows.

Resonance self-shielding calculations were performed when required using the code NITAWL. The Nordheim integral treatment was used in these calculations.

Using the NITAWL results in the code XSDRN, a cell calculation was performed for the pins. Cross sections for the pin cell were flux and cell weighted and collapsed to 17 groups in this calculation, resulting in cross sections which would be used in a homogenized blanket calculation. In this weighting, a source of 2-MeV neutrons was introduced in the outer portion of the moderator in the cell.

These results were then used in another XSDRN calculation in which the entire system was mocked up as shown in Fig. B-1. In this case a source of neutrons was introduced in the plasma region in the highest energy group, with an average energy of about 14 MeV. The source was normalized to one source neutron. The results are obtained in terms of activations per source neutron, which can be converted to a specific amount of nuclide production.

The results may be summarized as follows:

1. The breeding ratio for the Li_2O is about 1.26.
2. The amount of tritium produced in the sample is about 0.04 μCi per burst per gram of Li_2O .
3. The breeding ratio for the ThO_2 blanket is about 1.07.
4. The amount of protactinium produced in the sample is about 0.12 μCi per burst per gram of ThO_2 .

<u>Blanket Material</u>	<u>Activations Per Source Neutron</u>	<u>Fissions Per Source Neutron</u>	<u>Nuclei Bred Per Source Neutron</u>	<u>Energy Produced Per Source Neutron</u>
ThO_2	1.121	0.047	1.074	8.5 MeV
Li_2O	0.953 - ^6Li 0.311 - ^7Li	—	1.264	4.6 MeV

Figure B-2 shows the spatial distribution of tritium and U-233 breeding in the blanket. The ordinate is calculated assuming a cylindrically symmetric blanket.

Figure B-2.

APPENDIX C

GROUND RULES FOR BLANKET MODULE DESIGN AND OPERATIONS

- (1) The TFTR operating schedule will determine the schedule of blanket experimental operations.
- (2) All materials inside the igloo shield will be nonmagnetic and compatible with temperature and other environmental factors predicted by TFTR personnel.
- (3) Physical contact of the experimental module with the vacuum chamber is not permissible.
- (4) Normal laboratory services — electricity, air, water — are available to support the blanket experiments.
- (5) Shield penetrations for blanket electrical lines, pneumatic rabbit tubes, etc., will be made available.
- (6) Remotely operated manipulating devices capable of positioning a 2-ton load as required by the experiment will be available.
- (7) Provision can be made for insertion and removal of samples at any time between pulses. Some categories of samples will be removed only during scheduled maintenance periods.
- (8) Rapid-access samples will not be hot enough to require remote handling during transfer analysis.
- (9) Fuel rods will be handled only during TFTR maintenance periods. Lithium-bearing rods will not be sufficiently radioactive to require remote handling for removal, but thorium-bearing rods will require remote handling.
- (10) The blanket modules will require remote handling upon removal, and replacement assemblies must be installed remotely.
- (11) The blanket module construction will be such as to have insignificant effect on TFTR electromagnetic parameters.

Appendix E

14-MEV NEUTRON ACTIVATION REACTIONS

	Threshold MeV	$\sigma(14.4 \text{ MeV})$ Barns	Product's Decay Scheme		
			$T_{1/2}$	γ -KeV	Intensity
$^{19}\text{F}(n,2n)^{18}\text{F}$	11.6	.05	110m	511	1.94
$^{24}\text{Mg}(n,p)^{24}\text{Na}$	5.0	.18	15h	1368	1
$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	4.6	.12	15h	1368	1
$^{35}\text{Cl}(n,2n')^{34m}\text{Cl}$	12.1	.006	32.4m	145.6	.22
$^{39}\text{K}(n,2n)^{38}\text{K}$	13.1	.003	7.71m	2167	1
$^{45}\text{Sc}(n,2n')^{44m}\text{Sc}$	11.6	.12	2.44d	271.2	.86
$^{48}\text{Ti}(n,p)^{48}\text{Sc}$	3.2	.1	1.83d	983.5	1
$^{55}\text{Mn}(n,2n)^{54}\text{Mn}$	10.0	.78	312d	834.8	1
$^{56}\text{Fe}(n,p)^{56}\text{Mn}$	3.5	.11	2.59h	846.6	.99
$^{59}\text{Co}(n,\alpha)^{56}\text{Mn}$	5.0	.030	2.59h	846.6	.99
$^{59}\text{Co}(n,2n)^{58}\text{Co}$	10.8	.63	71.3d	810.8	.99
$^{58}\text{Ni}(n,2n)^{57}\text{Ni}$	12.4	.05	36h	1377.6	.85
$^{58}\text{Ni}(n,np)^{57}\text{Co}$	6.0	.66	270d	122.0	.85
$^{63}\text{Cu}(n,2n)^{62}\text{Cu}$	11.9	.61	9.7m	511	1.95
$^{70}\text{Ge}(n,2n)^{69}\text{Ge}$	11.5	.57	38.5h	574	.13

Reaction	Threshold MeV	$\sigma(14.4 \text{ MeV})$ Barns	Product's Decay Scheme		
			$T_{1/2}$	γ -KeV	Intensity
$^{75}\text{As}(n,2n)^{74}\text{As}$	10.3	.9	17.8d	595.9	.59
$^{99}\text{Tm}(n,2n)^{98}\text{Tm}$	11.5	.84	106.6d	898	.92
$^{90}\text{Zr}(n,2n)^{89}\text{Zr}$	12.2	.60	78.5h	909.2	1
$^{107}\text{Ag}(n,2n)^{106}\text{Ag}$	9.6	.58	24.1m	511.7	.17
$^{127}\text{I}(n,2n)^{126}\text{I}$	9.3	1.5	13d	388.6	.35
$^{133}\text{Cs}(n,2n)^{132}\text{Cs}$	9.0	1.5	6.47d	464.5	.0197
$^{169}\text{Tm}(n,2n)^{168}\text{Tm}$	8.1	2.1	87d	198.2	.534
$^{175}\text{Lu}(n,\alpha)^{172}\text{Tm}$	7.8		63.6h	1093.8	.10
$^{175}\text{Lu}(n,2n)^{174}\text{Lu}$	7.8	1.8	3.6y	1241.8	.09
$^{186}\text{W}(n,2n)^{185}\text{W}$	7.3	.002	75.1d	125.4	.00017
$^{197}\text{Au}(n,2n)^{196}\text{Au}$	8.1	2.2	6.18d	355.7	.88
$^{203}\text{Tl}(n,2n)^{202}\text{Tl}$	7.8	2.1	12d	440.2	1

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