PANEL PROCEEDINGS SERIES

INTERNATIONAL TOKAMAK REACTOR
Phase Two A, Part I

REPORT OF THE
INTERNATIONAL TOKAMAK REACTOR WORKSHOP
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN SEVEN SESSIONS IN VIENNA
DURING 1981 TO 1983

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Chapter XII

ENGINEERING TESTING

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1. INTRODUCTION

During Phase Two A, the work on the engineering testing programme concentrated on a number of key issues. These issues relate to the significance and type of information to be provided by INTOR operation for the subsequent construction of DEMO. To develop a common thinking it was necessary to outline the characteristics of the DEMO.

The DEMO reactor considered here is a fusion reactor to be constructed following the successful operation of the next-generation machine, such as INTOR, before the commercialization of fusion reactors occurs. The DEMO is, therefore, an important milestone in fusion reactor development. The concept of such a DEMO should be essentially the same as that of a commercial reactor, although scaling up of the systems and components may be necessary for commercialization. The DEMO has the following general objectives:

(a) to demonstrate feasibility of safe and reliable operation of a fusion power plant that can be extrapolated to a commercial reactor;
(b) to demonstrate the adequacy of the technological data base and of methods involving physics and engineering for designing, fabricating and operating the systems and components that can be extrapolated to a commercial reactor;
(c) to demonstrate the closed fuel cycle of a fusion reactor;
(d) to provide information for estimating the economics of a commercial fusion reactor.

In the light of the present state-of-the-art of the fusion technology and the objectives of the DEMO, an attempt was made to outline the major characteristics of the DEMO as follows:

(1)	Fusion power	1000 ~ 2000 MW(t) (300–800 MW(e))
(2)	Tritium breeding ratio	larger than 1.0
(3)	Neutron wall load	$\approx$ (2–3) $MW \cdot m^{-2}$
(4)	Heat load on first wall	$\approx$ 0.8 $MW \cdot m^{-2}$
(5)	Operational mode	steady state or quasi-steady state
(6)	No. of cycles in lifetime	$\lesssim 10^5$
(7)	Availability	$\approx$ 50–70%
(8)	Neutron fluence on first wall	$\approx$ 10–15 $MW \cdot a \cdot m^{-2}$

The test programme activities during Phase Two A were structured into four tasks. The first task has the general objective of estimating the benefits for the DEMO obtainable from testing structural materials in INTOR to various fluences. The principal measure of benefit is taken to be the reduction of uncertainties in predicting the structural material performance at the end of the DEMO life, (10 $MW \cdot a \cdot m^{-2}$). Two structural materials are considered: austenitic stainless steel and an advanced alloy (vanadium alloy). The primary differences between the evaluations for the two materials are the larger data base and the more meaningful simulation in the fission environment for the nickel-containing alloy. The impact of the availability of an accelerator-based 14 MeV neutron source on the testing requirements in INTOR is also considered.

Evaluations of the irradiation-dependent properties of the first-wall material showed the interdependence of the risk for INTOR and for DEMO. Increasing the testing fluence in INTOR decreases the uncertainty of the extrapolation of radiation effects to the DEMO fluence level. An attempt has been made to develop a strategy for the development of fusion reactor materials. It appears evident that such a strategy cannot rely exclusively on the tests to be carried out in INTOR. The role of INTOR is to serve as a source of a real fusion environment to calibrate and confirm results from simulation facilities.

The second task is concerned with testing requirements for the blanket with emphasis on five specific areas: 1) adequate tritium production, 2) acceptable tritium inventory, 3) compatibility of materials, 4) efficient heat recovery, and 5) acceptable and sufficient lifetime.

The objective of the third task is to quantify the benefits of long-term operation of INTOR components to DEMO. The concentration was on defining DEMO requirements and determining how long components must be operated in INTOR in order to provide confidence that they will operate satisfactorily in DEMO.

The objective of the last task is to design a blanket module for simultaneous tritium breeding and electricity generation. The construction of such a module has been evaluated and seems feasible.

Sections 2 through 4 describe the results of the first three tasks. The Appendix provides the results of the last task.

2. STRUCTURAL MATERIALS TESTING

2.1. Introduction

INTOR is currently perceived as the only 'engineering' fusion device between the next generation of fusion devices (TFTR, JT-60, JET) and the DEMO. The role which INTOR should play in developing the required data base on neutron radiation effects in candidate fusion reactor structural materials for a DEMO is a critical issue in establishing a goal lifetime for INTOR. It is prudent to examine this issue while we are still in the 'mission definition' phase.

Development of structural materials for fusion is a highly ambitious task. Aggressive goals have been established for such major service parameters as radiation lifetime and operating temperature but we lack a '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 to DEMO based on structural materials tests in INTOR as the goal fluences for such tests increase from 0 to 6 $MW \cdot a \cdot m^{-2}$. The 'benefit' is simply the reduction of uncertainties in predicting materials performance in DEMO that would result from higher test fluences (greater lifetime) in INTOR.

2.2. Evaluation methods

Thus, the primary emphasis in this evaluation was on the reduction in uncertainties (in the design data for DEMO) that would be provided as the test fluence available in INTOR is increased. Useful quantitative measures of these reductions in uncertainties were developed. Other factors that could influence the recommendation for the desired lifetime of INTOR (based on structural materials tests) include: a) the amount of information from complementary programmes, in particular from a high-energy, high-flux neutron source, b) the potential for deleterious changes in materials properties resulting from high-energy radiation damage but not apparent from fission data, c) the design impact of the data uncertainties. These subjects will be discussed in later sections following the description of the existing data and estimates of uncertainties.

Two materials, Ti-modified type 316 SS and an advanced alloy (vanadium), 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 INTOR. The selection of vanadium as an example of an advanced alloy does not represent an endorsement of the vanadium alloy as there are several other candidate materials. The two classes of 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. Nickel-containing alloys such as stainless steel can be irradiated in (mixed-spectrum) fission reactors and produce helium-to-dpa rates similar to those of a fusion reactor. 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. Because of the different states of development and the chemical and metallurgical characteristics of these two alloy classes, quite different answers may be expected to the question of risk vis-à-vis INTOR fluence experience.

Excellent characterizations of the reductions in the uncertainties in predictions of materials properties for a DEMO (10 $MW \cdot a \cdot m^{-2}$) as a function of increasing test fluence (lifetime) achievable in INTOR were obtained by using fluence-dependent models for selected materials properties. In terms of the reductions in uncertainties in the design data for a DEMO, the benefits from testing in INTOR are quite non-linear with fluence and show a strong breakpoint, (2–3) $MW \cdot a \cdot m^{-2}$, the value recommended for the INTOR test fluence goal. Testing to this fluence provides confirmation (for stainless steel) of the effects of radiation that occur fairly rapidly with fluence, e.g. changes in tensile properties. The lack of irradiation data on vanadium makes this conclusion somewhat more speculative in the case of advanced alloys.

Among the important caveats implicit in the approach taken here are:

(1) The fluence-dependent models provide a useful quantitative method for estimating uncertainties but the models themselves are not yet confirmed. The major effort in future materials R and D will be in validating such models rather than simply getting 'a few high-fluence data' points for predictions.

(2) The approach of using models based on fission data does not eliminate the potential effects of 'new unknowns' in moving to the fusion spectrum. Some effects (e.g. He in SS) are simulated in fission tests, but others may not be represented. There is no quantitative description of such 'wild card' effects in this evaluation; however, there is a later discussion of this subject.

(3) The data for vanadium alloys are insufficient to support conclusions about trends in properties at high fluence.

(4) Temperature is generally an important variable in radiation effects on materials. The specific temperature range in this study was 300–500°C for stainless steel. The conclusions for stainless steel, for example the predictions of adequate tensile ductility, would not be valid in a higher-temperature regime. For vanadium, the appropriate data were only available around 600°C.

(5) The assumption that the maximum test fluence for data obtained in INTOR converts directly to the INTOR lifetime desired implies a 'one shot' test programme where the potential choices of materials have been screened and the irradiation programme in INTOR is used to validate the primary selection or to choose between a very limited number of viable candidates.

The documentation of the major effort in this task, that of developing a method and then characterizing the uncertainties in the DEMO data base, is given in the national reports [3–6]. Only a brief summary of the work and a description of the general approach will be presented in this section.

2.3. Data base and estimates of uncertainty

Two examples of materials data projections are described below to demonstrate the method used to quantify uncertainties. The data are for the swelling and tensile strength of type 316 SS. As examples, 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. More complete data for stainless steel and data for vanadium are presented in Ref. [5].

2.3.1. Stainless-steel data

The swelling of cold-worked 316 SS has been well characterized experimentally by testing in fast-breeder reactor spectra. Many different chemistry variants and

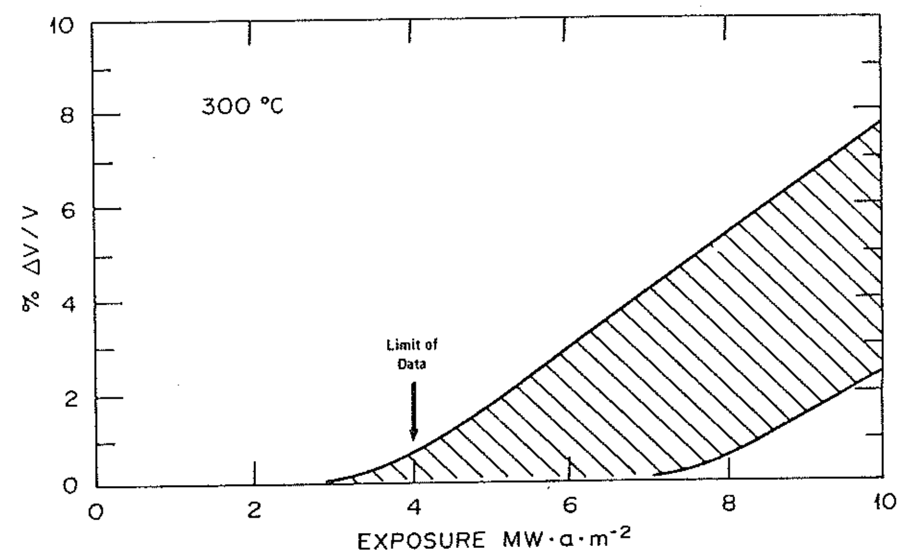

FIG.XII-1. Steel performance range at 300° C in fusion reactors.

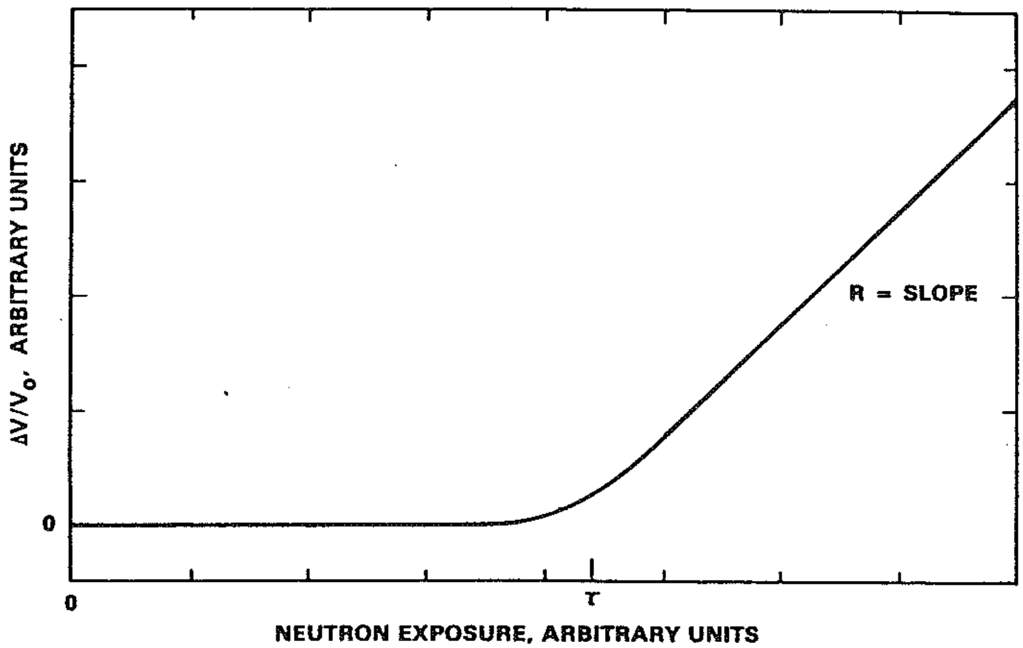

FIG.XII-2. Bi-linear swelling model.

fabrication processes are being explored in an attempt to increase duct and cladding lifetime which are limited by interference due to swelling-induced bowing, and swelling and irradiation creep dilations. The impact of swelling on designs for fusion reactors has not been sufficiently well studied to establish a firm guideline for maximum tolerable swelling; it is, however, clear that the tolerance is not limitless.

The high swelling of 316 SS at temperatures above 450°C led to the development of titanium-modified steel with a nickel content higher than 316 SS, e.g. alloys D9 in the breeder programme and PCA in the fusion programme. The expected performance range of these steels at 300°C in fusion reactors is shown in Fig.XII-1 (swelling is highly temperature-dependent; more data are presented in Ref. [5]). Because of the longer incubation time for swelling in the steels, definitive upper-limit swelling rate measurements require high-fluence data not yet available. Consequently, ranges of behaviour were estimated which provide for variation in incubation time and rates. It should be noted that the data presented in this section are bound by likely minima and maxima. The lower and upper limits are derived on the basis of expert judgement and, in many cases, are not based on actual data.

Figure XII-2 shows a bi-linear-swelling model. Its major parameters are the incubation time τ that marks the onset of swelling and the rate R at which swelling occurs after fluence τ. In projecting uncertainties in the amount of swelling that would occur after an exposure of 10 $MW \cdot a \cdot m^{-2}$ (DEMO lifetime), the swelling

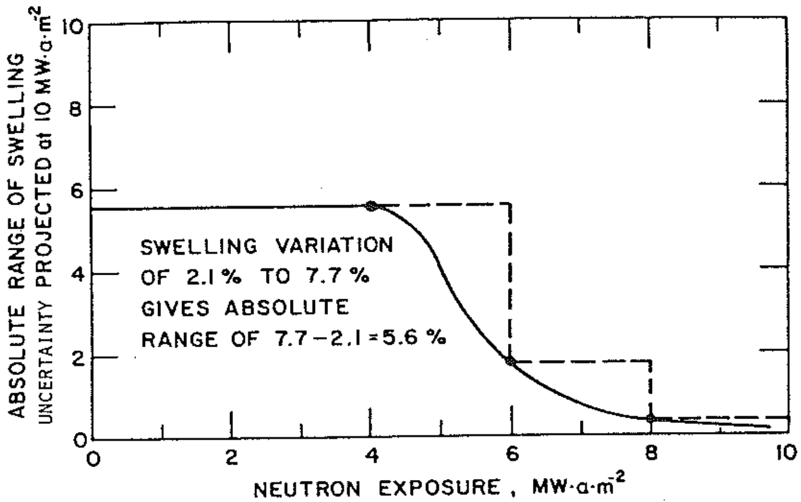

FIG.XII-3. Swelling uncertainty versus neutron exposure.

model, with τ and R varying within their allowable limits, was applied to data sets where the maximum assumed test fluence in INTOR was 0.2, 1, 2, 4, 6 and 8 $MW \cdot a \cdot m^{-2}$. The quantity of interest in these projections is the range in possible values for swelling at 10 $MW \cdot a \cdot m^{-2}$. In Fig. XII-1, this projected range is from about 2 to 8 percent swelling and the limit of our current data is a (fission) fluence equivalent to about 4 $MW \cdot a \cdot m^{-2}$.

As one would expect, when data become available at progressively higher fluences where swelling does occur, then two factors reduce the uncertainty in projections to even higher fluences based on these data. First, the data themselves begin to show the trend in swelling behaviour more clearly as increasing amounts of swelling are observed. Second, the interval between target fluence for performance projections (10 $MW \cdot a \cdot m^{-2}$ for DEMO) and the highest fluence in the data set becomes less. If data were continuously available (at all fluences), then the reduction in uncertainty for the amount of swelling projected at 10 $MW \cdot a \cdot m^{-2}$ as a function of the fluence of data actually available would begin at some relatively large value, before any (confirmatory) data were available, and decrease smoothly as these swelling data became available at progressively higher fluences as suggested by the solid line in Fig. XII-3. Note that the dependent variable in Fig. XII-3 is the range of uncertainty in the projected amount of swelling that would have occurred after a total exposure of 10 $MW \cdot a \cdot m^{-2}$, and this range is plotted as a function of the maximum fluence available in the test data used to estimate the projected value.

Figure XII-3 is a fairly simple conceptualization of the decrease in uncertainty that occurs as more data on swelling become available, and a somewhat more

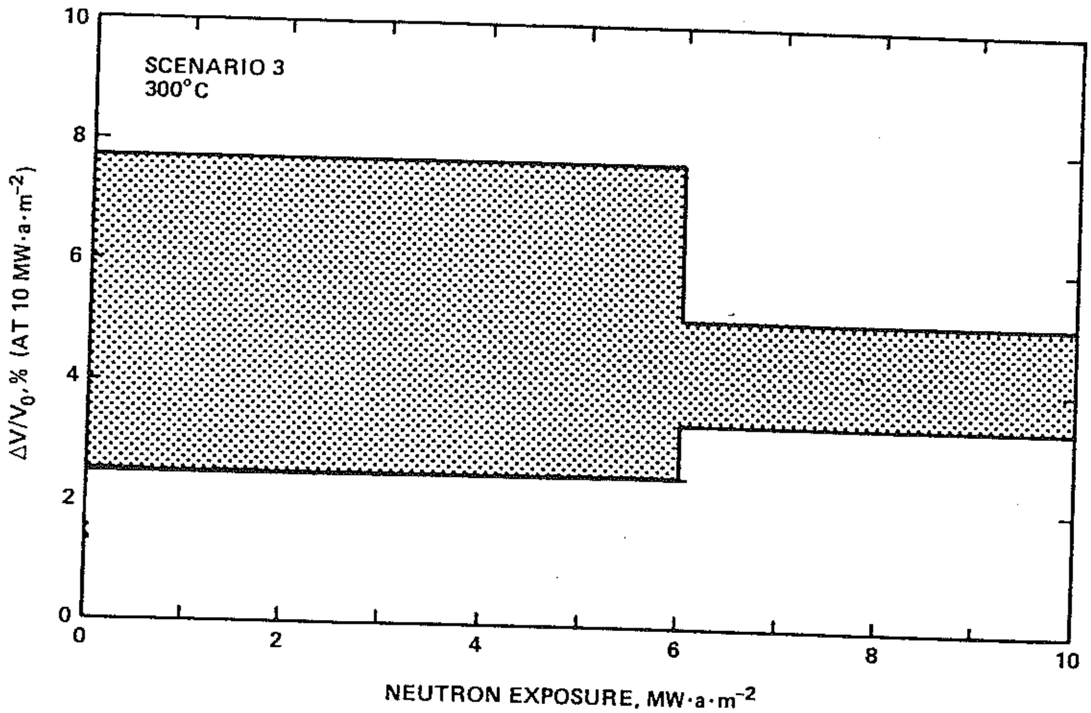

FIG.XII-4. Projected swelling uncertainty: medium swelling.

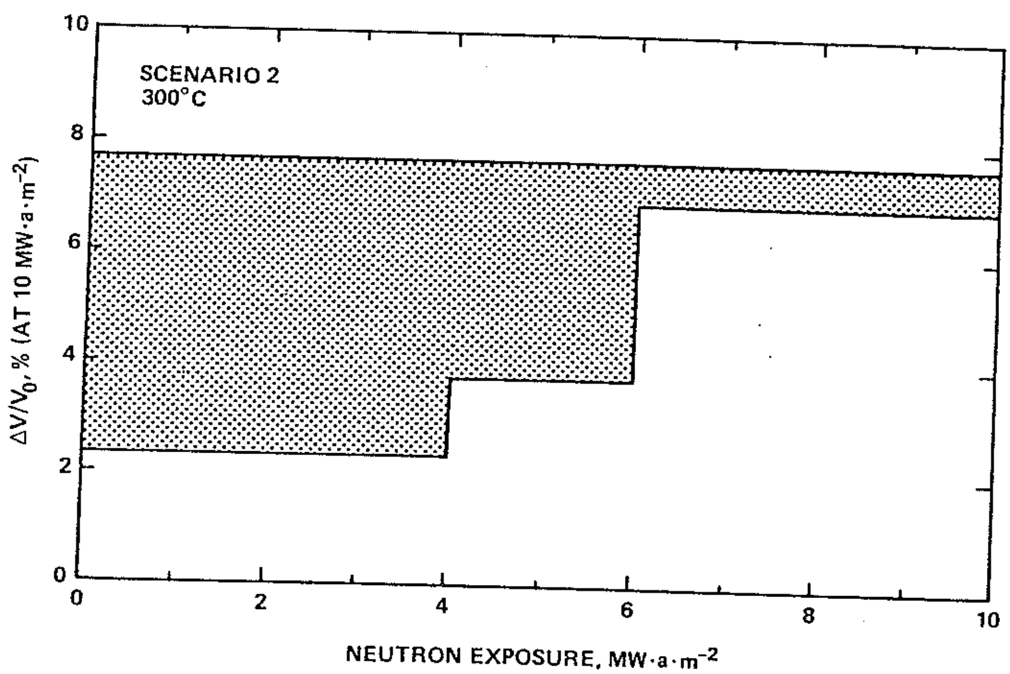

FIG.XII-5. Projected swelling uncertainty: high swelling.

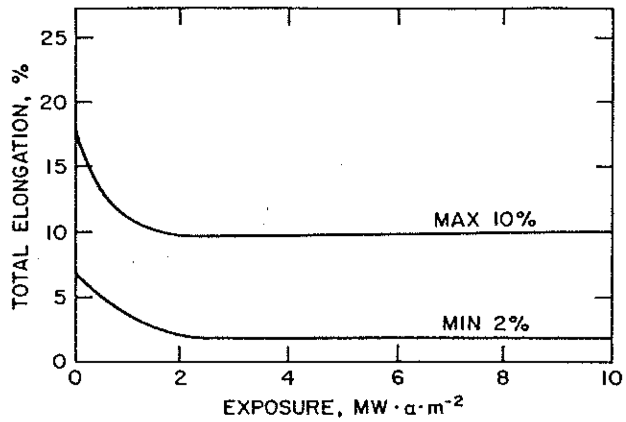

FIG.XII-6. Estimated range of values of total elongation for stainless steel.

sophisticated representation was adopted for this evaluation. High-fluence irradiation experiments are expensive, time-consuming, and typically produce data at a few widely separated fluences. Thus, the reductions in uncertainties will come in a series of steps as data become available, as shown by the dotted lines in Fig. XII-3 for discrete data at 4, 6 and 8 $MW \cdot a \cdot m^{-2}$. Relatively long times are needed to obtain high-fluence data, for example exposure to 6 $MW \cdot a \cdot m^{-2}$ would require about three years in EBR-II or HFIR, over ten years in ORR, and twelve years in INTOR (at a wall loading of 1 $MW \cdot a \cdot m^{-2}$ and an availability of 50%). Thus, the fluence axis in Fig. XII-3 also represents the passage of time in which the programme awaits data to reduce uncertainties.

Figure XII-4 shows a stepwise reduction in uncertainty in swelling but shows the actual position of the uncertainty range, e.g. initially 2.1% to 7.7%, rather than the absolute value of the range (5.6% in this case as depicted in Fig. XII-3). In Fig. XII-4 there is a sharp reduction in uncertainty when the data at a fluence of 6 $MW \cdot a \cdot m^{-2}$ are added to previous data with fluences up to 4 $MW \cdot a \cdot m^{-2}$. (There would be further reduction at 8 $MW \cdot a \cdot m^{-2}$; however, 6 $MW \cdot a \cdot m^{-2}$ was assumed to be the maximum possible exposure on INTOR.)

The analyses were actually performed for data sets where the (hypothetical) data followed low-, medium-, and high-swelling curves. Figure XII-4 is the projection for medium swelling (mid-range in Fig. XII-1). Figure XII-5 shows the projection for data representing high swelling. In the latter case, there is some reduction in uncertainty even at 4 $MW \cdot a \cdot m^{-2}$ because swelling has initiated at a lower fluence. (Such results would bode ill for the prospect of using that particular material in a fusion reactor.)

In contrast to swelling, post-irradiation 20% CM tensile properties of 316 SS have not shown great sensitivity to compositional variations. Irradiation hardening due to precipitate and Frank loop formation, etc. occurs at 370–480°C.

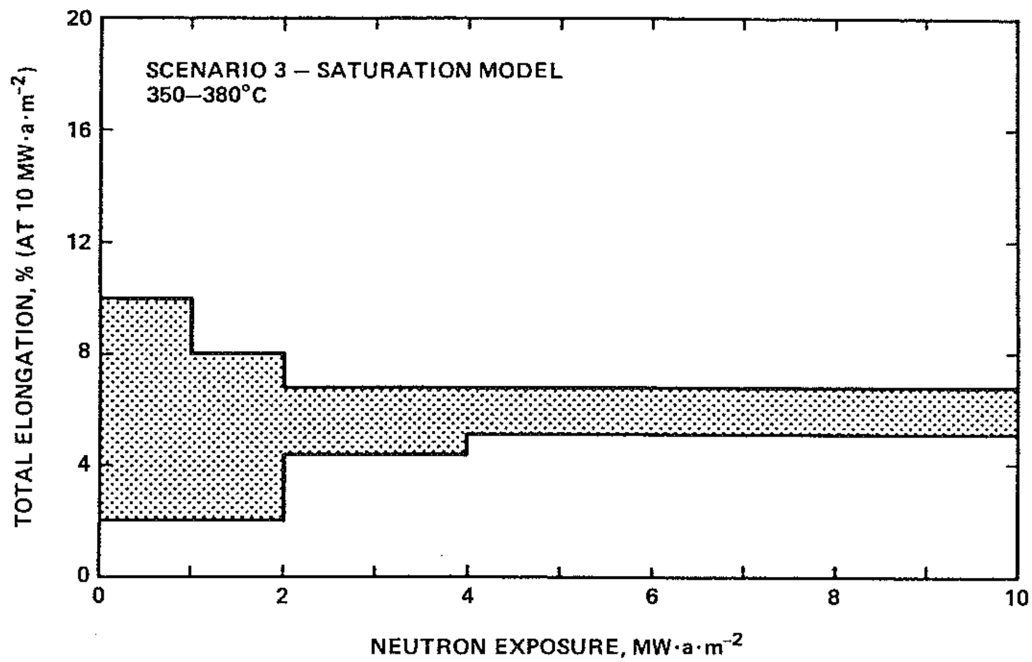

FIG.XII-7. Projected total elongation uncertainty.

At irradiation temperatures above 500°C, the yield strength decreases with irradiation. The data suggest that tensile strengths saturate with increasing fluence with the possible exception that at 370°C a slight trend for softening above 6×10^{22} $n \cdot cm^{-2}$ may be occurring.

The total elongation (ductility) generally decreases with fluence at all irradiation temperatures. The ductility appears to decrease more slowly with increasing fluence and may tend to a saturation value. In the temperature range of interest (below 500°C), fast-reactor data suggest that the total elongation will remain adequate (above 2%) for design purposes at high fluences.

Whereas very little helium is produced in fast-reactor irradiations, helium is produced copiously in 316 SS during HFIR irradiation; in fact, as much as 50 appm per dpa compared to 10–14 ppm obtained in a fusion 14 MeV spectrum. The HFIR test results will, therefore, allow identification of any potentially severe degradation of tensile properties due to helium embrittlement. Severe degradation is observed at 575°C and above but is not found in the temperature range of interest.

The total elongation after irradiation in HFIR at 350°C is representative of the lower temperature range and decreases rapidly at low fluence and appears to saturate at fluences above 30 dpa. Of crucial importance, the ductility saturation level after HFIR irradiation is 2–3%, the same as that following EBR-II irradiation at 371°C. Hence, helium has had no severe deleterious effect on ductility under these irradiation and test conditions.

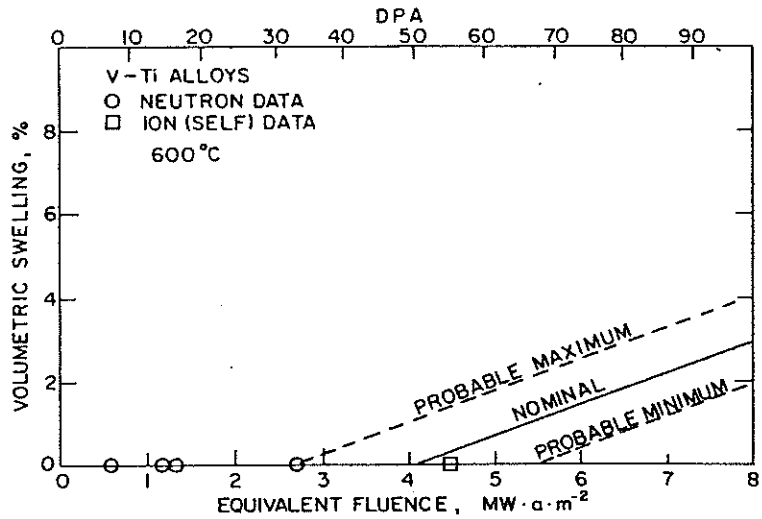

FIG.XII-8. Volumetric swelling versus equivalent fluence.

Figure XII-6 shows the estimated range of values of total elongation for stainless steel and Fig. XII-7 presents the projected reductions in uncertainty of the total elongation after an exposure of 10 $MW \cdot a \cdot m^{-2}$. The same general approach previously described for swelling (with, in this case, a model for tensile properties) was used to generate the projections of uncertainties.

The most important aspect of the results for tensile properties, represented by Fig. XII-7, is that significant reductions in uncertainties occur at the low to medium fluences achievable in INTOR. This result may be contrasted with Figs XII-4 and XII-5 for the swelling data, where significant reductions in uncertainties are not anticipated until much higher exposures.

2.3.2. Vanadium alloy data

As mentioned earlier, vanadium is selected here as an example of candidate materials for advanced alloys. Without exception, the relevant data base on neutron radiation effects in vanadium-base alloys is a remnant of the early evaluations of that class of alloys as candidate cladding for (fission) fast-breeder reactor applications and is limited to damage levels equivalent to less than 1.5 $MW \cdot a \cdot m^{-2}$ for tensile properties and about 2.7 $MW \cdot a \cdot m^{-2}$ for swelling. There are no neutron irradiation effects data for materials containing (n, α)-produced helium; the effects of irradiation on the mechanical properties under cyclic loads are also unavailable.

Figures XII-8 and XII-9 show the data for void swelling and uniform tensile elongation, respectively, as a function of neutron fluence represented here as equivalent test exposure (fluence) in INTOR. From the upper limits of the data,

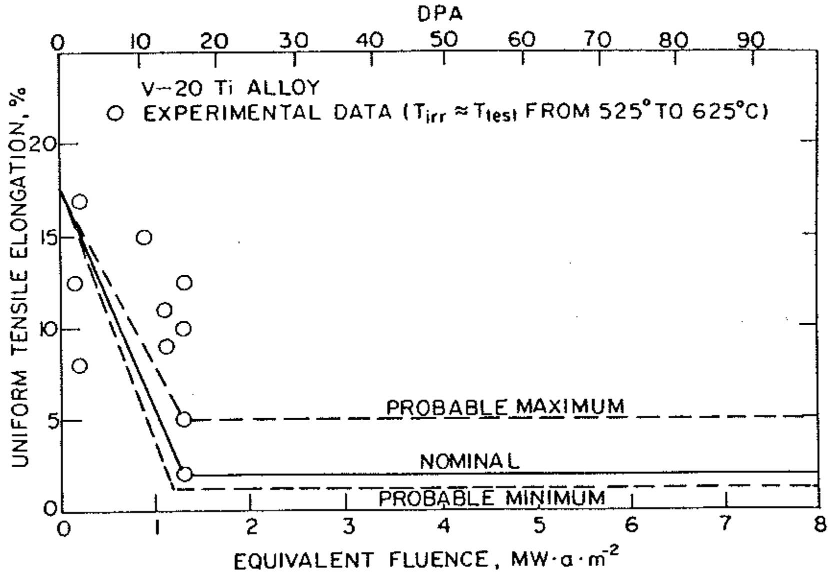

FIG.XII-9. Uniform tensile elongation versus equivalent fluence.

generally equivalent at only a few MW·a·m^{-2}, projected values are indicated for nominal, probable minimum and probable maximum behaviour. (Reference [5] gives a more complete explanation of the data and their interpretation.)

Since the data available on neutron radiation effects in vanadium alloys are extremely limited, a judgement 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 INTOR are still supportable and appropriate. For a DEMO, which is currently perceived as a reactor having an equivalent lifetime first-wall fluence 10 MW·a·m^{-2}, a minimum experience level of 6 MW·a·m^{-2} should be available if materials selections are to be made with an acceptable degree of confidence.

Data on radiation effects and the anticipated changes in materials properties from exposure to fusion neutron irradiation are summarized in Table XII-1. The judgements on 'Benefits of Testing' are based on available data from fission reactor irradiations.

2.4. Strategy for fusion materials development

The INTOR study has shown that, with respect to radiation damage, the existing data on the different candidate materials are not sufficient for a reliable construction of future fusion devices. The discussion of the Test Programme was based, as far as radiation damage is concerned, on the development of new

TABLE XII-1. CHANGES IN MATERIAL WITH FLUENCE

Maximum test fluence	Benefits of testing
Stainless steel	
0–1 $MW\cdot a\cdot m^{-2}$	Little useful information
1–3 $MW\cdot a\cdot m^{-2}$	Confirmation of low-fluence effects predicted with other sources (e.g. on tensile properties)
3–6 $MW\cdot a\cdot m^{-2}$	Model verification from observations of microstructure preceding long-term changes in behaviour
Above 6 $MW\cdot a\cdot m^{-2}$	Confirmation of performance near end of life (e.g. high swelling)
Vanadium alloys	
Up to 1.5 $MW\cdot a\cdot m^{-2}$ [a]	Tensile properties still change
Up to 3 $MW\cdot a\cdot m^{-2}$ [a]	No swelling observed

[a] Tests in range of 500–600°C.

materials which can be tested in INTOR for use in the subsequent devices. To satisfy, in an acceptable manner, the needs of construction of INTOR and the subsequent devices, it is necessary to prepare a strategy for materials development. To establish the strategy it is necessary to consider the present state of knowledge, the existing and required facilities, the materials data base required, the different steps in the development of fusion power, and the mission and schedule of each of these steps.

2.4.1. Materials development

For the following, it is essential to define the meaning of 'materials development for fusion reactors'. Materials engineering for any application involves (1) materials selection, (2) materials development, and, finally, (3) a compilation for the designer of materials properties as functions of all appropriate variables (e.g. temperature, stress, chemical environment, radiation, etc.)

Materials selection

Initial materials selection for technical applications is made by utilizing approximate data for the materials strength, basic chemical properties, basic physical properties, etc. For an application in a fusion reactor, examples of the important properties for a selection may be: a) reasonable high-temperature strength, which would permit operation at high temperatures; b) good chemical compatibility with breeder materials; c) high radiation damage resistance; d) high thermal conductivity; and e) low-activation or fast-decaying activation products.

The initial selection process can, in most cases, be performed on the basis of the existing data. Of course, the quantity of the data for different alloys may differ substantially. For the scope of the present study, the data on AISI 316 steel are by far the most extensive. On the basis of these data, one can also establish the upper limits on the performance of stainless steel.

Materials development

After a selection of the most promising alloys, a phase of extensive testing is necessary. During this phase, slightly different compositions of the selected alloy and different fabrications and thermomechanical treatments are tested and compared in order to improve the properties and to optimize the alloy with respect to its application. Although the number of alloys which enter this phase is rather limited, the work necessary to accomplish this phase is very large. In the case of materials for which radiation damage resistance is an important feature, it is necessary to irradiate several times during successive steps of optimization to the fluence desired. This can be a rather time-consuming step, if the necessary fluences are high and the existing sources deliver low fluxes.

Engineering data base

Generating the engineering data base involves working with commercial producers to produce the desired compositions and microstructures, to produce multiple heats and to demonstrate reliable production technology. It also involves determination of properties in unirradiated as well as irradiated conditions over the whole range of the required parameters. Producing these data on a sufficient number of specimens is required to reach the necessary confidence level. This phase requires an even larger effort than the development phase. It is assumed that the minimum time for a fusion reactor material to accomplish all three phases is 15 years, provided a high-flux neutron source is available. However, the experience with zircaloy in the light-water reactor development and with stainless steel for fast breeders shows that considerably longer times, of the order of 30 years, may be more realistic.

TABLE XII-2. MAIN REACTOR PARAMETERS

	INTOR	DEMO	Commercial
Thermal power (MW)	620	1000	4000
Gross electrical power (MW)	10	100	1000
Neutron wall loading ($MW \cdot m^{-2}$)	1.3	2–3	3–6
Surface wall loading ($MW \cdot m^{-2}$)	0.1	0.2–0.8	0.3–1.5
Burn time (s)	200	>1000	>1000
Number of cycles	7×10^5	10^5	10^5
Integrated wall load ($MW \cdot a \cdot m^{-2}$)	6.6	10–20	15–30

2.4.2. Boundary conditions

The materials development has to provide the data base necessary for the construction of INTOR, DEMO and the commercial reactor. It has therefore to consider the objectives and working conditions of these devices.

In Table XII-2 an attempt is made to compare some of the features of the three reactors. Undoubtedly, the development of a new material for use in INTOR, or any other machine of this generation, is impossible. Therefore, INTOR has to be built with rather well-known material, presumably stainless steel of the type AISI 316, for which enough data already exist.

The use of a newly developed material can only be planned for devices after INTOR. Whether the new material can be developed for use in the DEMO or commercial reactors, depends strongly on the time schedule and availability of resources and facilities.

The most time-consuming and most important problem in fusion material development is the behaviour of new materials under irradiation. Even for the preparation of a comprehensive data base for stainless steel, a significant radiation programme in existing facilities is necessary. Fission reactors with spectrum tailoring can be a fairly good means for radiation damage studies in nickel-containing alloys. However, for other alloys, this advantage does not exist and, therefore, radiation tests in a neutron spectrum similar to a fusion reactor with a sufficiently high flux become necessary. Machines like INTOR are not able to allow a re-iteration of radiation tests as are necessary for the development of new materials. They can be only used for a once-through radiation at sufficiently high fluence. It is important to have other high-energy high-flux neutron sources in order to run a reasonable development programme.

TABLE XII-3. SCENARIO FOR USE AND DEVELOPMENT OF STRUCTURAL MATERIALS

	INTOR	DEMO	First-generation commercial reactor
Scenario 1	Stainless steel	Stainless steel	Stainless steel
Scenario 2	Stainless steel	Stainless steel	New alloy
Scenario 3	Stainless steel	New alloy	New alloy

2.4.3. Development scenarios

Different scenarios for the use and development of fusion reactor materials may be envisaged as is shown in Table XII-3. In the following, a number of scenarios are outlined together with their advantages and drawbacks.

Influence of existing knowledge

A large data base on radiation damage in stainless steel in the temperature range between 400–700°C and for low He/dpa ratios exists. The development of improved stainless steels for fast breeders is under way. The date base can be extended to lower temperatures and, using tailored neutron spectra in fission reactors, it can be extended to the right He/dpa ratio. The experiments in fission reactors can probably be done up to fluences similar to those which are expected in DEMO.

Simulation experiments in accelerators for heavy and light ions can give qualitative information on radiation damage at high damage rates, but with a damage structure which might be rather different. However, a number of features can be clarified very well by accelerator experiments.

All fusion machines will also contain materials without nickel (e.g. in limiter/divertor) for which a simulation in fission reactors is not as helpful as for stainless steel. For these materials, radiation experiments in accelerators will be important. Intense neutron sources will be a further powerful means for radiation damage studies.

All simulation experiments have to be accompanied by a strong theoretical effort on damage modelling, which has to aim at the comparison of the different damage mechanisms.

A. Scenario 1

In Scenario 1, stainless steel is used in INTOR, DEMO and commercial reactors. This scenario is the least expensive and most conservative one. It does not ask for the development of new materials at any time of the development of a fusion power reactor. It allows, therefore, a rapid succession of the different steps. It relies completely on the existing materials and the experience gained in fission reactors. It does not, however, take into account the fact that the conditions in a fusion reactor are new and different from fission reactors. The existing stainless steels have been optimized for fast breeders. The goals in this optimization are different from those for fusion reactors. Therefore, there is a risk that they will not meet the requirements of fusion reactors and will lead to early failures and much less economically competitive fusion power production.

INTOR

Construction. The data necessary for the construction have to be produced in fission reactors, accelerators and intense neutron sources. The data base has to be developed not only for stainless steel, but also for other materials which will be used in the divertor, limiter, heating system, etc.

Testing. INTOR can produce radiation damage data for DEMO and commercial reactors (CR) on specimens and structural parts at fluence levels and damage rates below those of DEMO and CR. Simulation experiments for higher fluences will be necessary. The INTOR data will be very helpful for the calibration of simulation experiments and for deciding which simulation technique is the best. Not all simulation sources will be able to give fluences comparable to DEMO and CR, and, therefore, the best simulation method cannot necessarily be applied for high fluences.

Fluence. Several fluence levels in INTOR may be considered as its possible goals: It should surely arrive at a fluence which allows a reliable calibration to be made. However, higher fluences are desirable in order to allow a reasonable extrapolation of the radiation damage to DEMO and to reduce the risk for early failure in DEMO. A key problem is that a higher fluence goal for INTOR increases the risk to INTOR itself.

DEMO

Construction. DEMO will be built on the basis of radiation damage data obtained by simulation and in INTOR. Since DEMO should show the feasibility of fusion power, INTOR radiation should be pushed as far as possible to minimize the risk in DEMO.

Testing. DEMO will be a testbed for CR materials (stainless steel in Scenario 1) in the form of specimens, modules and full elements at nearly full power. It could also be used for tests on materials which are under development for later generations of commercial reactors.

Fluence. DEMO will probably arrive at the limit of the lifetime of stainless steel in a fusion reactor. Therefore, the tests on CR elements in DEMO will be full-fluence tests for stainless steel. Tests at higher fluences in this scenario are not necessary since it includes exchanges of the first wall in the commercial reactor.

COMMERCIAL REACTORS

Construction. The data furnished by DEMO will be sufficient if DEMO arrives at the lifetime limit of stainless steel. It must, however, be realized that in this case the first wall of a CR has to be replaced two to three times during the lifetime of the reactor.

Testing. Irradiation tests should be executed on new materials, to the extent that they are compatible with economic requirements.

General comments

(1) Stainless steel as a first wall material in the CR probably includes frequent exchanges. The reactor economics will be significantly affected by the frequency of replacing the first wall.
(2) The development of new materials can be distributed over a long period and will only slightly increase the budget for fusion technology.
(3) The development of fusion power will not depend on development of structural materials.

B. Scenario 2

In this scenario, stainless steel is used in INTOR and the DEMO but a new material is used in CR. This scenario seems to be the most reasonable one at present. It is based on the possibility of developing a material which has a higher end-of-life fluence than current stainless steels and which would meet other conditions as defined above. It also takes into account the fact that it would be very difficult to develop and test new materials in time for INTOR or DEMO. The new material could be improved stainless steel, a ferritic, vanadium, or titanium alloy, etc.

INTOR

Construction. The same consideration as for Scenario 1 is valid.

Testing. The task of INTOR will be largely as in Scenario 1; it might, however, include some screening tests for the new materials under development for CR.

Fluence. The considerations in Scenario 1 have to be extended in order to provide testing space and fluence for the new material.

DEMO

Construction. DEMO will be built on the data base developed together in INTOR and a simulation programme as in Scenario 1.

Testing. It will be the main testbed for the CR. An extensive test programme on specimens, modules and whole segments has to be executed on the new CR material.

Fluence. In this scenario, DEMO will play a double role: it has to demonstrate the possibility of electricity generation by a fusion reactor and to serve as a materials test reactor for the CR. These two tasks demand a rather long life and high fluence and will probably require the exchange of stainless-steel components exposed to peak fluence. The fluence in tests on the CR materials should arrive at least at one half of the CR lifetime fluence.

COMMERCIAL REACTORS

Construction. The construction will be based entirely on DEMO radiation tests and some parallel simulation tests. It is, however, questionable whether all simulation devices can arrive at fluences as high as necessary for a CR. The newly developed material could be an advanced stainless steel.

Testing. If compatible with the economic conditions, further radiation testing on CR materials would be helpful.

General comments

(1) The development of a new material with a higher lifetime fluence should reduce the frequency of first-wall exchanges in a fusion reactor. But the necessity of high-fluence tests in DEMO increases the constraints to which it is subjected.

(2) The materials development programme is compressed in a much shorter period of time than in Scenario 1. It will, therefore, require immediate strong action and a substantial budget.
(3) The influence of materials development on the development of fusion power is still limited since DEMO will not require a new material.

C. Scenario 3

In this scenario, a new material is developed for use in DEMO and CR. This scenario is very ambitious; it relies on INTOR for the development of a data base and for testing of a new material.

INTOR

Construction. As in the other scenarios, Scenario 3 requires the preparation of a data base with the existing simulation techniques.

Testing. INTOR will be the main testbed for the DEMO material. It should allow radiation damage tests on specimens, modules and sectors, in parallel with simulation experiments.

Fluence. The goal fluence should be as high as possible since, most probably, spectrum tailoring in fission reactors for the right He/dpa ratio of the new alloy will not be possible, and the damage in the INTOR fluence will have to be extrapolated to DEMO conditions.

DEMO

Construction. All data necessary have to be newly produced by INTOR and adequate simulation. Only an extensive programme will allow extrapolation with some degree of reliability. The construction will have to wait for these data.

Testing. Most data necessary for the CR will have to be generated in DEMO. Simulation will probably not be possible for the extreme high fluences. But, since DEMO and CR are of the same material, the operation experience of DEMO itself will be very helpful for the CR.

Fluence. Since DEMO will be almost the only source for the data base for the CR, a high fluence is necessary.

COMMERCIAL REACTORS

Construction. As in Scenario 2, but with the additional data coming from the operation of DEMO.

Testing. The condition of having an economic reactor limits the possibility for testing.

General comments

(1) An immediate strong effort for materials development is necessary.
(2) The data base for the construction of the CR will benefit from the operation experience in DEMO. Therefore, the risk in the CR is reduced with respect to Scenario 2.
(3) A high-flux high-energy neutron source is necessary.
(4) The pace of the fusion power development depends strongly on the materials development.

2.4.4. *Choice of scenario*

All three scenarios discussed use stainless steel for the construction of INTOR. It is, therefore, the first task in any strategy which develops materials for fusion reactors to prepare the data base for INTOR.

In all scenarios it is assumed that the current stainless steel may not be the ultimate material for a mature fusion power economy. The scenarios differ in the time allocated for the development of a new material which will substitute for stainless steel. The speed of this development depends on the effort one is ready to spend on this problem. It is rather difficult to imagine performing this development in parallel with the construction of INTOR, as is suggested by Scenario 3. On the other hand, it is very risky to provide for the introduction of the new material only in the second generation of power reactors (Scenario 1). Since existing steels have not been optimized for a fusion reactor environment they may not meet the lifetime goals of a CR.

2.4.5. *Strategy*

The first machine to be constructed will be an INTOR-like device. Our current knowledge and engineering capabilities have strongly affected the choice of stainless steel as first-wall and main structural material in this machine. Unfortunately, there exists a great variety of stainless steels with differences in the main properties. The development of a complete data base for the construction of the next machine will require several years (six to seven) since the simulation of the radiation damage has to rely on existing fission reactors and, therefore, will be limited by the fluxes available. For this reason, an early choice of one well-defined stainless steel is mandatory. This involves the risk that better materials possibly showing up in further developments will not be considered. Within the time schedule, however, this risk has to be accepted. The backbone of the radiation damage studies will be the irradiation in fission reactors with a tailored spectrum and, if they are available, in machines like FMIT in order to achieve the right He/dpa ratio. This work will have to be completed by light- and heavy-ion irradiation. In some cases, the simulation in these devices is as good as the irradiation in a tailored neutron spectrum. These activities have to be accompanied by theoretical studies, aimed at the comparison of the damage mechanisms in the different sources and at the development of a unified damage model. Conceptual design studies of fusion reactors have pointed to another irradiation damage problem, which has frequently been ignored. Besides the large structures, a fusion reactor will contain other elements which experience the same neutron irradiation as the first wall. These are: divertors, limiters, heating devices and diagnostic devices. They normally are built from materials that have not been investigated sufficiently as to their radiation damage behaviour. An activity has to be started on these materials. Since they mostly do not contain nickel, the spectrum tailoring is not effective for the simulation of the He-production. They are mainly parts for which an exchange is scheduled; therefore, the fluence requirements are lower and the decision on the materials may be delayed with respect to the stainless steel if sufficient radiation space is available.

Also in Scenarios 1 and 2, the structural material for DEMO is stainless steel. To utilize the full experience gained from INTOR, it will be necessary to build DEMO with exactly the same material. In this case, one will be able to organize the necessary long-term radiation experiments in such a way that the experiments for INTOR and DEMO form 'one block'. The overall number of experiments can be minimized and the results on the behaviour of the material in INTOR can be used to verify the value of the simulation techniques at any fluence. Such a verification at many fluences would increase the reliability of the extrapolation to the DEMO fluences. However, one would have to use a stainless steel optimized for fast breeders, and for properties which are not so important for a fusion reactor. Therefore, there exists the risk of building DEMO with a material not optimized for the fusion environment. At this moment, it is still difficult to completely evaluate the risk arising from this situation.

2.5. Conclusions

The projections of uncertainties in data on structural materials have provided a useful semi-quantitative method of establishing criteria for goal lifetime(s) for INTOR, based on its role in providing test data on structural materials. Separate criteria are necessary for stainless steel and for vanadium (or other advanced

TABLE XII-4. FLUENCE CRITERIA FOR INTOR LIFETIME

	With FMIT	Without FMIT
Stainless steel	$3\ MW \cdot a \cdot m^{-2}$ Low risk 0 Low-medium[a] risk	$3\ MW \cdot a \cdot m^{-2}$ Low-medium risk
Advanced alloys	? ? ($3-5\ MW \cdot a \cdot m^{-2}$) ? ? Medium risk	$3 \rightarrow 6\ MW \cdot a \cdot m^{-2}$ High → medium risk

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

alloys not containing nickel) as summarized in Table XII-4. Since specimens in a test module will receive less exposure than the first wall of INTOR, test fluences must be increased by a factor of about 1.2 to 1.5 to give equivalent lifetime criteria for INTOR. There was no need to do this rigorously in the current analysis. The range of $2-3\ MW \cdot a \cdot m^{-2}$ was simply rounded up to a criterion of $3\ MW \cdot a \cdot m^{-2}$. (However, the same factor applied to swelling data in the range of $6-7\ MW \cdot a \cdot m^{-2}$ would give lifetime requirements of $8-10\ MW \cdot a \cdot m^{-2}$.)

Judgements of 'risk' are also included in the table as well as several conclusions to be explained subsequently.

2.5.1. Fluence criteria and risk for stainless steel

The general consensus among the materials experts contributing to this task was that the existing data base, coupled with a future vigorous research programme in fission test reactors, could provide sufficient knowledge to proceed with the construction of a DEMO with only moderate risk. The recommended 'low-risk' test scenario would involve a complementary test programme using fission reactors and FMIT in conjunction with tests in INTOR on mechanical properties in the range of $2-3\ MW \cdot a \cdot m^{-2}$. 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 INTOR ($0\ MW \cdot a \cdot m^{-2}$) was given some credence. However, the lack of a large test volume in INTOR to provide engineering data with a 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 $MW \cdot a \cdot m^{-2}$ 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 behaviour of stainless steel at high fluences in a fusion environment.

Consider, for example, the implication of high swelling that might occur in a DEMO first wall. For this example, the upper bounds of swelling at 500°C (as given in Ref. [5]), will be used, i.e. a swelling rate of 4% per $MW \cdot a \cdot m^{-2}$ of first-wall exposure will occur after the onset of swelling at 5.5 $MW \cdot a \cdot m^{-2}$. Let us further assume an optimistic limit for tolerable swelling of 10% for which the allowable increment of lifetime after the onset of swelling is 2.5 $MW \cdot a \cdot m^{-2}$. If the onset of swelling occurred in DEMO at 5.5 $MW \cdot a \cdot m^{-2}$ first-wall exposure and DEMO were operated until a design lifetime of 8 $MW \cdot a \cdot m^{-2}$ when permanent shutdown was required because of observed swelling of first-wall components, its operation to 80% of planned life would undoubtedly be considered a success. However, if the operation of the DEMO indicated that economic break-even required a first-wall life greater than 8 $MW \cdot a \cdot m^{-2}$, there would be no data base to produce a design based on a 'better material' and no facility available with which to develop such data. This predicament is the problem with a testing strategy that does not provide for high-fluence testing of materials in a fusion-like environment.

2.5.2. Fluence criteria and risk for an advanced alloy (vanadium)

Vanadium, or more properly vanadium alloys, have been used in the evaluation to represent the general class of (non-nickel-bearing) materials for which there are neither many data available at present nor much optimism that useful simulations of the fusion (radiation) environment can be performed by using existing irradiation facilities.

The challenge that would face the fusion community, were an advanced alloy to be necessary for fusion to be viable, is to accomplish the following baseline activities as a minimum strategy toward developing an advanced alloy:

(1) Develop sufficient out-of-reactor data to characterize mechanical properties, physical data, corrosion, etc. so that crucial, meaningful comparisons of candidate advanced alloys could be performed.

(2) Develop sufficient information on the radiation effects in advanced alloys to identify critical problems. This step would require testing in a fusion-like irradiation environment.

(3) Develop an in-depth characterization of the critical design properties of the most promising advanced alloys.

TABLE XII-5. OPERATING CONDITIONS ASSUMED FOR INTOR AND DEMO

Category	Units	Proposed values	
		INTOR	DEMO
Neutron wall load	$MW \cdot m^{-2}$	1.3	2.0–3.0
Integrated neutron wall load	$MW \cdot a \cdot m^{-2}$	> 5.0	10.0–15.0
Availability	%	20–35	50–70

(4) Develop sufficient data on irradiated material to demonstrate that critical problems identified previously had been solved.

The conclusions in Table XII-4 on the use of INTOR to test advanced alloys must be evaluated in the light of a strategy such as the one outlined above. In a testing scenario with FMIT available, 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 INTOR testing, and that a programme consisting of benchmark tests on alloy candidates in FMIT with subsequent extensive tests in INTOR could develop a suitable advanced alloy for DEMO. A problematic aspect of this testing scenario is whether the data from FMIT (or some other fusion-like source) that would define the useful upper limit for the goal exposure in INTOR tests might be available at the time when decisions concerning the design lifetime of INTOR were being made.

In the second case for advanced alloys, where FMIT is not available, the basic rationale regarding INTOR testing is that greater lifetime and higher test fluences will increase the probability for an advanced alloy selected for DEMO meeting its performance goals. The predicament implied by this development strategy is similar to, but more acute than, that described previously for stainless steel. The logically separate steps of first identifying problems associated with radiation effects and then developing solutions through a two- or three-phase programme cannot exist in the scenario without FMIT. The alternative is simply to test many alloys extensively in the hope that one or more candidate materials will prove satisfactory. This approach clearly involves some risk (even when the test fluence approaches the DEMO lifetime).

TABLE XII-6. CATEGORIES OF INTOR BLANKET TEST SPECIMENS

Fidelity (Faithfulness to DEMO)	Size
Duplication	Full module
(Exact correspondence of all design and fabrication details to those of DEMO)	(Exact size of a complete DEMO module)
Simulation	Partial module
(Changes to some design details or operating conditions compared to those of DEMO)	(A small portion of a DEMO-sized module; includes most or all of the full module of thickness)
	Element
	(A very small part of a module, consisting of one coolant tube and the surrounding breeder)

3. BLANKET TESTING

3.1. Introduction

The objective of the blanket tests in INTOR is to obtain basic data for the design of the DEMO blanket. This blanket is required to have a high performance which meets the requirements of economy and safety under more severe neutronic and thermal conditions than those of INTOR. The ranges of fluence and neutron wall load for INTOR and DEMO are listed in Table XII-5.

The blanket testing strategy in INTOR was evaluated as five major requirements:

(1) adequate tritium production
(2) acceptable tritium inventory
(3) compatibility of materials
(4) efficient heat recovery
(5) acceptable lifetime

The blanket can be tested in the form of modules or complete sectors. Testing in modules is planned for the second and third phases of INTOR operations, with the purpose of proving the blanket design performance, tritium extraction methods and thermal and hydraulic characteristics.

Near-full-scale testing of prototypical DEMO-size units can be carried out during the third phase of INTOR operation in test sectors which represent one twelfth of the entire blanket structure. The choice of the blanket type for the full-scale testing in sectors will be done on the basis of the module tests.

The blanket designs will vary according to the type of breeder material and coolant. Testing of both solid (Li_2O, Li_2SiO_3, $LiAlO_2$) and liquid (Li, $Li_{17}Pb_{83}$) breeders is assumed. Up to the present time a large number of blanket designs has been suggested for the DEMO reactor, but the final choice will have to be made later. Some examples of electricity-producing blankets are described in the Appendix of this chapter.

The designs of modules should be relevant to the DEMO reactor and for this reason should include the first wall.

The types of blanket test specimens in the second stage of INTOR operation were classed as in Table XII-6 on the basis of (1) size, and (2) faithfulness (fidelity) to the DEMO blanket.

The test specimens to be used in the INTOR blanket tests are mainly of two types. The first is a model that corresponds exactly to the details of the DEMO unit and the second is a simulated model that includes changes in some design details or operating conditions compared to those of DEMO.

3.2. Evaluation methods

3.2.1. Selection of tests for evaluation

Five major types of test were considered; neutronic, tritium recovery, materials compatibility, heat recovery, and lifetime tests.

One of the major concerns in the INTOR blanket tests is how data relating to the major items of performance described above should be obtained for the DEMO blanket design. Along with testing in INTOR, results have to be obtained in complementary facilities, e.g. 10 $MW \cdot a \cdot m^{-2}$ tests in fission reactors and other engineering mock-up tests. Basic materials information from the irradiation tests in complementary facilities for the fluence equivalent of 10 $MW \cdot a \cdot m^{-2}$ would be available at the time of the start of DEMO construction.

The blanket module tests in INTOR will provide the data relating to the working conditions, the lifetime of the major components and the operating capability of the module as a whole. The benefit from the module testing in INTOR obviously depends on the neutron fluence achieved. Table XII-7 summarizes the estimation of the value of the data provided by the testing of blanket modules in INTOR and complementary facilities (fusion reactors, ion accelerators, 14 MeV sources, etc.). Estimated results from INTOR testing are included for four fluences. It was assumed that, for this estimation, the first-wall

TABLE XII-7. THE VALUE OF THE INFORMATION OBTAINED FOR DEMO FROM BLANKET TESTING IN INTOR AND IN COMPLEMENTARY FACILITIES[a]

Category of tests	INTOR tests at a fluence of ($MW \cdot a \cdot m^{-2}$)				Complementary facilities	
	0.2	2.0	6.0	10.0	Fission reactors	Special facilities[b]
1. Neutron flux distribution	H	H	H	H	L	H
2. Temperature distribution within module elements	H	H	H	H	L	H
3. Tritium breeding	H	H	H	H	H	M
4. Tritium recovery	M	M	H	H	H	L
5. Equilibrium tritium inventory	M	M	H	H	M	L
6. Tritium-containing elements lifetime	M	M	M	M	H	L
7. Surface thermal flux distribution	H	H	H	H	L	L
8. First-wall lifetime	L	M	M	M	H	L
9. Heat removal and energy conversion system operating mode	M	M	H	H	M	L
10. The entire module lifetime	L	M	H	H	M	L

[a] L means low, M medium, H high.
[b] Including ion accelerators and 14 MeV neutron sources.

loading is kept constant. The rating has been done for categories of information required for the DEMO design.

The neutron flux parameters within the module and the temperature distribution hardly depend on the neutron fluence, so the benefit from the evaluation of these characteristics is high for any INTOR fluence since it is practically impossible to qualify them by means of complementary facilities.

The data on tritium breeding and detector lifetime needed for DEMO can be obtained, to a considerable degree, from complementary facilities (fission reactors); so, the benefit of such tests in INTOR is relatively low, though they are essential from the viewpoint of the evaluation of the entire module working capability and the control of the operating regime.

The first-wall lifetime will be evaluated to a considerable degree on the basis of material sample tests in INTOR and complementary facilities though the testing of an entire first wall as one of the module components at fluences of 2.0 $MW \cdot a \cdot m^{-2}$, and higher, yields useful additional information.

A substantial part of the information about modes of operation and parameters of the module zone cooling and the energy conversion system can also be obtained from complementary facilities. The benefit of such tests in INTOR is high and increases with higher design fluences.

From Table XII-7, it is concluded that valuable information would be obtained by making the blanket module tests in INTOR at fluences larger than 2 $MW \cdot a \cdot m^{-2}$.

The probability of obtaining the required data for DEMO from tests in complementary facilities was also considered.

The testing of an entire blanket module in the existing and the complementary facilities foreseen is practically impossible because of the large dimensions of the module. Some substantial information can be obtained from the testing of separate blanket components (see Table XII-7).

From the table it can be seen that the complementary facilities could provide a good part of the information needed for DEMO.

The results [5] obtained from the evaluation of specific types of blanket tests in the five general categories are discussed in the next section. However, a prior discussion of some of the more general results of the study is in order since they have a strong bearing on the results from the evaluation of the specific tests.

Correlation between fission and fusion tests

Radiation damage in breeders containing solid lithium compounds is believed to occur principally as the result of the lower-energy neutron irradiation, and it was assessed that blanket specimen tests in fission reactors should yield essentially the same results as tests in a fusion (INTOR) reactor. Tests of blanket specimens up to 10 $MW \cdot a \cdot m^{-2}$ could be adequate for the support of the DEMO blanket design if confirmed by tests in INTOR to some lower fluence. Since the blanket structural materials are not expected to incur significant radiation damage during the 10 $MW \cdot a \cdot m^{-2}$ DEMO life, any changes in blanket performance should be the result of breeder radiation damage. Possible synergistic effects in the blanket due to the fusion neutron spectrum cannot be quantified at this time.

Saturation of solid-breeder radiation damage

It is estimated [5], based on an analogy with mixed-oxide fuels, that significant bulk radiation damage to the Li_2O solid breeder may saturate at a fluence of 3×10^{25} $n \cdot m^{-2}$ (≈ 0.2 $MW \cdot a \cdot m^{-2}$). The DEMO blanket has to be designed to accommodate the expected variation in Li_2O thermal conductivity (K_B) over the blanket lifetime, so that breeder temperature limits are not violated. The design also has to accommodate estimated uncertainties in the nominal value of K_B at any fluence.

INTOR blanket tests to fluences less than the DEMO lifetime (10 $MW \cdot a \cdot m^{-2}$) must, therefore, be considered to present an added risk to DEMO if the tests are supported only by experiments in fission reactors.

Effects of reduced power density

The results of the study indicate that operating a DEMO-duplicate blanket test specimen (designed for steady-state operation at 2.0 $MW \cdot m^{-2}$) in the pulsed INTOR at a neutron wall load value of 1.3 $MW \cdot m^{-2}$ would result in a breeder temperature below the 410°C design basis minimum required over a large fraction of the breeder volume. This would be unacceptable for all tests for tritium recovery, materials compatibility, heat recovery and lifetime tests because the results depend very strongly on breeder characteristics, which in turn are strongly temperature-dependent. For these tests, the specimens should be partial blanket modules or elements for which certain design details are modified to enable simulation of important aspects of the DEMO blanket performance (e.g. breeder temperature range) to be made. Neutronics tests can and preferably should be conducted, by using DEMO-duplicate partial modules as test specimens.

For all the tests evaluated, the results are predicted to have a linear relationship to the fluence and to be independent of the power density level.

3.3. Results of the evaluation

3.3.1. Neutronics tests

These tests concern only the nucleonic aspects of blanket testing, which include tritium breeding, nuclear heating, the neutron spectrum, activation and decay heat. The emphasis in these tests is on identifying the impact of INTOR operating conditions (availability, plasma duty cycle, neutron wall loading) on the blanket test programme.

As regards the neutronics testing, the difference in the instantaneous flux level of neutron and gamma rays in INTOR and in DEMO is small enough for an extrapolation to be made. The required operating period for the measurements will be about one week to one month, depending on the difficulty experienced in the measurement techniques.

Tritium breeding

To verify the predicted tritium breeding capability of the DEMO blanket the tritium production rate will be measured at different locations in the blanket, and these measurements will then be compared with three-dimensional neutronic analysis. Several techniques are available for measuring the tritium production rate: tritium radiochemical methods, mass spectrometry, track recorder and the gas counter. The test time is about one pulse under INTOR conditions. The expected uncertainty is 5% for the radiochemical and mass-spectrometric techniques. In these cases the tritium breeding measurements can be performed during the early stage of operation with only a few pulses, each pulse having a few seconds of burn time.

Nuclear heating

Nuclear-heating measurements are essential in order to validate the calculational capability of predicting the nuclear-heating profiles for the purpose of heat removal and tritium inventory analysis. Calorimeters and thermoluminescent dosimeters (TLDs) have been considered for measuring the local heat deposition. Calorimeters have been used to measure the neutron-plus-gamma energy deposition rate in fission reactor environments at various power levels. The expected uncertainties are about 5% for measurements during a single INTOR pulse.

The TLD technique has been considered for measuring gamma heating in fusion blankets. TLDs are small and do not perturb the radiation field, but require a large correction factor for gamma heating, due to their response to the hard spectrum of fusion neutrons. The expected uncertainty is less than 30% for the gamma heating determination.

Neutron spectrum

Two methods are considered for measuring the neutron spectrum in the fusion environment: multi-material activation and gas recoil proportional counters. The activation technique has been tested in a fusion-type spectrum [5], where the uncertainty in the unfolded spectrum is 10 to 20%. The proportional counter is capable of 5% accuracy but requires very low neutron wall loading (about 1 $W \cdot m^{-2}$).

Material activation and decay heat

Calorimeters and TLD techniques can be used to measure the decay gamma heating with about 3% accuracy since the radiation field is neutron-free after shut-down. The gamma-ray spectrum can be measured by utilizing sodium-iodide scintillation detectors.

The fluence required is determined by the activation of structural material used in the blanket. For the 316 SS of the reference blanket the decay heat and the induced radioactivity after 1000 $MW \cdot s \cdot m^{-2}$ (four pulses at 1.3 MW wall loading) is 10% of the value after 20 $MW \cdot a \cdot m^{-2}$. After 0.03 $MW \cdot a \cdot m^{-2}$, steel attains 50% of the activation level at 20 $MW \cdot a \cdot m^{-2}$.

Based on this analysis, a very small fraction of 1 $MW \cdot a \cdot m^{-2}$ is adequate to perform activation and decay heat testing.

3.3.2. Tritium recovery tests

The objective of this test is to confirm that steady-state (or quasi-steady-state) tritium recovery (STR) from the DEMO blanket can be accomplished. There is a difference between the power densities in INTOR and in DEMO, and this has a considerable influence on the breeder temperature which may therefore have to be kept within the small allowable range for continuous recovery by additional heating.

The principal factors to be considered are: (1) the number of burn cycles required to build up the tritium inventory in the breeder material 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 (i.e. front to back), and (3) the number of full power pulses needed to saturate the radiation and time-induced changes in the thermophysical, microstructural, and macrostructural properties of the breeder material (i.e. in order to stabilize the tritium release characteristics).

It is estimated that the minimum continuous operating period required to reach the equilibrium condition for tritium recovery will be 20–40 h in a 10 cm region on the plasma side of the blanket, and 300–500 h for a region 40–45 cm deep into the blanket under the INTOR condition (neutron wall loading: 1.3 $MW \cdot m^{-2}$), when the tritium inventory in the blanket is 1 wppm. However, when the tritium inventory in the blanket is 10 wppm, the required continuous operating period to reach the equilibrium will be longer by one order of magnitude. The lithium channel is designed [6] for carrying out studies of some problems of lithium production technology for the blanket. The channel design provides for carrying out the following investigations:

– study of the lithium accumulation, separation and extraction,
– determination of physical-mechanical properties of lithium-containing elements under irradiation,
– estimation of tritium production coefficients in different lithium-containing materials,
– study of the channel-cooling conditions in various regimes,
– testing of the channel design and lithium-containing elements.

3.3.3. Materials compatibility tests

Purge system corrosion and mass transfer

The objective of this test is to examine, at the operating temperature, mass transfer of solid breeder material in the presence of a purge stream including moisture.

It has been estimated that the LiOT (LiOH) formed in the high-temperature breeder zone does not make much contribution to the tritium inventory in the DEMO blanket because the LiOT (LiOH) is re-decomposed into Li_2O and T_2O (H_2O) in the low-temperature ($\approx 400°C$) breeder zone.

Corrosion of structural material within the breeding zone of the DEMO blanket module and in the purge system is expected to be a function of the partial pressure of T_2O, and the associated mass transfer rate of LiOT from Li_2O is, in turn, a function of the partial pressure of the moisture (T_2O). The percentage of moisture in the blanket is estimated to remain essentially constant as long as steady-state tritium recovery conditions are maintained, which should be the case as long as breeder temperatures during irradiation remain within the 410°C and 660°C limits. Since the DEMO blanket is designed to keep breeder temperatures within these limits for all anticipated variations in design and uncertainties in data for blanket design parameters, 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.

Coolant tritium level

The permeability of the oxide film on the breeder side of the coolant tube may possibly change with fluence, but no relevant data exist whilst the permeability of the oxide film on the coolant side of the tube may change with time as the blanket operates. These effects should, however, be measurable from off-line and fission reactor blanket tests and are not expected to differ for the fusion environment.

Based on the above remarks, the coolant tritium concentration (1) is expected to be predictable for DEMO based on the results of fission reactor tests and confirmed by testing in INTOR, and (2) is not expected to change significantly with increasing fluence once steady-state tritium recovery and temperature conditions are reached in the blanket.

3.3.4. Heat recovery tests

The objective of this test is to confirm the possibility of heat recovery for electricity generation in DEMO. The test should demonstrate the temperature conditions of the coolant and the breeder in DEMO.

A simulated model for achieving the temperature conditions of DEMO at the INTOR power density is considered to be necessary for the test specimen, because, in a duplicated model, the temperature of the greater part of breeder remains under 400°C, which is the lower limit for the continuous tritium recovery.

The operating period required for thermal equilibrium in heat recovery tests to be reached depends, of course, on neutron wall loading (power density), operating mode (duty factor) and dimensions and on the physical properties of the specimen. It is estimated that the following continuous operating periods are required for heat recovery tests in INTOR (wall loading: 1.3 MW, operating mode: 200 s burn/46 s dwell)

element of the module in the zone adjoining the plasma:	$>$ 1000 s (4–5 pulses)
element of the module in the rear zone of the blanket:	$>$ 20 000 s ($\approx$ 80 pulses)

Breeder coolant temperature tests

The evaluation of heat recovery tests indicates that testing of blanket specimens in INTOR to fluences lower than 10 $MW \cdot a \cdot m^{-2}$ will not significantly increase the quantifiable risk to DEMO, and, conversely, that testing to fluences higher than 0.2 $MW \cdot a \cdot m^{-2}$ will not produce additional benefits in information. This conclusion results primarily from (1) the estimate that breeder radiation damage saturates to an unchanging condition at a fluence of 0.2 $MW \cdot a \cdot m^{-2}$ (at a given point in the blanket), (2) the expected correspondence between fission and fusion test results, and (3) the anticipated confirmation of the breeder radiation damage versus fluence estimate through fission reactor tests to 10 $MW \cdot a \cdot m^{-2}$ conducted before the DEMO final design.

The main objectives of experiments during the third phase of INTOR are as follows:

- full engineering tests of a fusion reactor blanket segment, including lifetime testing;
- engineering tests of the heat transfer circuit.

Breeder cylinders surrounding the blanket coolant tubes indicate that the maximum temperature difference of 250°C which could occur radially through the cylinder would result in axial and circumferential thermal strains greater, by a factor of seven, than the allowable fracture strain. 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, testing in INTOR to fluence lower than 10 $MW \cdot a \cdot m^{-2}$ should not pose a significant added risk to DEMO in terms of potential loss of breeder physical integrity if fission reactor tests have previously been performed to a fluence of 10 $MW \cdot a \cdot m^{-2}$ to confirm the adequacy of the design. This is principally the case because (1) fracture strain is not believed to change significantly once radiation damage has saturated, and (2) the results of fission reactor tests are believed to be adequate to predict fracture strains for solid breeders irradiated in fusion reactors.

In the case of a DEMO blanket which contains solid Li_2O breeder in the form of small spheres (pebbles) with a diameter of 1 mm [4], the maximum temperature difference across the sphere is predicted to be less than 20°C so that the thermal stresses will not constitute a severe problem. In this test, the deformation, the breakage and the change of packing fraction of Li_2O pebbles due to thermal expansion and under irradiation are to be examined. Effects on the temperature profile in the breeder region and purge stream blockage are also to be investigated.

Interface thermal conductance

This interface consists basically of a stainless-steel layer metallurgically bonded to the solid breeder and to the coolant-containing structure. The fluence should not affect the conductivity of this assembly, except through transmutation of the interface materials (considered negligible). Any major effects would be mechanical, e.g. failure through loss of braze ductility or braze separation from breeder or structure. The development programme anticipated for this (or any other) interface approach would, however, address methods of avoiding such failures. Also, results from fission reactor tests during such a programme are believed to be adequate to predict the long-term performance of the interface in the fusion environment.

Tests of the interface in INTOR to fluences less than 10 $MW \cdot a \cdot m^{-2}$ would not pose a significant added risk to DEMO if fission reactor tests have previously been performed to the DEMO fluence level in order to confirm the adequacy of the design.

3.4. Summary

The blanket tests in INTOR should be carried out to obtain basic data for the design of the DEMO blanket. The tests can be classed mainly as neutronic, tritium recovery, materials compatibility, heat recovery, and lifetime tests. The results of the tests will be used for the confirmation of data from complementary facilities such as fission reactors and engineering mock-up devices, the evaluation and the calibration of the predicted analytical results and the demonstration of the performance of DEMO blanket prototypes.

The benefit and risk estimates of the materials and blanket module testing in INTOR show that INTOR, with the design fluence objective of 2.0 $MW \cdot a \cdot m^{-2}$, will be able to provide sufficiently reliable information on breeder materials and blanket components at an acceptable risk level for extrapolation to the DEMO conditions.

4. BENEFITS TO DEMO OF LONG-TERM OPERATION OF INTOR COMPONENTS

The objective of this task is to quantify the benefits to DEMO of long-term operation of INTOR components. The effort to date has concentrated on defining INTOR and DEMO requirements and determining how long a component must be operated in INTOR to provide confidence that a similar component would also operate satisfactorily in DEMO. Substantial reliability data for many short-life components and components where many duplicate parts are used can be developed by long-term operation of INTOR. Also, confirmation of design method, analytical techniques, environments, manufacturing processes, and operational characteristics will be provided on all components by short-term INTOR operation but will be improved by sustained INTOR operation.

4.1. Approach

The approach used in this effort included: (1) assessing the possible benefits that might be derived from long-term operation – the most obvious benefit was possible improvement of the life/reliability of components; (2) establishing a detailed list of INTOR and DEMO components and identifying the mean-time-between-failures (MTBF) and replacement times; (3) defining the test time that would be required to provide assurance of meeting DEMO's goals as a function of the time INTOR is operated. Test times were estimated by assuming constant failure rates and similar operating environments in DEMO and INTOR. The following criterion was adopted as providing adequate confidence for reliability tests: test times in INTOR should be, at least, 350% of the MTBF needed for the DEMO application. A more in-depth investigation into each component is required to provide more definitive results.

4.2. Potential benefits of long-term INTOR component operation

Design, construction, and operation of INTOR components will provide an abundance of information for DEMO, especially when the INTOR and DEMO

TABLE XII-8. PRINCIPAL BENEFITS TO DEMO

Benefit	Operating time required
Design definition	short
Analytic tool development	short
Fabrication/quality development	short
Performance validation	short, long
Design improvement	short, long
Failure mode definition	long
Failure rate/life	long
Failure recovery time	long
Operational characteristics:	
– Maintenance	
– Safety	
– Durability	

components are similar and operate in similar environments. Most benefits will be derived from short-term operation. Exceptions are performance degradation and life/reliability data which will provide a basis for design improvements. The benefits and the test duration required are summarized in Table XII-8. The benefits can be greatly enhanced by the following actions:

(1) The parts are run through extensive post-test examination at the end of the test.
(2) Improvements are incorporated into INTOR replacement parts.
(3) Failure modes characteristic of the system are detected, the causes identified, and appropriately corrected.
(4) The major components are suitably instrumented, with initial and periodic parameter measurements employed to characterize component parameters and drift. This will significantly aid in determining characteristic life and in the definition of preventive maintenance plans.

4.3. INTOR and DEMO requirements

The major components of INTOR and DEMO were examined to establish requirements. Requirements addressed for INTOR include the mean-time-between-failures (MTBF) and mean-time-to-repair (MTTR). Table XII-9 lists the major systems and the components which dominate the outage for that subsystem.

TABLE XII-9. INTOR AND DEMO COMPONENT REQUIREMENTS

Component	INTOR anticipated (h)		DEMO anticipated (h)		Required test time in INTOR	
	MTBF	MTTR	MTBF	MTTR	Constant failure rate basis	Reliability growth basis
ECRH system	1724	34	3448	17	12 068	20 700
(Launching window – 10)	20 000	256	4×10^4	128	1.4×10^5	2.4×10^5
ICRH system	840	77	1680	38	5880	10 000
(Waveguide launcher – 4)	5000	256	10^4	128	3.5×10^4	6×10^4
(Launching window – 4)	50 000	256	1×10^5	128	3.5×10^5	6×10^5
Diagnostics	2500	198	5000	100	17 500	30 000
Information and control system	166	2	320	1	1120	1920
IF coil system	2747	437	5400	200	18 900	32 400
(Coils – 10)	5×10^4	4380	1×10^5	2000	3.5×10^5	6×10^5
(Power leads – 20)	2.5×10^4	1176	5×10^4	600	1.7×10^5	3.0×10^5
OH coil system	3150	75	6×10^3	40	21 000	36 000
(Power leads – 4)	500 000	1176	10^6	600	3.5×10^6	6×10^6
(Winding)	106 000	1344	2×10^5	700	7×10^5	12×10^5
EF coil system	1400	399	2.8×10^3	200	9800	16 800
(Power leads – 18)	3×10^3	1176	6×10^3	600	2×10^4	3.6×10^4
(Winding No. 3)	62 500	10 248	1.2×10^5	5000	4.2×10^5	7.2×10^5
Control coils	16 600	89	3.3×10^4	45	1.1×10^5	2.0×10^5
Magnet vacuum vessel system	29 400	215	5.8×10^4	107	1.7×10^5	3.5×10^5
Structure	2×10^6	15 000	4×10^6	7500	14×10^6	24×10^6
(Centre post)	10^7	60 000	2×10^7	30 000	7×10^7	12×10^7

Sector modules	5880	194	11 600	100	40 600	69 600
Shield system	180 000	63	3.6×10^5	30	12.6×10^5	21.6×10^5
First-wall system	1075	29	2000	15	7000	12 000
(Armour tiles – 6500)	3000	336	6000	170	21 000	36 000
(Panels – 60)	2200	420	4400	210	15 400	26 400
Pumped limiter	3300	285	6600	140	23 100	39 600
(Limiter module – 10)	4000	336	8000	168	28 000	48 000
Electrical intrasector						
Connector system	21 200	212	42 400	106	148 400	259 400
(Bellows contract – 330)	4500	420	9000	210	31 500	54 000
Spool structure system	3200	167	6400	80	22 400	38 000
Vacuum pumping system	4000	33	8000	17	28 000	48 000
Tritium processing system	10^5	96	2×10^5	50	7×10^5	12×10^5
Fuel storage	10^4	48	2×10^4	25	7×10^4	12×10^4
Heat transport	1000	8	2×10^3	4	7×10^3	12×10^3
Cryogenics system	1000	24	2×10^3	12	7×10^3	12×10^3
AC power system	10 000	96	2×10^4	50	7×10^4	12×10^4
Pellet injector – 2	500	48	1000	24	3500	6000
OVERALL SYSTEM	72	84	144	42	504	864

INTOR's estimates are based on a 20% availability. Reaching the goal of 50% availability for DEMO will require more emphasis on maintenance and increased component reliability. Overall availability is affected by scheduled maintenance intervals, cooldown, bakeout, redundancy, critical path, and numerous other factors but, as a first approximation, DEMO's MTBF was assumed to be twice INTOR's value and the DEMO's MTTR to be one half of INTOR's value. This scaling is based on the following:

$$\text{Availability} = \frac{\text{Operating time (taken as average MTBF)}}{\text{Operating time + downtime (taken as average MTTR)}}$$

Availability = 0.2 for INTOR, 0.5 for DEMO.

Solving the above for DEMO values with respect to INTOR results in a possible combination of:

$$\text{DEMO MTBF} = 2 \times \text{INTOR MTBF}$$

$$\text{DEMO MTTR} = 1/2 \times \text{INTOR MTTR}$$

The resulting MTBF and MTTR values are included in Table XII-9 along with the INTOR values. In some instances improvements in reliability will not be achievable. In these cases all effort will be placed on decreasing the time-to-replace. Conversely, in other cases all emphasis may be placed on improvements in reliability. When neither of the above modifications is possible or cost-effective, adjustments will also be required between component requirements. Some components will not be on the critical path and not require these improvements.

It appears that an INTOR test time that is a factor of three to five times longer than the required MTBF of DEMO components can provide the necessary confidence for proceeding with DEMO (see Section 4.4). For purposes of initial calculations, a factor of 3 1/2 was selected and used to develop the test time requirements shown in Table XII-9 for INTOR for specific components. Operation of INTOR for 10 to 15 years with a 20% to 35% availability was considered possible. This range of variables results in possible test times of 10^4 to 5×10^4 hours in INTOR. A review of the items in Table XII-9 indicates that little information can be provided to develop MTBF data for the components which dominate the outages. Nevertheless, INTOR operation will provide MTBF data on the overall system and all major subsystems. As the system begins operation the weak components will fail and the system MTBFs will become apparent. Multiple failures of the same component will identify where design changes would be beneficial to a DEMO system or necessary to meet INTOR's availability goal. Failure and reliability

TABLE XII-10. TEST TIME REQUIRED TO ATTAIN AN ADEQUATE DATA BASE FOR DEMO[a]

Item	Test time required (h)
Overall INTOR system	500
Pellet injectors	3500
System interface monitor and control	2700
ICRH system	5880
First-wall system	7000
Heat transport system	7000
Cyrogenics system	7000
EF coil system	9800
ECRH system	12 000
First-wall panels	15 400
ICRH reactor building transmission	17 500
Diagnostics	17 500
TF coil system	18 900
OH coil system	21 000
Spool structure system	22 400
OH power conversion/protection	23 000
TF power conversion/protection	28 000
Divertor/pumped limiter	28 000
Vacuum pumping system	28 000
ICRH control	35 000
ICRH waveguide launcher	35 000
Sector modules	40 000
ICRH frequency generator	46 000

[a] Values for NBI were not estimated.

analysis of components will lead to recommendations for design changes or addition of redundancy.

Although it will not be possible to develop many reliability data on many of the critical components that will dominate the availability of the reactor, it will be possible to provide much information for DEMO on components in which failures will occur frequently in INTOR but do not result in reactor shutdown.

TABLE XII-11. COMPONENTS WHERE AN ADEQUATE DATA BASE CAN BE DEVELOPED IF INTOR OPERATES 25 000 HOURS

Item	Failure rate (Failures per hour)	Number of components required to develop data base
Heat exchanger	5×10^{-6}	28
Thermocouples	1×10^{-6}	140
Microswitches	1.7×10^{-6}	82
Strain gage	2×10^{-6}	70
Flexible couplings	2×10^{-6}	70
Motors – small	5×10^{-6}	28
Penetrations	1×10^{-7}	1400
Pumps – centrifugal	15×10^{-6}	10
Pumps – reciprocating	100×10^{-6}	2
Relays	1.5×10^{-7}	930
Hydraulic scrubbers	2×10^{-6}	70
Transformers	1×10^{-6}	140
Valves – ball	$1–2 \times 10^{-6}$	70–140
Valves – electric gate	$3–6 \times 10^{-6}$	20–40
Valve operators	$2–4 \times 10^{-6}$	35–70
Pressure vessels	1×10^{-6}	140
Circuit breakers	$1–2 \times 10^{-7}$	700–1400
Switch gear assemblies	$1–2 \times 10^{-7}$	700–1400

Table XII-10 lists those systems and components which have been tabulated to date for which test times will be adequate to characterize the MTBF. Again the factor of 3.5 times the DEMO's MTBF was used to estimate test time required.

For many other components, such as the items listed in Table XII-11, INTOR will also provide adequate data to identify the proper levels of redundancy for DEMO. No part counts are available for the INTOR. Table XII-11 shows the number of components as well as the failure rates of the components in INTOR that would be needed to provide adequate data on reliability.

This evaluation has identified a clear need to establish a procedure through which data are gathered in INTOR as a useful basis for DEMO. Many of the components have 'known' failure rates that were based on non-fusion environments. Changes in environment can affect results substantially.

INTOR will provide an environment similar to DEMO, and using INTOR to improve the reliability base for the non-critical components will assure that DEMO can operate more reliably.

Operation of the entire INTOR reactor system for a period of three to five times longer than the time required for demonstrating its 20% availability can also provide adequate confidence that DEMO can meet its system requirements. The nuclear system in DEMO is an exception because its environments are much more severe, and in this DEMO differs from INTOR.

4.4. Determination of test time required in INTOR

The testing time required in INTOR to support DEMO depends on the exact mission of INTOR:

(1) *Confidence building* – If the INTOR objective is simply to achieve its availability target of 20% rather than to continuously improve its availability, then extended tests could be used to build confidence that the INTOR components will have reliability that is acceptable to DEMO (assuming they have that inherent capability in the first place).

(2) *Reliability growth* – If INTOR is to be used as a test bed for evolving a DEMO system which can meet an availability goal of 50%, then the Test, Analyse and Fix method can provide a large amount of useful data for DEMO.

The two options above are described in the following sections.

4.4.1. Confidence building

With this option a component is assumed to have an acceptable MTBF for DEMO but data are insufficient to give confidence that the MTBF can really be achieved. The purpose of testing in INTOR then is simply to build confidence.

Unknowns to be resolved with confidence building are failure modes, failure distributions, and failure rates. For purposes of establishing a required test time for this option constant failure rates for components were assumed. This approach is commonly used to estimate equipment reliability and derive test plans for continuously operated equipment and is generally appropriate for large complex systems. The true failure rate distribution can only be determined by a test programme and variations from a constant failure distribution can have a substantial effect on the MTBF; if, however, both the INTOR and DEMO components use the same design and experience similar environments then the INTOR and DEMO parts are likely to have the same failure distribution with the exception of the nuclear system. This presumption of similar failure distributions helps to mitigate the shortcoming of the constant-failure-rate assumption.

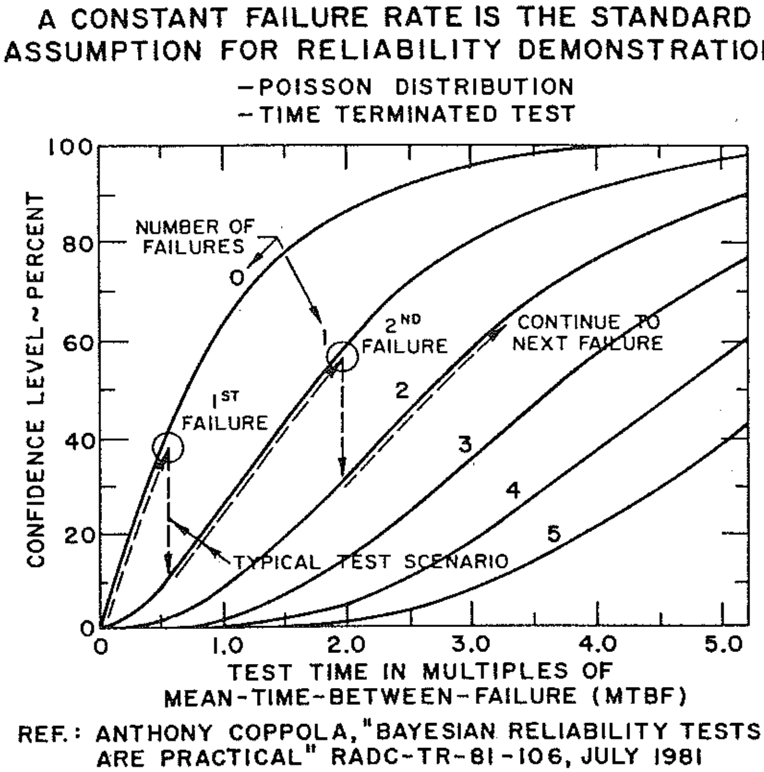

FIG.XII-10. Confidence level versus test time.

The confidence level required for DEMO is unknown at this point, but it is clear that establishing a level based on statistical considerations would require very high confidence levels in all components; however, it is assumed that a level of 80% on components is likely to be appropriate for this stage of reactor development.

The Poisson curve (Fig. XII-10) indicates that a test time of 1.6 times the goal MTBF is appropriate for 80% confidence if no failures occur during the test and this factor increases to 3 and 4.3, respectively, for one and two failures of the component.

If the confidence level required for DEMO is assumed to be the same as the level required to justify a go-ahead on INTOR, then a test of twice the duration required for the INTOR would be adequate to demonstrate an MTBF for DEMO that is double that for INTOR.

4.4.2. *Reliability growth*

INTOR long-term operation could be extremely valuable for DEMO if INTOR adopts an approach to reliability growth for which the primary purpose is to develop the overall system availability by increasing component reliability and

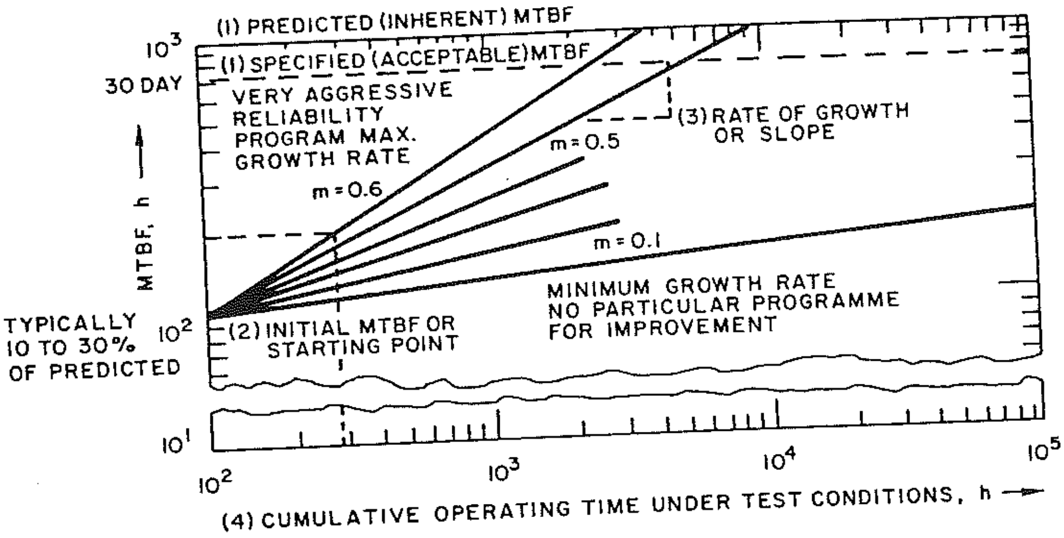

FIG.XII-11. MTBF versus time.

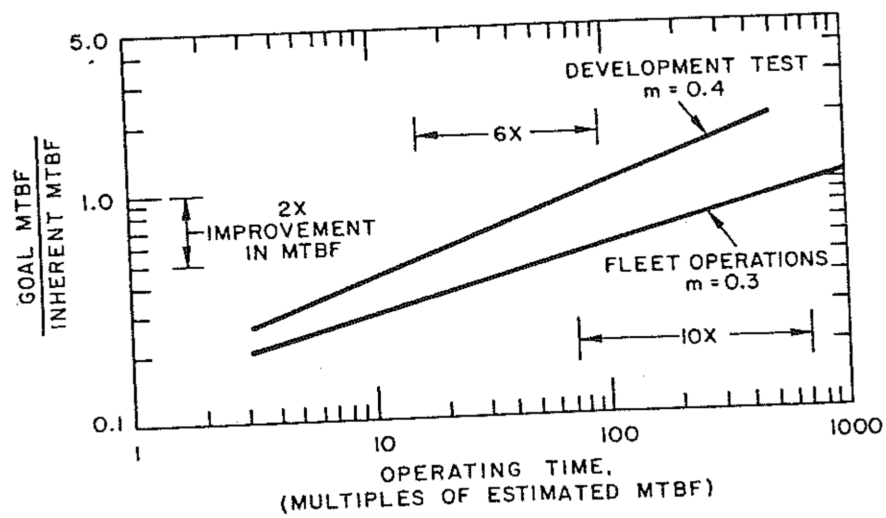

FIG.XII-12. MTBF ratio versus multiples of estimated MTBF.

demonstrating reductions in times-to-replace through improvements and maintenance equipment and methods.

Past experience has shown that the rate of increase in the MTBF of components is a constant for most programmes. The rate-of-growth constant is related to a programme's emphasis on reliability and ranges from 0.1 for programmes with little emphasis on reliability to 0.6 for programmes where everything possible is done. A plot showing this growth is given in Figure XII-11. Past experience indicates that complex systems generally start operation with MTBFs that are 10%

to 30% of their initial goal and must grow to meet their goal. This experience implies that long-term operation is needed just to attain INTOR's availability goals. Furthermore, this experience underscores the desirability of developing reliability growth with INTOR to try to improve DEMO-relevant MTBFs. Figure XII-12 is similar to Fig. XII-11 but the curve is normalized so that the MTBF ratio is plotted versus multiples of the estimated MTBF. For example, an MTBF ratio of 0.1 indicates that the actual value of the MTBF being measured in operation is 10% of the potential MTBF predicted by analysis. Since INTOR is developmental in nature, it is assumed that it will have a growth of 0.4. The impact of this type of analysis on the required operating test time is described in the following example in which DEMO must have a 100% higher MTBF than INTOR.

Assumptions:

Inherent component MTBF = 400 h

Goal MTBF = 100 h for INTOR, 200 h for DEMO

slope = 0.4 for INTOR, 0.3 for DEMO

With the asumptions above, the following values may be derived from Fig. XII-12:

	For INTOR	For DEMO
To indicate 100-h MTBF:	$\frac{\text{Goal MTBF}}{\text{Inherent MTBF}} = 0.25$	$\frac{\text{Goal MTBF}}{\text{Inherent MTBF}} = 0.25$
	t = 2.5 × (Inherent MTBF) = 1000 h	t = 7 × (Inherent MTBF = 2800 h)
To indicate 200-h MTBF:	$\frac{\text{Goal MTBF}}{\text{Inherent MTBF}} = 0.5$	$\frac{\text{Goal MTBF}}{\text{Inherent MTBF}} = 0.5$
	t = 15 × (Inherent MTBF) = 6000 h	t = 70 × (Inherent MTBF) = 28 000 h

Therefore, INTOR can be used to increase the MTBF to a level acceptable to DEMO by testing for an extra 5000 h (6000–1000) whereas DEMO would have to test 25 200 h longer to achieve the same growth. Testing a component in INTOR to six times (15/2.5) longer than required to achieve the confirmation that INTOR can meet its 20% availability goal will provide adequate confidence that DEMO can proceed. (INTOR is also a more economical machine for developing component reliability than DEMO.) An obvious concern in proposing longer testing in INTOR is the increase in INTOR operating time that results from applying the factor of six to the INTOR operating time to demonstrate its MTBF required for a 20% availability. In the previous example, for a part with an inherent MTBF 200 h (actual), the test time to indicate a 100-h MTBF in INTOR is 3000 h and the test time to achieve a 200-h MTBF is 18 000 h. Low inherent MTBFs (relative to the required MTBF) can greatly extend the test time. The test times required for major DEMO components are listed in Table XII-9; the times given are based on the assumption that the inherent MTBF is about twice that required for DEMO.

A comparison of the test times in Table XII-9 obtained by using the two differing criteria indicates that for some components a wide range in necessary test times exists and also indicates the need for detailed evaluation of individual components to identify past operating histories, evaluation to determine potential failure modes and definition of a complete test programme including INTOR and complementary test programmes.

4.5. Specific component evaluation

A preliminary examination of the plasma heating and current drive systems and TF coils was performed to identify the potential failure modes and to determine what specifically might be learned for DEMO by testing in INTOR.

The current drive and heating system survey identified many potential system limitations and identified a test time required to understand these limitations. Long-term testing benefits include determination of irradiation effects on dielectric windows and phase monitors and an understanding of possible damage to the phase monitor probe due to electron density build-up.

Results of the TF coil survey indicate that the predominant cause of failure is a result of design deficiencies and not the result of long-term effect. Operation in INTOR will provide some longer-term effects such as conductor, stabilizer, and insulation degradation. These effects can be monitored by addition of test specimens and also through failure analysis of any failures in the INTOR magnets.

4.6. Future effort required

To maximize the benefits to DEMO from INTOR operation and from the INTOR development programme, two programmes are recommended: (1) an active programme to collect the past-operational experience, to monitor development of INTOR components, and to monitor INTOR operation, and (2) a programme to improve components and designs during INTOR's operation.

An early effort is required to determine past experiences with large prototype nuclear and related facilities which may be applicable to the INTOR system. The study should determine past start-up and early operational experience of other facilities and use this information to identify those components likely to be a source of difficulty in INTOR. This study should develop operational information on reactor components which are currently in use. Information on nuclear coolant loop components (pumps, valves, pipes, etc.) and nuclear containment buildings can be obtained from experience on fission test reactors. Information on large power supplies, magnet systems, and vacuum system components can be obtained from experience on particle accelerators and small fusion devices. In this manner, not only would the scope of components to receive detailed evaluation be narrowed, but actual operational data would be used in developing initial operational predictions for some of these critical components and would provide a basis for additional development.

During INTOR operation and INTOR component development testing, numerous failure distributions will presumably be observed. These will depend on the component observed, the environment, and failure mechanisms. The failure and repair data should be extracted from the test programme and used to identify the failure distributions, failure rates, restoration times, and ultimately the observed availability of the INTOR system. From these data, additional improvements can be recommended and incorporated, including design improvements, equipment redundancies, individual and composite scheduled preventive maintenance intervals, and associated safety improvements. Expectations of how a future DEMO system will perform can then be predicted.

A principal objective of the INTOR programme should be identification of problems, implementation of improvements, and evaluation of the operational data to characterize the reliability/availability and safety of the INTOR system. The design and the operation of a future DEMO system would benefit from these findings and improvements, provided that the reliability data are well documented and include proposed and implemented improvements. To verify that any proposed improvements work as intended, ideally they should be incorporated into the INTOR system and their acceptable operation should be verified. In practice, incorporating changes into INTOR based on proposed improvements in component design for DEMO may be difficult, especially where such changes represent high costs. However, even where such proposed changes are not carried out in INTOR before their use in DEMO, the changes and expected improvements in performance and reliability should be well documented.

5. CONCLUSIONS

The evaluation of structural material testing described in Section 2 shows that INTOR will have to be constructed with a well known material, i.e. austenitic stainless steel. The data base necessary for construction has to be completed by simulation experiments using the existing devices, such as fission reactors, accelerators and, if possible, FMIT. It would be difficult and costly to arrive at the construction of DEMO with a fully developed new alloy. Therefore, there is a large probability that DEMO will use the same structural material as INTOR. This will cause a delay of the introduction of a new alloy as the structural material for the commercial reactor. In this case, the sorting of the most promising candidates is done in INTOR and the most important irradiation studies are performed in DEMO. Although austenitic stainless steel is regarded, at present, as the only realistic structural material to be used up to DEMO, a strong effort should be made to develop new alloys to be used for future commercial fusion reactors, in parallel with the effort to improve the present austenitic stainless steel.

Testing of structural materials in INTOR at a higher neutron fluence will provide a better data base for constructing the DEMO. The evaluation of the necessary neutron fluence for such testing indicates that a fluence of 2 $MW \cdot a \cdot m^{-2}$ will probably be the minimum because tests with a lower fluence will provide little useful information on some of the important structural material properties. Thus, there are doubts whether the value of 2 $MW \cdot a \cdot m^{-2}$ is sufficient and some of the experts strongly feel that higher fluences should be the goal in INTOR testing. It should be noted here that the degree of risk to be associated with designing DEMO decreases with the increase in neutron fluence to be achieved in INTOR testing, whereas the degree of design risk of INTOR itself becomes larger with higher fluence in INTOR.

The evaluation of the blanket tests to be conducted in INTOR has shown that there appear to be no significant benefits that can be quantified at this time from conducting blanket tests in INTOR to fluence levels significantly beyond about 3×10^{25} $n \cdot m^{-2}$, about 0.2 $MW \cdot a \cdot m^{-2}$ as measured at the first wall. The primary factors which lead to this conclusion are, among others: 1) the present estimate that breeder bulk radiation damage may saturate at a fluence of about 0.2 $MW \cdot a \cdot m^{-2}$, 2) the conclusion that fission reactor tests are likely to produce essentially the same results as fusion reactor tests, 3) the estimate that a cyclic INTOR operation with a significantly lower power level than that in DEMO will reduce the usefulness of results in some tests, such as heat and tritium recovery tests. The expert evaluation shows that irradiation tests of solid breeders should be conducted as soon as possible, using both fission reactors and high-energy neutron sources, to establish firmly the relationship of changes in breeder properties with neutron fluence levels, and to verify the correspondence between results of breeder tests in fission and fusion environments. It is concluded that useful information will be obtained from INTOR blanket tests with a projected neutron fluence at around 2 $MW \cdot a \cdot m^{-2}$.

The evaluation of the benefits of long-term operation of INTOR components to DEMO has shown that it is necessary to establish an appropriate procedure with which the data should be collected from INTOR operation, so that it can be analysed properly to provide the data base for designing DEMO components. In the evaluation, failure rates of many of the components have been estimated, based on various assumptions. It should be noted that these assumptions can affect the results of the evaluation very substantially. Nevertheless, it is concluded that operation of INTOR will create an environment similar to that in DEMO and provide very useful information for more reliable construction and operation of DEMO. In particular, using INTOR to improve the reliability of the components will contribute to achieving the desired availability goal in DEMO.

Appendix

6. TEST MODULES FOR SIMULTANEOUS TRITIUM BREEDING AND ELECTRICITY GENERATION

6.1. Introduction

In the tritium-breeding blanket described in Chapter VIII, the coolant temperature and pressure are kept at low values. This restriction aids the design by keeping to an acceptable level the rate of tritium permeation through the containing walls and, by enabling a minimal amount of structural material to be used, assists in the neutron economy of the system. The conservative rating of the components helps in the attainment of high reliability and hence good tritium production. The coolant temperature ($\approx 100°C$) and pressure of about 0.1 MPa are, however, well below those required for efficient electricity production, and it is therefore planned in Stage III of the operation of INTOR to replace at least one of the low-temperature breeding blanket sectors by a sector having coolant conditions suitable for reasonably efficient electricity production.

Three preliminary blanket designs were considered at the INTOR Phase-Two-A Workshop in Vienna, two with solid breeders, one helium-cooled [3] and the other water-cooled [4], and one with a liquid-metal breeder with carbon dioxide cooling [6]. Since production of large amounts of electrical power is not one of the objectives of INTOR, none of these three designs, nor the overall problems of electricity production, were extensively discussed at the Workshop in Vienna. In this section, therefore, the emphasis is mainly on a brief description and comparison of the designs.

6.2. Overall design requirements

The sector must be a direct replacement for a standard low-temperature tritium-breeding sector. It must, therefore, have the same overall dimensions, fixing arrangements in the torus and be capable of being handled by the remote equipment used for the standard sectors. As in the standard sectors, the breeding blanket covers only the top and the outermost sides of the torus. For the sake of simplicity, it is assumed that no neutral injector or RF heating aperture will be included in the electricity-producing sector initially used in INTOR.

As far as divertor/limiter and first wall are concerned, these should be of the same general design as for the standard sectors, so as not to disturb the toroidal symmetry of the system with respect to the plasma. In the present designs, no attempt is made to recover, for electricity generation, the substantial portion of the output power impinging upon the divertor/limiter and first wall.

A local tritium breeding ratio (TBR) of at least 1.25 should be the aim; for a reactor with 80% blanket coverage in the torus, this would give a net TBR of unity.

6.3. General considerations for blanket design

6.3.1. Neutron multiplier

The increased proportion of structural material and hence parasitic absorption of neutrons in the high-temperature blanket makes the use of a neutron multiplier even more important than in the low-temperature tritium-breeding blanket, in which lead is proposed for the multiplier. The higher operating temperature of the electricity-producing blanket would result in melting of the lead in the solid-breeder blanket and hence to difficulties in supporting and cooling it. Beryllium has, therefore, been proposed for both solid-breeder blanket designs.

6.3.2. Coolant

Pressurized water and high-pressure helium have been considered as coolants for the solid breeders. Water has a high capability of heat removal and can maintain the temperature of the structure at a comparatively low level. It allows the use of existing materials for the structure, assuming that the steam conditions are about the same as in a PWR. However, detritiation of the water coolant may be a difficult problem, so that countermeasures to reduce the tritium permeation rate through the coolant tubes will have to be investigated. The compatibility of the breeding material with water, especially in the event of failure of a coolant pipe, requires consideration. Control of the solid-breeder temperature within its rather narrow temperature window by means of a gas-filled space of adjustable thermal conductance between the breeder and the water, as for the low-temperature blanket, may be difficult because of the smaller temperature difference between the breeder and the water in the case of the electricity-producing blanket.

Helium can be used both as purge gas for carrying away the tritium from the breeder and as a coolant for the breeder, thus leading to some simplification of the services to the sector. In principle, helium could be used at higher temperatures than water, leading to increased thermal efficiency of electricity generation, but temperature limitations in the present structural material, as set by corrosion and tritium permeation, may preclude exploiting the higher operating temperature. The helium coolant pressure is only about one third of that for pressurized water, which leads to thinner-walled tubes for helium with reduced neutron absorption. This advantage may, however, be somewhat offset by the increased space required for helium ducts and manifolds compared with those for water. In terms of safety, helium has an advantage in that the stored mechanical energy in the helium is only about 0.3% of that for the equivalent pressurized water coolant.

Carbon dioxide is a well tried coolant gas in fission reactors; it has good heat transfer characteristics. Unfortunately, its use in solid breeders as a combined coolant and purge gas is precluded by its possible chemical reactions with the breeder and with tritium. However, in the case of the liquid-metal breeder, the coolant is contained in pipes immersed in the breeder and is, therefore, separated from both the breeder and the tritium. In this case, carbon dioxide can, therefore, be used as coolant.

6.4. Helium-cooled solid-breeder blanket [3]

The initial main objectives were to develop a design which operated at high temperature, but which at the same time gave maximal safety for personnel and for the reactor in the case of coolant pipe failure; hence the choice of helium as a coolant. A pressure of 5 MPa is used as a combined breeder coolant and tritium purge gas in order to simplify the coolant pipe arrangements as much as possible. Tritium has, therefore, to be recovered from a helium stream containing a much lower concentration of tritium than in the low-temperature breeding blanket of Chapter VIII and in the water-cooled blanket of Section 6.5 below. Methods for this extraction have not yet been considered in detail.

The multiplier is beryllium, giving an acceptable local TBR of 1.28 when using lithium silicate enriched in ^{6}Li to 30%.

Austenitic stainless steel is used for the structural material which limits the outlet temperature to 580°C, so that there is no advantage in using a directly coupled helium turbine. A heat exchanger and steam cycle are, therefore, proposed.

Figure XII-13 indicates the stainless-steel hexagonal tube structure which holds the breeder and the multiplier. These hexagonal tubes are aligned along the toroidal direction. The helium cooling gas flow is arranged to flow first along the cusp-shaped channels between the hexagonal tube and the breeder (or breeder and multiplier) which is canned in a thin stainless-steel circular tube. It then returns via the breeder (or breeder and multiplier); the diameter and the spacing of the coolant holes are arranged to give the correct temperature (400–600°C) of the breeder (Li_2SiO_3) under operating conditions. Each hexagonal tube immediately

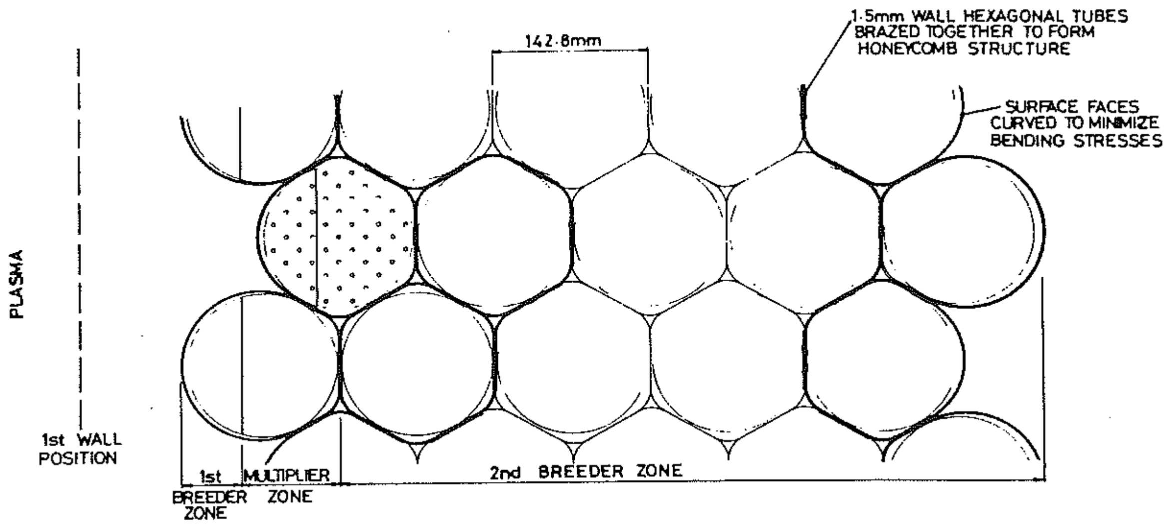

FIG.XII-13. Helium-cooled blanket (cross-section).

behind the first wall contains a lithium silicate breeder zone at the front of the tube to capture the neutrons reflected by the beryllium multiplier zone at the back of the tube. The tubes further back in the blanket contain lithium silicate only. With beryllium as neutron multiplier, it was found unnecessary in this design to use any additional neutron moderator.

The helium coming out of the blanket at 580°C is passed through a super-heater and boiler to produce steam at 17 MPa and 500°C for passing to the turbine. About 4% of the helium is bled off from the helium loop and processed for recovery of the tritium.

Specifications of this blanket are given in the first column of Table XII-12.

6.5. Water-cooled solid-breeder blanket [4]

This has some similarity with the low-temperature tritium-breeding blanket of Chapter VIII in that the helium purge gas is also used as an adjustable thermal barrier between the breeder and the water coolant.

Pressurized water was chosen as coolant because it seemed that there would not be time for developing new structural materials capable of working at higher temperatures than present-day steels. A titanium-modified austenitic stainless steel (modified 316 SS) is proposed.

The concept of a breeder outside tube (BOT) is used as shown in Fig. XII-14, with Li_2O as the breeder material. It is used in the form of small pellets with helium gaps between the pellets and the water cooling tube for temperature control of the breeder. In a DEMO reactor this may, however, have to be modified

TABLE XII-12. MAJOR SPECIFICATION OF TEST BLANKETS FOR ELECTRICITY GENERATION AND TRITIUM BREEDING

	Solid breeder Helium-cooled blanket	Solid breeder Water-cooled blanket	Liquid breeder CO_2-cooled blanket
1. LOCATION			
Location	Outer blanket region	Outer blanket region	Outer blanket region
Fraction of major circumference occupied by sector	$\frac{1}{12}$	$\frac{1}{12}$	$\frac{1}{12}$
Number of test blankets per sector	2 modules	6 modules	2 modules
2. BREEDING			
Breeder			
Material	Li_2SiO_3	Li_2O	$Li_{17}Pb_{83}$
Concept	Cylindrical pellets (inside cooling)	Small spheres, BOT (inside cooling)	Liquid in tank with internal cooling pipes
Temperature range	400–600°C	500–800°C (nominal)	
Control of breeder temperature	Density and size of coolant channels	He gap and coolant tube arrangement	Spacing of cooling pipes
Content of ^6Li	30%	Natural	Natural
Breeder inventory			
Blanket module		1.6 t	
1 sector	7.4 t (Li_2SiO_3)	9.6 t (LiO_2)	15 t

Neutron multiplier			
Material	Beryllium	Beryllium	Lithium-lead eutectic
Thickness	100 mm	50 mm	As blanket ≅ 50 cm
Neutron moderator			
Material	None	None	None
Tritium breeding ratio			
Local	≈ 1.28 (F/W cooled by D_2O)	≈ 1.25 (F/W cooled by D_2O) ≈ 1.15 (F/W cooled by H_2O)	1.2 (F/W cooled by CO_2)
Tritium recovery			
Type	Continuous	Continuous	Continuous
Purge gas	Helium coolant	Helium	None
Pressure	5 MPa	0.1 MPa	—
Tritium permeation per sector[a]			
into water cooling through breeder containers or moderation jackets	360 $Ci \cdot d^{-1}$ (He/water heat exchanger) Recovered via sector pumping system	480 $Ci \cdot d^{-1}$ (blanket cooling tubes)	10^4 $Ci \cdot d^{-1}$ (10 $Ci \cdot d^{-1}$ with coating)

[a] Without coating.

TABLE XII-12 (cont.)

	Solid breeder Helium-cooled blanket	Solid breeder Water-cooled blanket	Liquid breeder CO_2-cooled blanket
3. HEAT RECOVERY			
Breeding region			
(including multiplier, moderator, and blanket wall)			
Coolant	Helium	H_2O	CO_2
Inlet pressure	5 MPa	15 MPa	4 MPa
Inlet temperature	390°C	280°C	220°C
Outlet temperature	580°C	320°C	500°C
Flow rate (per sector)	30 $kg \cdot s^{-1}$	116 $kg \cdot s^{-1}$	14 $kg \cdot s^{-1}$
Electricity generation			
Thermal efficiency	39%	33%	37%
Electrical power output	8 MW(e)	10.3 MW(e)	5 MW(e)
4. STRUCTURE			
Blanket			
Concept	Breeder in tube	Breeder out of tube	Breeder in tank
Material	Precipitation-hardened austenitic SS	Titanium-modified austenitic SS	SS tank walls and cooling tubes
Minimum wall thickness	2.5 mm	6 mm	11 mm
Coolant tube size	≅ 150 mm (breeder container)	8 mm^{ID}/10 mm^{OD}	20.4 mm^{ID}/22.0 mm^{OD}
Maximum temperature of structural material	450°C	≈ 400°C	650°C

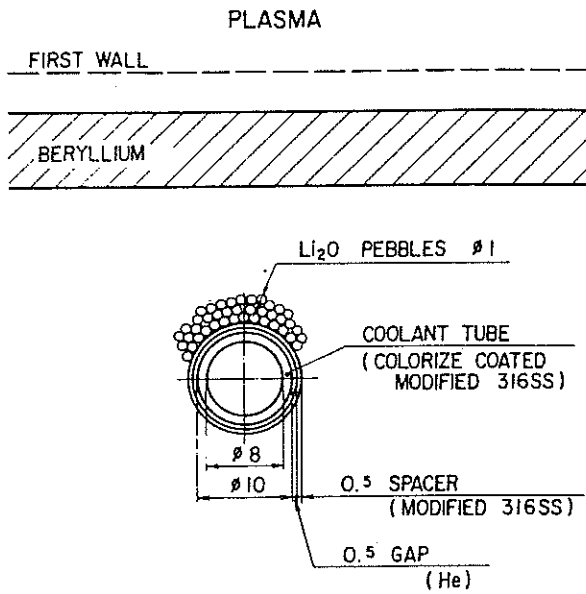

FIG.XII-14. Concept of breeder outside tube.

because of the combination of high heat loads and high-temperature coolant leading to impractically narrow gaps.

The allowable temperature range of the breeder was assumed to be 400 to 1000°C. From the point of view of corrosion, it is desirable that the lowest temperature should be lower than 550°C. The temperature range leaves a margin for the uncertainty in the prediction of the power density. In this design, the maximum temperature of the breeder is about 800°C and the minimum about 500°C.

To improve the TBR, the first wall is integrated with the blanket vessel, and a beryllium neutron multiplier of 50 mm thickness is used. Natural lithium oxide pellets are packed into the blanket vessel. A diameter of 1 mm was chosen for the pebbles in order to give good heat conduction to the coolant tubes. The tritium generated is recovered by the helium purge gas flowing in the breeding zone. The local tritium breeding ratio is 1.15 with H_2O as the coolant in the first wall and 1.25 with D_2O.

The tritium permeation rate into the coolant was estimated as being much too high and, therefore, needs to be reduced by coatings or other means. Such a coating material would need to have a small tritium diffusion coefficient, a high thermal conductivity and good integrity during reactor operation.

Specifications of this blanket are given in the second column of Table XII-12.

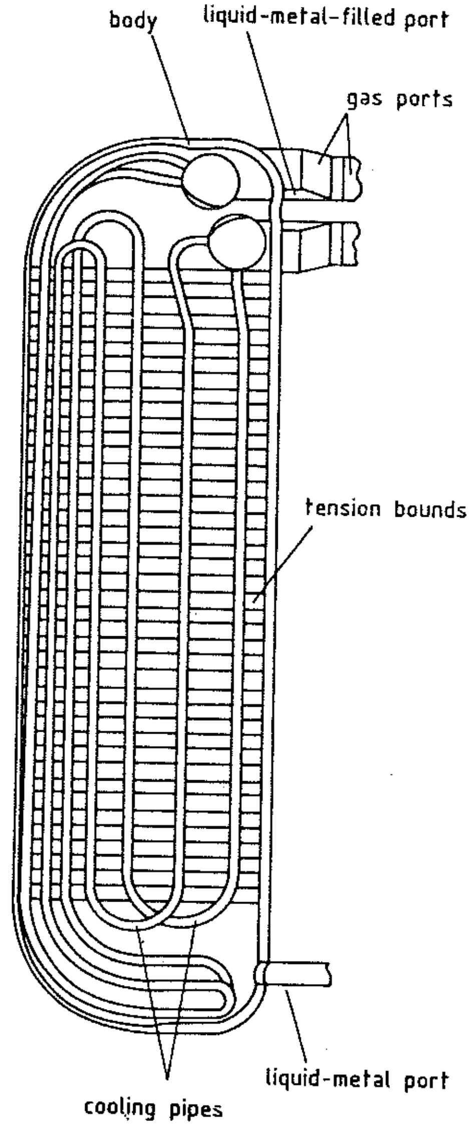

FIG.XII-15. Carbon-dioxide-cooled blanket.

6.6. Carbon-dioxide liquid-metal breeder blanket [6]

The liquid metal, $Li_{17}Pb_{83}$, is contained in modules 2.5 m × 2.0 m with a depth from the front to the back of 0.7 m (Fig. XII-15) made of stainless steel. To increase the strength, the top, bottom and side walls of the modules are circular, and the front and back walls have bracing bars between them.

The breeder has serpentine tubes immersed in it, containing CO_2 at 4 MPa pressure, and the density of the pipes in the breeder is adjusted so as to correspond to the local heat production of the region. The length of a pipe is 10 m, the average spacing 32 mm and the total number of pipes is 62. The pipes are connected to hot and cold manifolds. In addition to gas ports to the module, there are 75-mm-diameter ports to enable the lithium lead eutectic to be pumped out for recovery of the tritium.

The first wall is attached to the front of the blanket module and is cooled separately.

The energy conversion scheme proposed is as follows: the carbon dioxide at a pressure of 4 MPa emerges from the blanket at 500°C and is passed through a boiler to give steam at a pressure of 3.5 MPa at a temperature of 435°C which is passed to the turbine. In the CO_2 circuit, a heat store, consisting of a cylinder filled with steel rods, is fitted in order to reduce the fluctuations in temperature of the CO_2, caused by the pulsed operation of the reactor, to 10°C, so reducing the thermal stresses in the components. Detailed calculations of these stresses have been made for various parts of the thermal circuit. The most severe stresses occur in the components in the region of the breeder nearest to the plasma where the temperature excursions are at a maximum, but it is estimated that these stresses can be accommodated.

Specifications of this blanket are given in the third column of Table XII-12.

6.7. Conclusions

From the three preliminary designs discussed, it appears that a blanket sector for INTOR where both electricity generation and tritium breeding are carried out simultaneously, would be feasible. Both helium cooling and water cooling seem to be practicable for the solid breeder and carbon dioxide for the lithium lead eutectic. Further work will be required in order to make a choice between these breeders and coolants and to solve several problems which exist in all the designs.

REFERENCES TO CHAPTER XII

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