ENGINEERING TESTING REQUIREMENTS IN FED/INTOR

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ABSTRACT

The FED/INTOR Critical Issues activity examined three key testing requirements that have the largest impact on the design, operation and cost of FED/INTOR. These are: 1) the total testing time (fluence) during the device lifetime, 2) the minimum number of back-to-back cycles, and 3) the neutron wall load. These requirements were quantified by investigating the benefits/risks to the DEMO from testing structural materials, blankets, and main reactor components in FED/INTOR.

I. INTRODUCTION

One of the primary objectives of the next major fusion device, e.g., FED or INTOR, is to serve as a test facility for demonstrating the performance of engineering components. Engineering testing imposes a number of requirements that have a large impact on the design, operation and cost of FED/INTOR. The FED/INTOR activity in 1982 examined[1] the testing requirements in three key areas. The first area, summarized in Sec. II, is concerned with estimating the benefits to the DEMO from testing structural materials in FED/INTOR to various goal fluences. The second area, Sec. III, is focused on quantifying the importance of the neutron wall load and testing time (fluence) to blanket testing. The third area, Sec. IV, attempts to quantify the benefits to DEMO of long-term operation of FED/INTOR components.

II. STRUCTURAL MATERIALS - FLUENCE CRITERIA

The development of structural materials for fusion is a highly ambitious task. The step to fusion power reactors implies aggressive goals for such major service parameters as radiation lifetime and operating temperature. However, prior to the existence of such reactors there will be no "fusion reactor" quality testing environment. To the degree that relevant experience with fusion radiation effects is not acquired, there will exist some risk, perhaps of grave proportions, in the selection of materials for future fusion reactors. This assessment was intended to calibrate the dimensions of that risk.

The general objective of this task was to estimate the benefit for a DEMO reactor resulting from structural materials tests in FED/INTOR. The "benefit" is simply the reductions of uncertainties in predicting materials' performance in DEMO (with a design lifetime of 10 MWy/m^2) that would result as the test fluence available in FED/INTOR was increased from 0 to 6 MWy/m^2.

A. Method of Evaluation

Two materials, Ti-modified Type 316 SS and vanadium (alloys), and four areas of materials performance, i.e., tensile properties, swelling, fatigue and crack growth, were selected to provide representative information on critical materials issues related to testing in FED/INTOR. The two materials present distinctly different challenges for development. Stainless steel has a substantial data base that includes irradiation effects and there is a general consensus that tests in fission reactors can provide much of the information needed for fusion applications because of the capability to simulate key radiation effects (helium). On the other hand, testing of advanced (non-nickel bearing) alloys such as vanadium (or niobium, molybdenum, ferritic steels, etc.) in fission reactors suffers from serious shortcomings in simulating the fusion environment. Furthermore, the volume of data needed from tests of these advanced alloys is quite large since the present data base is poor.

The fluence dependent models provide a useful quantitative method for estimating uncertainties. However the models themselves are not yet confirmed. Consequently there is a variety of caveats implied in using this method. The important fact here is that this method has supplied a semi-quantitative articulation of the general perceptions that must be used to make this type of judgment in situations where hard data is lacking.

Two examples of materials data projections (for Type 316 stainless steel) demonstrate the method used to quantify uncertainties: swelling represents a long-term effect, i.e., one observed only at high fluences in this material, and the effect of radiation on tensile strength represents a short-term or low fluence effect. Only swelling will be described here due to space limitation.

The high swelling of 316 SS at temperatures above 450°C led to development of titanium modified steel with higher nickel content than 316 stainless steel. The expected performance range of these steels at 300°C in fusion reactors is shown in Fig. 1.

In projecting uncertainties in the amount of swelling that would occur after an exposure of 10 MWy/m^2 (DEMO lifetime), a bilinear swelling model, with major parameters of incubation time (fluence) and swelling rate R varying within their allowable limits, was applied to data sets where the maximum assumed test fluence 0.1, 1, 2, 4, 6 and 8 MWy/m^2. In Fig. 1 the projected range (at 10 MWy/m^2) is from about 2 to about 8 percent swelling and the limit of our current data is a (fission) fluence equivalent to about 4 MWy/m^2.

If data were continuously available (at all fluences), then the reduction in uncertainty would decrease smoothly as these swelling data became available at progressively higher fluences as suggested by the solid line Fig. 2. A more realistic scenario is that the reductions in uncertainties will come in a series of steps as data become available, as shown by the dotted lines in Fig. 2 for discrete data at 4, 6 and 8 MWy/m^2.

Figure 2 was obtained by evaluating swelling models that predicted data within the overall range of Fig. 1. Since these detailed evaluations are omitted here Fig. 2 cannot be directly derived from Fig. 1. There is a sharp reduction in uncertainty when the data at a fluence of 6 MWy/m^2 is added to previous data with fluences up to 4 MWy/m^2 (the incuba-

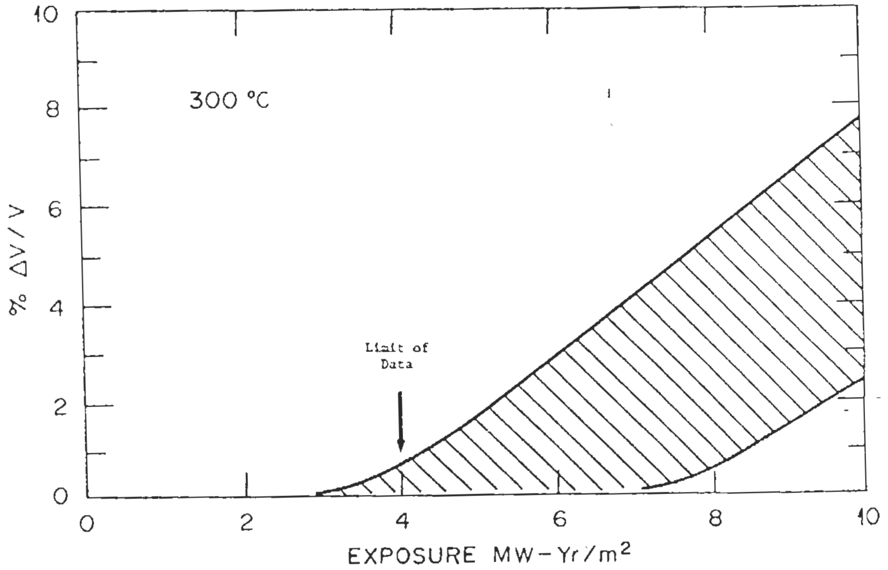

Fig. 1. Projected swelling for Ti-modified austenitic SS in fusion reactor service based on behavior of similar alloys in EBR-II and HFIR at low exposures.

tion period), as shown in Fig. 3. (There would be further reduction at 8 MWy/m^2; however, 6 MWy/m^2 was assumed to be the maximum possible exposure on FED/INTOR.)

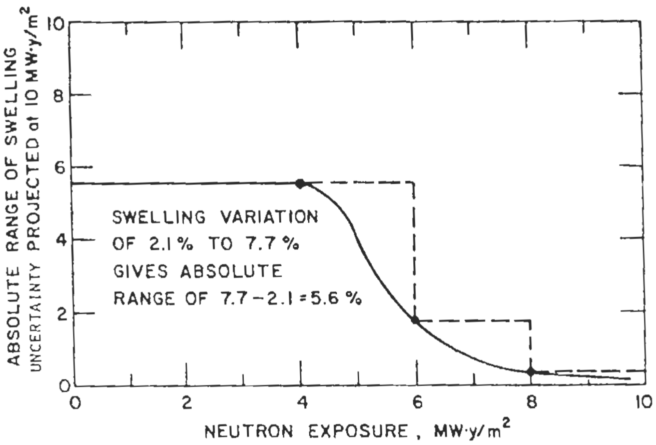

Fig. 2. Absolute range of uncertainty in projected values of swelling at 10 MWy/m^2 as a function of the maximum fluence of data available to make the projection. Dotted lines show stepwise reductions for discrete data at 4 (no swelling), 6 and 8 MWy/m^2.

Because the data available on neutron radiation effects in vanadium alloys is extremely limited, a judgment was made that the methodology used to evaluate the reduction in uncertainties for the data on stainless steel could not be effectively applied to the case of vanadium alloys. However, general conclusions about testing needs and the role of FED/INTOR are still supportable and appropriate. For a DEMO, which is currently perceived as a reactor having an equivalent lifetime first wall fluence 10 MWy/m^2, a minimum experience level of 6 MWy/m^2 should be available if materials selections are to be made with an acceptable degree of confidence.

B. Conclusions

Table 1 summarizes the role of testing at various fluences and the related changes anticipated in materials properties from exposure to fusion neutron irradiation. The judgments on "Benefits of Testing" are based on available data from fission reactor irradiations.

The projections of uncertainties in data on structural materials has provided a useful semi-quantitative method to establish criteria for goal lifetime(s) for FED/INTOR based on its role in providing test data on structural

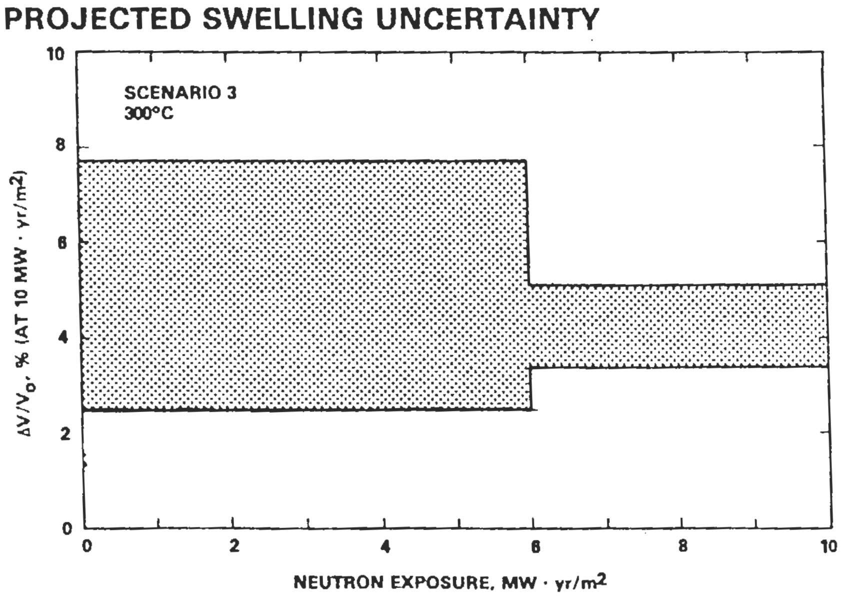

Fig. 3. Uncertainty in swelling values at 10 MWy/m^2 as a function of the maximum test fluence in FED/INTOR used to make the projections of swelling (at 10 MWy/m^2). In scenario 3, the data were assumed to lie in the middle of the range of data shown in Fig. 1.

materials. Separate criteria are necessary for stainless steel and for vanadium (or other advanced alloys not containing nickel) as summarized in Table 2.

TABLE 1. CHANGES IN MATERIAL WITH FLUENCE

Maximum Test Fluence (MWy/m^2)	Benefits of Testing
Stainless Steel	
0 - 1	Little useful information
1 - 3	Confirmation of low fluence effects (e.g., tensile properties) predicted with other sources
3 - 6	Model verification from observations of microstructure preceding long term changes in behavior.
Above 6 MWy/m^2	Confirmation of performance near end of life (e.g., high swelling).
Vanadium Alloys	
Up to 1.5 MWy/m^2*	Tensile properties still change.
Up to 3 MWy/m^2*	No swelling observed.

* Tests in range of 500-600° C.

TABLE 2. FLUENCE CRITERIA FOR FED/INTOR LIFETIME

	With FMIT
Stainless Steel	3 MWy/m^2: Low risk 0 MWy/m^2: Low → medium* risk
Advanced Alloys	?? (3-5 MWy/m^2): ?? Medium risk
	Without FMIT
Stainless Steel	3 MWy/m^2: Low → medium risk
Advanced Alloys	3 → 6 MWy/m^2: High → medium risk

* Medium risk is defined as a good chance that material will exceed 50% of design life but significant risk that 100% will not be met.

The general consensus among the U.S. materials experts contributing to this task was that the existing data base on stainless steel coupled with a future vigorous research program in fission test reactors could provide sufficient knowledge to proceed with the construction of a DEMO using stainless steel with only moderate risk. The recommended "low risk" test scenario would involve a complementary test program using fission reactors and FMIT in conjunction with tests in FED/INTOR on mechanical properties in the range of 2-3 MWy/m^2. The use of FMIT would provide high fluence, high energy data to confirm end-of-life trends in swelling and confirmation of models of mechanical properties through correlations based on microstructures and hardness and bend tests.

Two other alternatives are also noted in the table. First, given the availability of FMIT, the potential for establishing a design data base for the DEMO without appreciable testing in FED/INTOR (0 MWy/m^2) was given some credence. However, the lack of a large test volume (FED/INTOR) to provide engineering data with large number of specimens and specimens of a standard design (as opposed to miniature specimens in FMIT) for this option was felt to increase the risk. In the second alternative, without FMIT, the same fluence criterion of 3 MWy/m^2 is cited but with increased risk compared to the reference case with FMIT. The increased risk is associated with the inability to confirm trends in the behavior of stainless steel at high fluences in a fusion environment.

In considering these same questions with regard to development of vanadium alloys it is possible that high fluence data available from FMIT would establish that trends in radiation effects on critical properties of the advanced alloys would saturate in a range of moderate fluence, i.e., within the capability of FED/INTOR testing, and that a program consisting of benchmark tests on alloy candidates in FMIT with subsequent extensive tests in FED/INTOR could develop a suitable advanced alloy for DEMO. A problematic aspect of this testing scenario is whether data from FMIT (or some other fusion-like source) that would define the useful upper limit for the goal exposure in FED/INTOR tests could be made available at the time when decisions concerning the design lifetime of FED/INTOR were being made.

III. BLANKET TESTS

Three objectives related to tests of tritium breeding blankets were addressed: 1) continuous operating time requirements; 2) benefits/risks of blanket long-term tests; and 3) importance of power density in blanket tests. Candidate blanket tests for FED/INTOR were categorized into the five major areas discussed in Section III-A. The tests involved would be performed to fulfill one or more of the following test goals: 1) correlation of test results from FED/INTOR with those from tests in complementary facilities, 2) verification of predictive analyses, and 3) performance demonstration of DEMO blanket prototypes

The DEMO blanket evaluated was the lithium oxide (Li_2O) solid breeder blanket described in the STARFIRE/DEMO Interim Report.[2] Fluence ranges and specific neutron wall load values evaluated for FED/INTOR[3] are listed in Table 3, together with DEMO goals for those parameters.

TABLE 3. OPERATING CONDITIONS ASSUMED FOR FED/INTOR AND DEMO

Category	Units	Proposed Values	
		FED/INTOR*	DEMO
a	MW/m^2	0.4 to 1.3	2.0
b	MWy/m^2	0.2, 1.0, 2.0, 4.0, 6.0	10.0
c	%	20 to 35	50
d	y	≤ 15	10

a - Neutron Wall Load, P_{nw}
b - Fluence
c - Availability
d - Operating Life

*Values for FED/INTOR must be self-consistent.

The following section discusses the results from evaluation of a small number of candidate tests which were relevant to one or more test goals.

A. Discussion of Results

1. Neutronics Tests (Tritium Breeding, Nuclear Heating, Neutron Spectrum, Activation and Decay Heat). To verify the predicted tritium breeding capability of the DEMO blanket, the local tritium production rate will be measured at different locations in the blanket. These measurements, performed using radiochemical and mass spectrometric techniques, will then be compared with results of 3-D neutronics analyses for the blanket. The fluence required is ~ 1 $MW\text{-}s/m^2$, a fraction of a pulse for FED/INTOR conditions. These measurements can be performed during the early stage of operation with only a few very short pulses.

Calorimeters and thermoluminescent dosimeters (TLD) have been considered for local nuclear heating measurements. For FED/INTOR heating rates near the first wall, a calorimeter response time of ~ 1 s is expected. This response and the FED/INTOR burn times of 100 to 200 seconds allow measurement within one pulse without difficulty.

Two methods were considered to measure the fusion neutron spectrum: multi-material activation and gas recoil proportional counters. Calorimeters and TLD's can be used to measure decay gamma heating with ~ 3% accuracy after shutdown since the radiation field is then neutron-free. The evaluations indicated that a very small fraction of 1 MWy/m^2 is adequate to perform neutron spectrum, activation, and decay heat measurements in FED/INTOR.

2. Tritium Recovery Tests. The effects of FED/INTOR operating characteristics on the capability to perform steady-state (or quasi-steady-state) tritium recovery (STR) tests were evaluated. Principal factors considered were 1) the number of burn cycles required to build up breeder tritium inventory to the steady-state level, 2) the impact of test module thickness (depth) on the time required to obtain a uniform steady-state tritium inventory throughout the test module, and 3) the number of full power pulses needed in order to stabilize the tritium release characteristics.

Because neutron flux decreases exponentially with depth into the blanket, the relative time scales for tritium inventory buildup as a function of depth must be considered. For projected FED/INTOR operating conditions, it will take ~ 25 times longer to achieve the steady-state inventory (assumed equivalent to ~ 1 wppm) at the back of a DEMO blanket test module (~ 45 cm thick) than at the front. The continuous operating time required to achieve the STR inventory level across a full-depth DEMO module will be ~ 62 days in FED (see Fig. 4) and ~ 17 days in INTOR. If STR levels turn out to be ~ 10 wppm, these times would increase by a factor of ten. The requirements on the FED would then be unacceptable; a significantly higher P_{nw} value would be necessary to make the requirements for continuous operation acceptable. Some advantage can be gained from testing DEMO-simulation specimens reduced in depth to ~ 5 to 10 cm; these could be irradiated to the STR level in considerably less operating time that full-depth specimens.

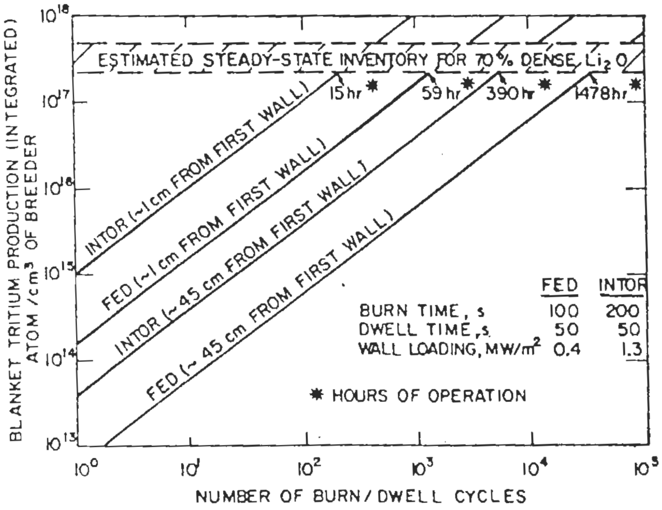

Fig. 4. Blanket tritium production vs. number of burn cycles for selected test locations in FED and INTOR.

The third factor evaluated was the number of full power pulses needed to supply a fluence corresponding to estimated property-change saturation levels in Li_2O (~ 0.2 MWy/m^2; see Fig. 5). The equivalent times required to reach saturation are ~ 4 y for FED and ~ 0.5 y for INTOR for 5-10 cm depth modules. However, many aspects of this issue remain to be resolved.

3. Materials Compatibility Tests. Two types of tests in this category were evaluated: 1) coolant tritium level, and 2) purge system corrosion and mass transfer.

The blanket coolant tritium level is considered to be a function of 1) coolant tube permeability, 2) tritium partial pressure in the helium purge gas, and 3) coolant tube oxide film permeability. No significant change in stainless steel permeability is expected to occur for fluences up to 10 MWy/m^2. The partial pressure of tritium in the blanket is not expected to change with increasing fluence, once STR conditions are reached. Permeability of the oxide film on the breeder side of the coolant tube could possibly change with fluence, but no relevant data exist. Any changes in permeability of the oxide film on either side of the coolant tube should be measurable from off-line and fission reactor blanket tests. Corrosion of structural material within the blanket and in the purge system is expected to be a function of the partial pressure of LiOT in the purge stream. The LiOT partial pressure, and the associated mass transfer rate of LiOT from the

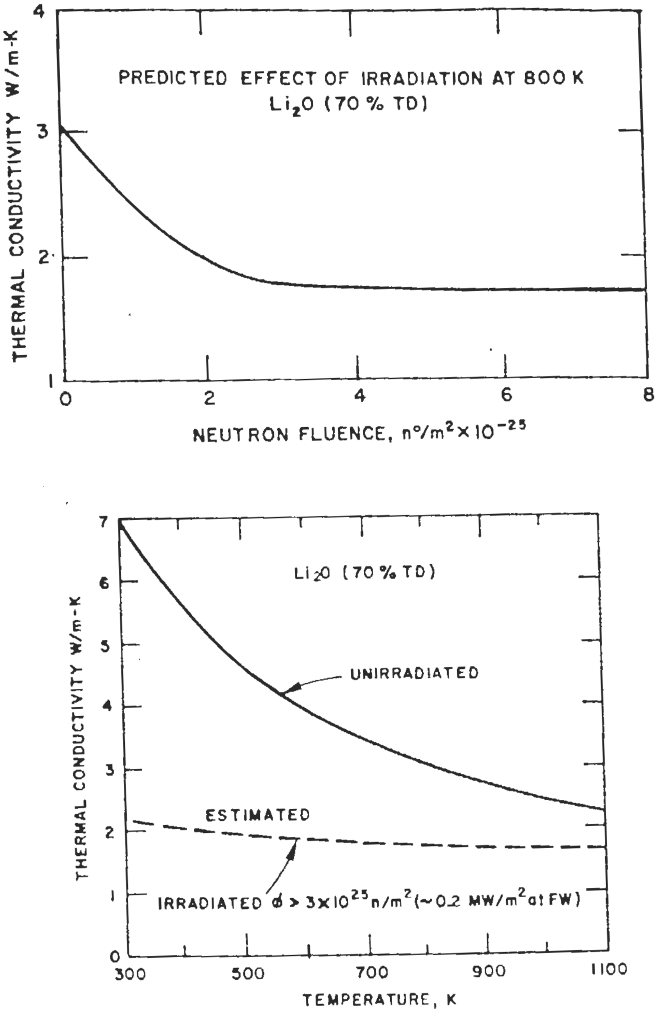

Fig. 5. Estimated effects of fluence on Li_2O thermal conductivity.[4]

Li_2O, are in turn functions of the partial pressure of moisture (T_2O)[4], which is estimated to remain essentially constant as long as STR conditions are maintained. Thus, coolant tritium levels corrosion and LiOT mass transfer rates 1) should not change significantly with fluence once steady-state temperature and tritium recovery conditions are reached in the blanket, and 2) should be adequately predicted for the fusion environment from the results of tests in fission reactors.

4. Heat Recovery Tests. Analysis of DEMO blanket operation under FED/INTOR conditions indicated that exact duplicates of DEMO blankets cannot be properly tested under FED/INTOR conditions. Because of both the reduced P_{nw} level and the cyclic operation of FED/INTOR, a large fraction of the volume of solid breeder will remain below the 410°C T_{min} limit during operation. This would invalidate the results of all tests except neutronics. Therefore, test assemblies to simulate the DEMO blanket in non-neutronics tests in FED/INTOR should be specifically designed to be operated at one pre-selected FED/INTOR P_{nw}/burn cycle combination. Table 4 lists the minimum continuous operating time and cycles necessary in FED/INTOR to reach steady-state coolant and breeder temperatures in the region of maximum nuclear heating in the test assembly. Regions deeper into the test assembly would require more time and additional cycles to reach their steady-state temperatures.

5. Lifetime Tests. The specific tests evaluated concerned 1) breeder physical integrity, and 2) breeder-to-structure interface thermal conductance.

Evaluation[5] of monolithic solid breeder cylinders surrounding blanket coolant tubes indicated that the maximum temperature gradient of 250°C radially through the cylinder would result in maximum axial and circumferential thermal stresses greater than the al-

TABLE 4. FED/INTOR OPERATING REQUIREMENTS BASED ON BLANKET TESTS EVALUATED[c]

	Test Requirements[c,d]					
	Minimum Continuous Operating Time		Minimum Continuous Operating Cycles		Minimum Useful Wall Load[a]	
Test Category	P_{nw} = 0.4, 100 s Burn	P_{nw} = 1.3, 200 s Burn	P_{nw} = 0.4, 100 s Burn	P_{nw} = 1.3, 200 s Burn	DEMO Simulation Specimen	DEMO Duplicate Specimen
Neutronics	b	b	b	b	0.4 MW/m^2	0.4 MW/m^2
Tritium Recovery[e,g]	59 hr (~ 109 hr)	15 hr (~ 65 hr)	1400 (~ 2600)	220 (~ 950)	0.4	NOT ACCEPTABLE
Materials Compatibility	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE	0.4	NOT ACCEPTABLE
Heat Recovery[f]	~ 3600 s	~ 1230 s	~ 24	~ 5	0.4	NOT ACCEPTABLE
Lifetime	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE	0.4	NOT ACCEPTABLE

a – Range considered: $0.4 \leq P_{nw} \leq 1.3$ MW/m^2. (Lower values may also be acceptable.)
b – A few pulses are sufficient in general. See text.
c – Fluence requirements and effects are not included.
d – Cyclic conditions assumed per Figure 4.
e – Time and cycles are for minimum-depth (~ 5-10 cm) test specimens; thicker specimens require longer operation. Effects of breeder radiation damage are unknown and therefore are not included.
f – Time and cycles are for maximum nuclear heating region only. Regions deeper into test specimen will require longer times and additional cycles.
g – Number in parentheses include 50 h allowance after blanket production has reached steady-state inventory level, to permit achievement of steady-state recovery rates. Inventory level assumed equivalent to ~ 1 wppm. Times and cycles are directly proportional to the actual STR level (see text).

lowable fracture stress by factors of 20 or more. This indicates the need for segmentation of the breeder both axially and circumferentially; the degree to which this is necessary remains to be determined. However, the results of fission reactor tests are believed adequate to predict fracture stresses for solid breeders irradiated in fusion reactors. The breeder-to-tube interface proposed for the DEMO blanket consists of a stainless steel metallic felt, bonded to the breeder and to the coolant tube. Fluence is not expected to affect the conductivity of this assembly. Any postulated major effects would be mechanical, e.g., failure of the bond. The blanket development program would address methods for avoiding such failures, and would use results from fission reactor tests to confirm the long-term performance of the interface. Thus, tests of both breeder physical integrity and the interface in FED/INTOR to fluences less than 10 MWy/m^2 should not pose a significant added risk to DEMO.

B. Conclusions

The conclusions regarding FED/INTOR blanket tests are strongly influenced by the estimates made within this study regarding the relationship of fluence level to property changes in solid breeders (Fig. 5), and the expected correspondence of fission and fusion test results for solid breeders. If future tests show these estimates to be incorrect, the conclusions must then be re-examined.

1. Estimated requirements for FED/INTOR minimum continuous operating times and minimum useful wall load are shown in Table 4, for the assumed cyclic operation conditions (see Fig. 5).

2. The benefits (in terms of reduced risks for DEMO) from conducting blanket tests in FED/INTOR to fluence levels significantly beyond $\sim 6 \times 10^{25}$ n/m^2 (~ 0.4 MWy/m^2 at the first wall) cannot be quantified at present.

3. DEMO-simulation specimens should be designed for and operated at a specific FED/INTOR P_{nw}/burn cycle combination in order to produce usable, quantifiable test results for all non-neutronics tests evaluated.

4. Tritium recovery tests that utilize reduced-depth specimens to obtain steady-state tritium recovery at conditions equivalent to 1 or 10 wppm appear acceptable for FED or INTOR in terms of requirements imposed on machine continuous operating times.

IV. LONG-TERM COMPONENT OPERATION

The benefits to DEMO of long-term operation of FED/INTOR components were investigated. An initial review indicated that most design verification, analytic tool confirmation, performance validation and design improvement needs could be determined during a few months of FED/INTOR operation. Benefits that required long-term operation were largely confined to determining failure modes, failure rates, failure recovery times and life. These can generally be classed as reliability data.

When reliable components were required for the Minuteman missile, an extensive reliability development program was undertaken where hundreds of duplicate parts were tested in environments equal to or more severe than those expected during normal operation. Fusion, however, has a unique challenge in developing reliability data for its components. The sheer size and related costs of most fusion components limits the ability to use many duplicate parts, and the scaleup nature of fusion reactors makes duplicating or "over-stressing" of the environmental conditions impractical. The process of reliability data development was therefore reviewed to determine what might be done with FED/INTOR for the DEMO reactor.

Reliability testing procedures for MIL STD 781 indicate that testing of components at the anticipated operating conditions for a time equivalent to ~ 3.5 times the goal mean-time-between-failure (MTBF) will provide approximately an 80% confidence level that the components will actually exhibit an MTBF that is equal to or larger than the goal. The MTBF requirements of the non-nuclear DEMO components and systems indicates that adequate reliability data can be derived for most major reactor systems, including magnets, during a total reactor operating time of ~ 20,000 h. No data is obtained for long-life components within those systems, however, because the overall failure rate of major systems is high even though some of their components fail infrequently. The components that do fail frequently can be identified through testing, and their failure modes and rates determined with adequate confidence. Components that fail infrequently will not be operated long enough in FED/INTOR to determine what the failure rate might be. A list of the required test times for each major system is shown in Table 5. Components requiring test times greater than 46,000 h, the maximum FED/INTOR operating time considered, are omitted from this list. These test times, 3.5 times the goal MTBF's of DEMO, were established by scaling the FED/INTOR MTBF data to DEMO considering the availability goals and replacement times of each. Components such as TF coil windings, which are expected to have low failure rates but require very long replacement times, could still dominate DEMO availability. Test times of ~ 350,000 h are required to develop adequate data on these components.

TABLE 5. FED/INTOR TEST TIME REQUIRED TO OBTAIN AN ADEQUATE DATA BASE FOR DEMO

ITEM	TEST TIME REQUIRED (h)
Overall FED/INTOR System	504
Pellet Injectors	1,750
System Interface Monitor & Control	2,700
ICRH System	5,880
Computers, Consoles, & Displays	7,000
First Wall System[a]	7,000
Heat Transport System	7,000
Cryogenics System	7,000
EF Coil System	9,800
ECRH System	12,068
First Wall Panels[a]	15,400
ICRH Reactor Building Transmission	17,500
Diagnostics	17,500
TF Coil System	18,900
OH Coil System	21,000
Armor Tiles[a]	21,000
Spool Structure System	22,400
OH Power Conversion/Protection	23,000
TF Power Conversion/Protection	28,000
Pumped Limiter	28,000
Vacuum Pumping System	28,000
ICRH Control	35,000
ICRH Wave Guide Launcher	35,000
Sector Modules	40,600
ICRH Frequency Generator	46,000

[a] Neglecting irradiation effects.

Implementation of a program for reliability development is essential to fusion's progress. Such a program should include more comprehensive analysis and test efforts to generate new data and to permit comparison with data from other sources (e.g., fission) and a means of gathering and documenting data from present fusion experiments. This effort would provide a data base for reliability estimates and would identify particularly troublesome components or design features that should be changed in next generation designs.

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