Fusion Materials – Adapting to Expanding Timetables

G.L. Kulcinski

May 1986

UWFDM-687

Presented at IAEA Workshop on Fusion Reactor Design, 26 May – 6 June 1986, Yalta, USSR.

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

Fusion Materials – Adapting to Expanding Timetables

G.L. Kulcinski

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

May 1986

UWFDM-687

Presented at IAEA Workshop on Fusion Reactor Design, 26 May – 6 June 1986, Yalta, USSR.

FUSION MATERIALS -

ADAPTING TO EXPANDING TIMETABLES

by

G.L. Kulcinski

Nuclear Engineering Department
Fusion Technology Institute
University of Wisconsin-Madison
Madison, Wisconsin
United States of America

To be presented at the IAEA Workshop on Fusion Reactor Design
Yalta, USSR, May 26 to June 6, 1986

FUSION MATERIALS - ADAPTING TO EXPANDING TIMETABLES

Abstract

It now appears that commercial DT fusion reactors will not be available much before the mid 21st century. This delayed schedule has reduced the urgency with which fusion material research is viewed since the last (1981) IAEA Reactor Design Meeting. Nevertheless, progress since the last meeting has produced significant results.

There have been considerable advancements in the development of ferritic steels, both from an experimental and theoretical standpoint. More is understood about the steady state swelling regime for austenitic alloys and the microchemical evolution in both types of steels are beginning to yield important insights. Two important areas of progress in nonmetals have been the insitu-extraction of T_2 from solid breeders during irradiation and improved performance of nonmetallic coatings for high heat flux materials. The community also has a better understanding of the melting and vaporization of materials during intense heat pulses. The ICF design community has accepted the internal protection schemes afforded by gas or liquid environments and these greatly mitigate the effects of the short pulsed damage characteristics of unprotected bare walls. Finally, irradiation tests of electrical insulators and superconducting materials reveal that the shielding requirements can be relaxed from previous designs.

There are some areas where very little progress has been made. There still is no "permanent" first wall design for fusion plants (except for some liquid metal protected ICF reactors) and very little experimental work has been forthcoming on nonferrous metal alloys for fusion structures. There is very little information on how nonmetallic coatings will stand up to simultaneous neutron and ion bombardment. The lack of high energy neutron sources is still a hindrance to progress in fusion and very little data on damage to short wave length laser mirror coatings is available. First wall designers are still waiting for more quantitative descriptions of the radiation associated with disruptions and from ICF targets and there is essentially no information on the effects of pulsed irradiation on the response of metals and alloys.

New ideas which have been largely developed in the past 5 years include the low activation steels, V alloys for possible structural applications and Cu alloys for high heat flux components. The development of Be/BeO coatings for tokamak limiter or divertor surfaces has also progressed. In the area of ICF, the first complete materials test facility for pulsed damage, SIRIUS-M, has been proposed. Several new tandem mirror materials test facilities have been designed in the past 5 years. The discovery of a large resource of He^3 on the lunar surface could have a profound impact on the choice of materials for commercial plants.

	Structural Metals	Non Structural Metals	Radiation Environment
Considerable Progress	• Understanding of Steady State Void Swelling • Swelling Resistance - Ferritic Steels • Miniature Specimens	• Performance of High Heat Flux Coatings • In-situ Extraction of T_2 from Ceramic Fuels • Higher Performance S/C Magnet Materials	• Internal Protection of ICF First Walls
Not Much Progress	• Non Stainless Steel Structural Material • Permanent First Wall • Pulsed Radiation Damage	• Stability of FW Coatings to Radiation Damage • Radiation Damage Resistance of Laser Mirror Coatings	• High Energy Neutron Source • Pulsed Neutron Source • Disruption Characteristics • ICF Target Spectra
New Ideas (since 1981)	• Low Activation Steel • V Alloys • Cu Alloys	• Be or BeO Coatings for High Heat Flux Materials	• ICF Materials Facility Design - SIRIUS-M • Tandem Mirror Materials Test Facility - TASKA-M/TDF/FEF • D-He^3 Resource Problem Solved

Fig. 1. Status of Fusion Materials Research, 1986 vs. 1981

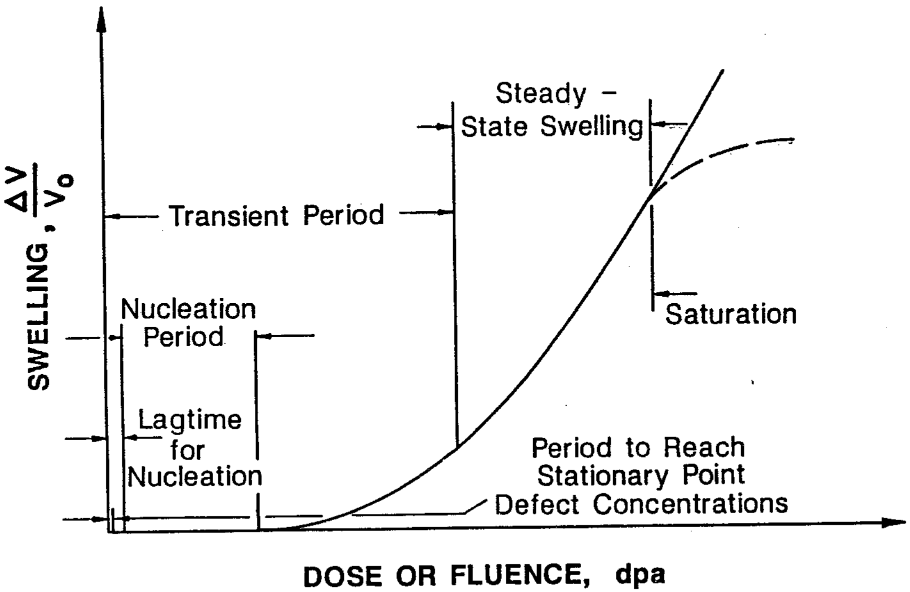

Fig. 2. Current understanding of the void swelling phenomenon in metals.

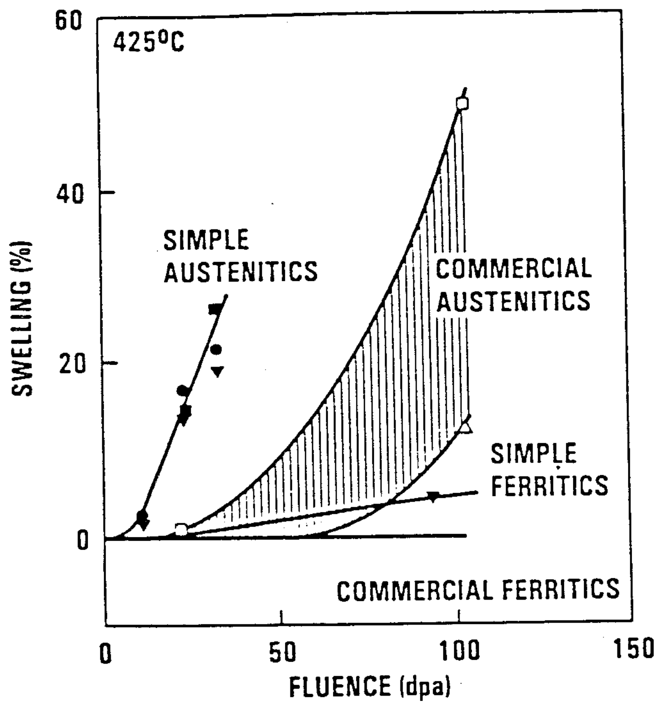

Fig. 3. Comparison of the swelling behavior of austenitic with ferritic steels [10,11].

2.1.3 Miniature Specimens

The effort to build a high flux, high energy neutron facility in the late 1970's revealed that the test volume might be quite limited. Therefore, an aggressive program was mounted to develop miniature specimens that might expand the data taking ability of such a device. Scientists at HEDL [13] in the U.S. were very successful in their attempts and a recent publication [14] summarizes the state of the art in the use of small-scale specimens for testing irradiated material. Without the development of such specimens the usefulness of point neutron sources (see Section 4.2.1) would be questionable.

2.2 Not Much Progress

2.2.1 Non Stainless Steel Structural Materials

For various reasons (i.e., higher temperature operation, lower long-lived radioactivity, coolant compatibility, etc.) materials scientists have continued their search of non steel structural materials. New metal designs have been proposed over the past decade ranging from Ti, Al, Mo, V or rapidly cooled alloys to ceramics such as C and SiC. None of these systems have been found to possess an integrated set of properties which could surpass the "normal" stainless steels. This is not to say that such a structural material can not be found, it merely says that up to now, we have not found a system that can beat the stainless steels.

2.2.2 Permanent First Wall

Because of the swelling and helium embrittlement problem, early estimates of the first wall (FW) lifetimes were in the 2-3 MW-y/m^2 range [12]. The early tokamak operated at 1 to 2 MW/m^2 and therefore reactor lifetime exposures were in the 30-60 MW-y/m^2 range. More recent designs of tokamaks and mirrors have used wall loadings in the 3 to 5 MW/m^2 range and hence reactor lifetime exposures now are in the 100-150 MW-y/m^2 range (~ 1000 to 1500 dpa). It was shown in Section 2.1.2 that from a swelling standpoint, austenitic alloys may achieve lifetimes of 5 to 7 MW-y/m^2 and ferritics may achieve 10 or more MW-y/m^2, but both lifetimes are far from the 100-150 MW-y/m^2 currently required for a reactor "lifetime" component. In fact, the calendar year life of components now (2-3 years) is not much different than it was 10 years ago with lower wall loading designs. The safety and cost implications of frequent first wall replacement as well as waste disposal issues are obvious and more work in this area is urgently needed.

2.2.3 Pulsed Radiation Damage

It has been known for over 10 years that pulsed radiation damage can yield different results than those accumulated under steady state conditions [15,16,17]. The wide disparity between the damage rate in ICF and MCF facilities is illustrated in Fig. 4 [18]. Ion irradiation simulation studies are also included in that figure and these studies have shown wide variations in the resulting microstructure as a result of pulsing. However, aside from some early theoretical work in this area [16], and a small effort at ORNL [17], virtually no attention has been given to this topic. It is expected to be most serious for ICF systems but pulsed tokamaks could also be affected.

2.3 New Ideas Since 1981

2.3.1 Low Activation Steels

Perhaps because of our inability to solve the first wall lifetime problem, there has been a great deal of pressure to reduce the impact of disposing associated with tonnes of highly radioactive material generated per reactor year of operation. The main thrust has been to tailor the alloys such that the waste can, after a suitable time, be buried near the earth's surface instead of constructing deep geological waste disposal facilities [19,20,21]. The effort has concentrated on removing the Mo and Ni elements from both austenitic and ferritic steels. Figure 5 [22] shows how such efforts have affected the decay profile in traditional and modified steels. The effect is mainly on the radiation emitted after 20 years of decay and both of the "modified" steels can qualify for near surface waste disposal (Class C waste according to 10 CFR-61). Current testing of the modified alloys is being conducted around the world and early indications are that such alloying changes can be easily accommodated in the designs.

2.3.2 Vanadium Alloys

The V alloy group was originally proposed in the 1970's because of its high void swelling resistance [23]. In the 1980's interest in this systems was reviewed because of its low activation aspects. In spite of cost, fabrication, compatibility, safety, T_2 permeation and mechanical property problems, this alloy does show some promise for high temperature fusion reactors [24]. A low level effort around the world has been initiated in the mid 1980's and by the time of the next IAEA meeting (1990?) a more definitive picture should emerge.

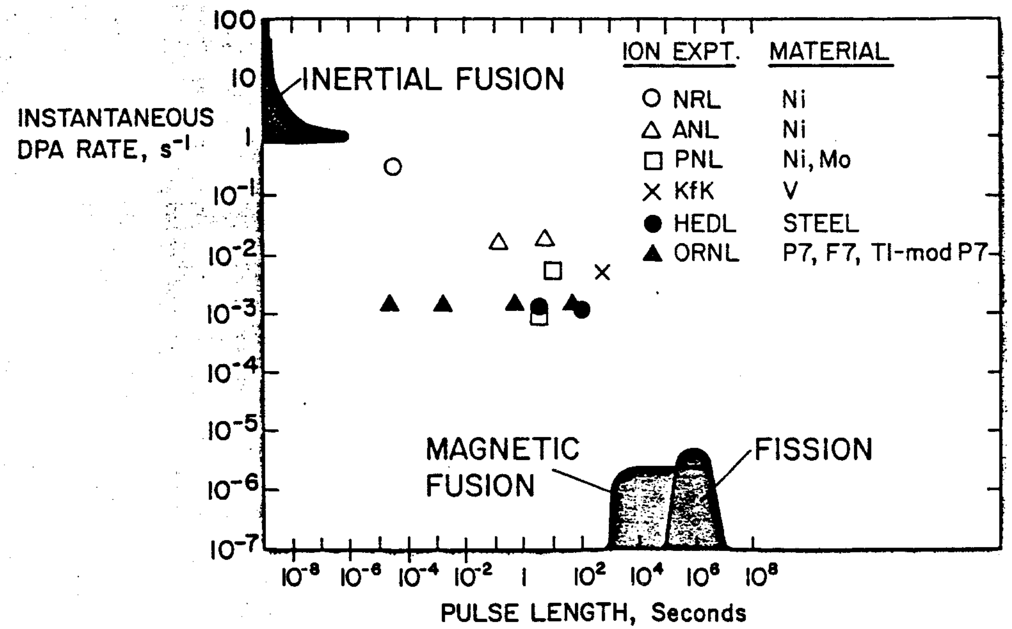

Fig. 4. Summary of experimental conditions studied for pulsed irradiation studies [18].

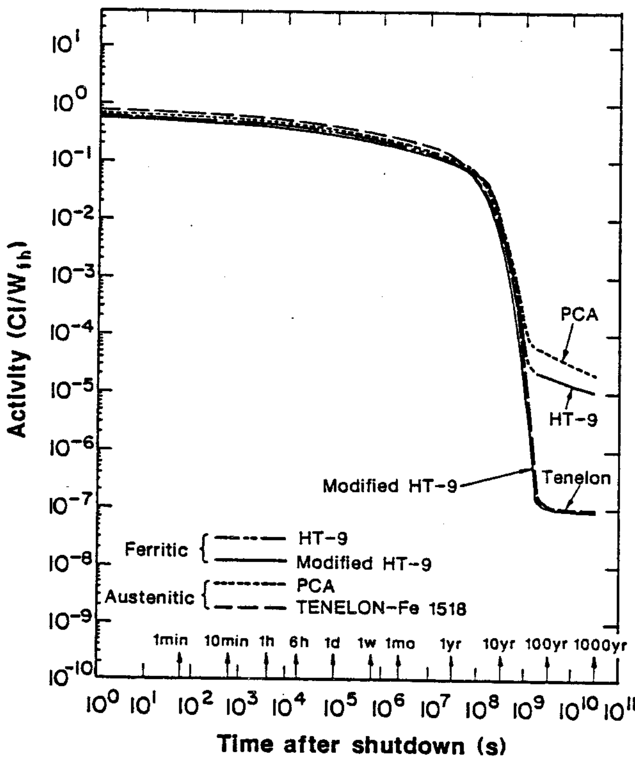

Fig. 5. Comparison of 'normal' ferritic (HT-9) and austenitic (PCA) steels with alloys modified to reduce long lived activity (courtesy of H. Attaya and M.E. Sawan).

2.3.3 High Strength, High Temperature Cu Alloys

Reactor systems which have very high power densities (such as RFP's) require a unique combination of high thermal conductivity and high strength at elevated temperatures. Copper alloys have been overlooked in past investigations but work on the Riggatron concept [25] brought the attention of the materials community to the Cu-Ni-Be system. Yield strengths of 600 MPa at 300°C seem to be possible while still retaining a large fraction of the thermal conductivity of the copper system. Even more recently, the $Cu-Al_2O_3$ system, commonly marketed under the name of Glidcop showed strengths at even higher temperatures [26]. Strength values of 250 MPa at 500°C. This means that if all other attributes such as radiation damage resistance, fabrication and compatibility are favorable, such alloys could play a very important role as reactor designers strive to increase the power density of reactors.

3. Non Structural Materials

3.1 Considerable Progress

3.1.1 Performance of High Heat Flux Coatings

There is a need in current vessel components to withstand up to 5000 watts/cm^2 for several seconds. In addition, disruptions can generate 10's of kW/cm^2 for 20-30 ms on plasma components. Each major facility has tried to solve the problem in different ways but TiC coatings on Mo or the use of pure graphite or Be components themselves have had considerable success. Both theoretical models [27,28] and experimental observations [29] are starting to yield a more complete picture of the response of materials to such high heat fluxes and the worldwide program seems to be well equipped to face the problems of the next level of devices in the 1990's.

3.1.2 In-situ Extraction of T_2 from Ceramic Fuels

The use of Li ceramics to generate T_2 in fusion reactors has only been seriously investigated since the mid 1970's. Since the last IAEA workshop in Tokyo several in-situ experiments have been conducted and information about the in-situ extraction of T_2 has been obtained [30-32]. The results show that it is possible to extract up to 99% of the tritium that is generated but they also show that most of the T_2 is extracted as T_2O instead of T_2.

3.1.3 Higher Performance Limits for Superconducting Magnet Materials

The radiation damage limits on superconducting (S/C) magnets have undergone a steady upward revision as we learn more about the basic mechanisms and as more data is generated. A current set of design limits is given in Table 1 which is contrasted to those used in an early (1974) reactor study, UWMAK-I [33]. It can be seen from Table 1 that there are currently essentially no limits on the thermal insulation or copper stabilizers due to design modifications. The nuclear heating is still governed by economic considerations and the limits on electrical insulators now appear to be less stringent than for degradation of Nb_3Sn filaments [18,34].

An example of why the limit for Nb_3Sn has been raised by a factor of 10 over past studies is shown in Fig. 6 [34]. Here the ratio of the critical current density before and after the irradiation is plotted as a function of damage energy. The damage energy has been converted into equivalent RTNS; MINIMARS and HFBR (a fission reactor) fluences. It can be seen that over a wide range of temperatures and fluences that the critical current density actually increases up to ~ 10^{19} n/cm^2 and thereafter exhibits a slow decrease. Therefore the design limit for Nb_3Sn has been moved up to 10^{19} a factor of 3 above previous design limits.

3.2 Not Much Progress

3.2.1 Stability of First Wall Coatings to Radiation Damage

As outlined in Section 3.1.1, the use of TiC, C or Be coatings on metallic or graphite substrates has been successful in handling high heat loads to in-vessel components of current devices. However, the success of these coatings depends on their ability to adhere to the substrates during operation and the differential expansion due to irradiation on such components is now known. This is especially critical for Be which will undergo a large number of (n,α) reactions and graphite which exhibits anisotropic growth. Considerable work needs to be done before the successful coatings for current devices could be considered for use in neutron generating facilities like NET, FER or the ETR.

3.2.2 Radiation Effects to Short Wave Length Mirror Coatings

One of the great successes of the laser fusion program has been the understanding of the wavelength dependence of laser light coupling to targets. It has been found that long wavelength light, such as that in CO_2 laser (10.6 μ) does not couple as efficiently as shorter wavelength and also generates more hot electrons which cause preheating of the target fuel. Unfortunately, the wavelengths that have been indicated as most efficient, 1 micron or less, require dielectric coatings to reduce absorption in the mirror itself. Examples of coatings include, but are not limited to Ta_2O_5, T_2O_3, HfO_2, Al_2O_3; all on SiO_2 substrates. These compounds are prone to color center

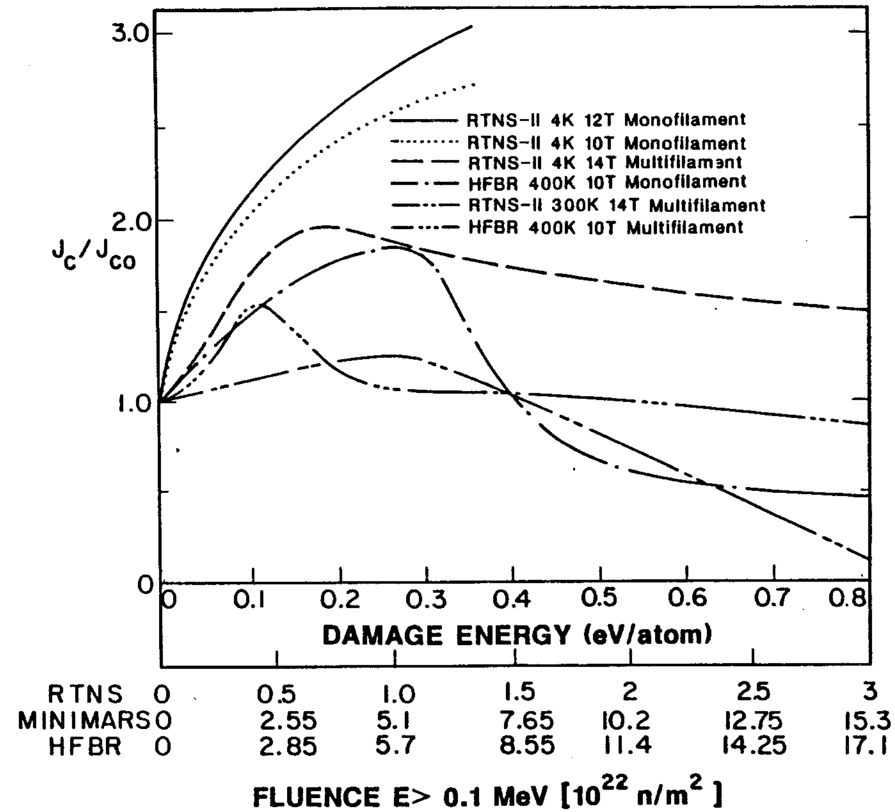

Fig. 6. Comparison of critical current properties of Nb_3Sn. Damage energy conversion to fusion and fission reactor spectra discussed in Reference [34].

Table 1. Progress in Radiation Damage Design Limits for Superconducting Magnets

Parameter	Units	UWMAK-I (1974)	MINIMARS (1986)
Thermal Insulator	rads	10^8	"No Practical Limit"
Electrical Insulator	10^9 rads	3	1000
Cu Stabilizer	dpa/y	10^{-4}	"No Practical Limit"
Peak Nuclear Heating	mW/cm^3	< 0.01	1-10 (Economics)
NbTi	10^{18} n/cm^2 (> 0.1 MeV)	4	> 100 (Sat. at 80% J_c)
Nb_3Sn	10^{18} n/cm^2 (> 0.1 MeV)	3	10

formation during irradiation and then usefulness in the harsh radiation environment of a fusion plant is not known. The mirrors could be placed far away from the target (10's of meters) but the expense associated with shielding and containment rapidly make such solutions too expensive. Unfortunately, there is essentially no information available on this phenomenon now.

3.3 New Ideas (Since 1981)

3.3.1 Be and BeO Coatings for High Heat Flux Materials

Coatings for past high heat flux components have generally been carbides or graphite. However as the metallic carbides are sputtered or evaporated, the metal ions contaminate the plasma. There was a general desire to reduce this contamination by using lower atomic number elements. The lowest Z metal, Be, was originally thought to be unsuitable for high heat flux components because of fabrication, safety and cost problems. Its low melting point (1277°C) and dimensional instability problems also contributed to the lack of early use.

Early design work by ANL on STARFIRE [35] and in the INTOR project [36] was coupled with experimental work by Sandia-

Albuquerque [37] to show that Be was a viable coating for pure element for in-vessel components. Early tests on ISX support this contention and soon to be conducted tests on JET will provide a basis for further assessment of this idea.

4. Radiation Environment

4.1 Considerable Progress

4.1.1 Internal Protection of ICF First Walls

The idea of using gaseous protection to shield the first walls from target irradiation was first proposed at Wisconsin in the mid 1970's [38]. Subsequently, the idea of using jets of liquid metals was proposed by LLNL scientist [39] and columns of liquid metals in porous tubes was proposed by scientists at the University of Wisconsin. In the past 5 years the gaseous, free liquid metal jet or INPORT unit protection schemes have all undergone considerable reviews and found to be reasonably sound. The TDF [40], LIBRA [41], HIBLIC [42], HIBALL-II [43], and HYLIFE [39] concepts have all been the subject of analysis which have shown the promise of such concepts. Even more recently the use of solid materials such as five Li_2O particles have been proposed in the Cascade concept [44].

4.2 Not Much Progress

4.2.1 High Energy Neutrons Source

In 1977, the fusion community was anticipating three major materials test facilities to be operating in the early 1980's. Table 2 lists those facilities, their status as of November 1977 and their status as of June 1986. Only the RTNS-II was completed and will run for about 9 years [45]. The Intense Neutron Source at LANL was canceled in 1978 and the FMIT project, after a series of delays was finally cancelled in FY 1986. In one year (1987) this will put the fusion materials community in the position of having no high energy neutron sources, nor will there be any prospect of such a facility. One might conclude that worse than making little progress in the field, the community is actually slipping backwards with respect to facilities.

4.2.2 Pulsed Neutronic Source

The lack of a high energy neutron source is compounded even more in the ICF community where it also lacks a neutron source with the appropriate flux intensity. Analyses of typical fusion target spectra reveal that the current neutrons can be as high as 10^{23} cm^{-2} s^{-1} for a nano second in a first wall. It was pointed out in Section 2.2.3 that such high

Table 2. Status of Major High Energy Neutron Irradiation Test Facilities

Neutron Source	Neutron Flux $cm^{-2}\ s^{-1}$	cm^3 of Test Volume	STATUS November 1977	June 1986
RTNS-II	2×10^{13}	1	Expect Op. March, 1978	To be shut down FY-1987
INS	10^{14}	3	Expect Op. Oct., 1979	Never Completed
FMIT	10^{14} 10^{15}	300 10	Expect. Op. Late 1983	Never Completed (Accl. Completed)

damage rates could give quite different results when compared to steady state damage. At the present time there are no known proposals to construct a source with such a time profile and the ICF community will have to depend on the fission and magnetic fusion community for all of its neutron damage data.

4.2.3 Disruption Characteristics

It is expected that disruptions will be a major problem for both the near term and eventual commercial tokamak reactors. Worst case scenarios from INTOR showed that such disruption might cause the failure of in-vessel components and similar catastrophics could occur in commercial units. The JET device has experienced numerous disruptions one of which actually lifted the entire device off its foundations. Unfortunately, the certainly with which such disruptions can be predicted, their duration and the peaking factors cannot be predicted with any reasonable degree of accuracy today. Without such information it is doubtful that commercial operation will be allowed for tokamaks.

4.2.4 ICF Target Spectra

This is an area where there has been essentially no information released to reactor designers in the past decade. This is partly due to the multiplicity of target designs and partly due to classification issues. Nevertheless, the end result is that there is no way to confidently design the first

wall of a bare (i.e., no gas or liquid metals between the target and the first wall) cavity. Fortunately, the internal protection schemes alleviates the need to know the exact target spectra but if such protection schemes do not work, then it will be very difficult to develop a credible first wall design.

4.3 New Ideas (Since 1981)

4.3.1 ICF Materials Facility Designs

From 1974 thru 1985 there were 34 magnetic fusion test reactor designs published compared to 1 ICF study. This fact coupled with the lack of pulsed neutron facilities has put the ICF community at a distinct disadvantage compared to the magnetic confinement approach. This situation was partially corrected by the preliminary design of SIRIUS-M [46], a laser driven materials test facility. Such a facility addresses the near term needs of the ICF community and hopefully will encourage other design group to address such problems.

4.3.2 Tandem Mirror Materials Test Facilities

Three new materials test facilities have been proposed since the last meeting in 1981; the TASKA-M [47], TDF [48], and FEF [49] reactors. Each of these reactors were attempts to reduce the costs and physics risk of going to low power, now tritium breeding units with Q values of 1 or less. All the designs show that such high neutron flux test facilities could be built relatively cheaply with modest extrapolations of present physics and in a time frame to significantly impact the design of a Demo.

4.3.3 D-He^3 Resource Problem Solved

Scientists have known for a long time that the D-He^3 reaction would be much more desirable than the DT cycle because of the much lower neutron production rate and because there is no requirement for breeding tritium in the blanket. However, the D-He^3 reaction was rejected as a commercial possibility because there seemed to be no large source of He^3 available on earth. Recently scientists at the University of Wisconsin [50] have shown that there is an extremely large source of He^3 on the surface of the moon deposited there by the solar wind for 4 billion years. Given that this source of He^3 can be economically delivered to earth, a major question to the materials community is, how would this change the current materials development program?

It has been shown that the neutron flux in a D-He^3 reactor can be reduced by a factor of 100 to 1000 when compared to the DT cycle [50]. Hence lifetime exposures to metals would be 1 to 10 dpa instead of the 1000 or more dpa now faced by reactor designers. This also means that induced radioactivity will be reduced by a factor of 100 to 1000 making almost every material (except Nb) a Class B or C waste material.

Sure no breeding material is required there would be less constraints on the cooling material (e.g. there would be no need for liquid Li) and water could be more seriously considered. The reduced afterheat (and T_2 inventory) would make the blanket designs "inherently" safe. Finally, since neutron economy is no longer of concern, new materials could be considered for reactor construction.

It is apparent that if a fusion economy is based on the D-He^3 cycle, no large scale neutron generating facilities need be built, thus shorting the development paths for commercialization. By the time of the next meeting, the full implications of such a discovery will be apparent.

5. Conclusion

Slippage in fusion commercialization dates has diminished the driving force behind advanced materials development. Nevertheless, the momentum of the materials program has resulted in considerable progress during the past 5 years since the last IAEA meeting. On the positive side, a better understanding of void swelling and the acceptance of ferritic alloys has increased our estimate of useful lifetimes. On the negative side, major "holes" in the ICF fusion materials program persist understanding pulsed damage effects, development of coatings for short wavelength lasers, and knowledge of target spectra. The lack of a high energy neutron source in FY '87 will severely limit the progress in materials development. The lack of progress on disruption characterization and their prevention is also disturbing.

A more positive note can be obtained from the new ideas put forward in the past 5 years. Modest efforts have continued to design the cost, high intensity fusion neutrons sources for the magnetic fusion program and a new effort to design a pulsed neutron facility has begun. The possible use of the D-He^3 fuel cycle could revolutionize the world fusion materials program and could be the biggest influence in the program in the past 30 years.

6. Acknowledgements

The author wishes to acknowledge the assistance of the staff at the Fusion Technology Institute and in particular Drs. H. Attaya, and M. Sawan. Partial financial support for this paper has come from the Department of Energy, the Wisconsin Electric Utility Research Institute, and the Grainger Foundation.

References

[1] R.L. HIRSCH, "Fusion Power: Past, Present and Future", p. 216 in Proc. Int. Conf. on Nuclear Solutions to World Energy Problems, ANS, Hinsdale, IL, 1972.

[2] "Fusion Reactor Design Problems", IAEA Workshop, Culham, England, IAEA, Vienna, 1974.

[3] "Fusion Reactor Design Concept", IAEA Workshop, Madison, Wisconsin, IAEA, Vienna, 1978.

[4] "Fusion Reactor Design and Technology", IAEA Workshop, Tokyo, Japan, IAEA, Vienna, 1983.

[5] G.L. KULCINSKI, "Progress in Materials Research for Fusion Reactors", p. 573 in Ref. 3.

[6] G.L. KULCINSKI, "Fusion Materials: Adapting to Realistic Reactor Environments", p.353 in Ref. 4.

[7] C. CAWTHORNE and E.J. FULTON, Harwell Symposium on "The Nature of Small Defect Clusters", ed. M.J. Makin, AERE-R5269 (1966) 466.

[8] W.G. WOLFER, "Advances in Void Swelling Helium Bubble Physics", J. Nucl. Mater., 122/123 (1984) 367.

[9] F.A. GARNER, "Recent Insights on the Swelling and Creep of Irradiated Austenitic Alloys", J. Nucl. Mater., 122/123 (1984) 459.

[10] P. GELLES, "Swelling in Several Commercial Alloys Irradiated to Very High Neutron Fluence", J. Nucl. Mater., 122/123 (1984) 207.

[11] T. LECHTENBERG, "Irradiation Effects in Ferritic Steels", J. Nucl. Mater., 133/134 (1985) 149.

[12] G.L. KULCINSKI, R.C. BROWN, R.A. LOTT and P.A. SANGER, "Radiation Damage Limitations for Materials in the Wisconsin Fusion Reactor Design", Nucl. Tech. 22 (1974) 20.

[13] N.F. PANAYOTOU, S.D. ATKIN, R.J. PUIGH, and B.A. CHIN, "Design and Use of Nonstandard Tensile Specimens for Irradiated Materials Testing", The Use of Small Scale Specimens for Testing Irradiated Material, ASTM-STP-888 (1986) 201.

[14] The Use of Small Scale Specimens for Testing Irradiated Material, ed. W.R. Corwin and G.E. Lucas, ASTM-STP-888, ASTM, Philadelphia, PA (1986).

[15] J.A. SPRAGUE and F.A. SMIDT, JR., p. 27 in Cooperative Radiation Effects Simulation Program, NRL Memorandum Report 2629, Aug. 1973.

[16] For a Summary of the Theory of Pulsed Radiation Damage See N.M. GHONIEM and G.L. KULCINSKI, "A Critical Assessment of the Effects of Pulsed Irradiation on the Microstructures, Swelling and Creep of Materials", Nucl. Tech./Fusion, 2, April 1982.

[17] N.H. PACKAN, "The Effect of Pulsed Irradiation on Microstructural Evolution", to be published in Radiation Effects.

[18] G. KULCINSKI, M. DuPOUY and F. ISHINO, "Key Materials Issues for Fusion", to be published in J. Nucl. Mater.

[19] The original idea for Substituting V for Mo in Ferritic Steels came from N. Ghoniem, UCLA.

[20] D.G. DORAN, H.L. HEINISCH and F.M. MANN, "Reduced Activation Guidelines in Perspective", J. Nucl. Mater., 133/134 (1985) 892.

[21] F.W. WIFFEN and R.T. SANTORO in Proc. Top. Conf. on Ferritic Alloys for Use in Nuclear Energy Technologies, AIME, 1985.

[22] H. ATTAYA and M.E. SAWAN, unpublished results.

[23] R.E. GOLD, E.E. BLOOM, F.W. CLINARD, D.L. SMITH, R.D. STEVENSON and W.G. WOLFER, Nuclear Tech/Fusion, 1 (1981) 169.

[24] D.L. SMITH, "Evaluation of Candidate Blanket Materials for Fusion Reactor Blanket Applications", J. Nucl. Mater., 122/123 (1984) 51.

[25] S.N. ROSENWASSER, R.S. STEVENSON, G. LISTVINSKY, D.L. VRABLE, J.E. McGREGOR and N. NIR, "Materials and Design Aspects of the Riggatron Tokamak", J. Nucl. Mater. (1984) 1107.

[26] Proc. of Workshop on Cu and Cu Alloys for Fusion Reactor Applications, Washington, DC 1983, CONF-830466.

[27] A.M. HASSANEIN, G.L. KULCINSKI AND W.G. WOLFER, "Surface Melting and Evaporation During Disruptions in Magnetic Fusion Report", Nucl. Engr. & Design/Fusion, 1 (1984) 307.

[28] A.M. HASSANEIN, "Simulation of Plasma Disruption Induced Melting and Vaporization by Ion or Electron Beam", J. Nucl. Mater., 122/123 (1984) 1453.

[29] C.D. CROESSMANN, G.L. KULCINSKI and J.B. WHITLEY, "Measured Vaporization and Melting of High Heat Flux Materials Due to Intense Heat Loads"

[30] D.L. BALDWIN and G.W. HULLONBERG, "Measurements of Tritium and Helium in Fast Neutron Irradiated"

[31] J.M. MILLER, S.R. BOKWA and R.A. VERRALL, "Post Irradiation Tritiuim Recovery from Lithium Ceramic Breeder" to be published in J. Nucl. Mater.

[32] R.G. CLEMMER et al., J. Nucl. Mater., 133/134 (1985) 234.

[33] B. BADGER et al., "UWMAK-I, A Wisconsin Toroidal Fusion Reactor Design", University of Wisconsin Report UWFDM-68, (1974).

[34] M.E. SAWAN, to be published in Nucl. Tech./Fusion.

[35] "STARFIRE - A Commercial Tokamak Fusion Power Plant Study", Argonne Report ANL/FPP-80-1 (1980).

[36] "INTOR - An International Tokamak Reactor", IAEA, Vienjna Reports in 1980, 1982, 1983, 1986.

[37] J.B. WHITLEY, J.M. McDONALD and D. CROESSMANN, "Sandia Electron Beam Test System (EBTS) SAND86-2039 (1986).

[38] B. BADGER et al., "SO(LASE - A Conceptual Laser Fusion Reactor Design", University of Wisconsin Report UWFDM-220 (1977).

[39] HYLIFE Summary, Laser Annual Program Report, 1978, Vol. 3, Chapter 9, UCRL-50021-78 (1979).

[40] G.A. MOSES et al., "Preconceptual Design of the Light Ion Beam Fusion Target Development Facility", University of Wisconsin Report UWFDM-664, November 1985.

[41] "Annual Progress Report for LIBRA Light Ion Beam Fusion Reactor Project-1985", Fusion Power Associates Report FPA-85-5, December 1985.

[42] "Heavy Ion Fusion Reactor - HIBLIC-I", Nagoya University, IPPJ-663, January 1984.

[43] B. BADGER et al., "HIBALL-II - A Heavy Ion Beam Fusion Reactor", Fusion Power Associates Report FPA-84-4, Karlsruhe Report KfK-3870 , 1984.

[44] J.H. PITTS, "CASCADE - A Centrifugal - Action Solid Breeder Reaction Chamber", Nucl. Fusion/Tech., 4, (1983) 967.

[45] D.W. HEIKKINEN et al., "The RTNS-II Fusion Materials Irradiation Facility", to be published in J. Nucl. Mater.

[46] S.I. ABDEL-KHALIK et al., " SIRIUS-M: A Symmetric Illumination, Inertially Confined Direct Drive Materials Test Facility", to be published in J. Nucl. Mater.

[47] G.A. EMMERT et al., "TASKA-M - A Low Cost Technology Test Tandem Mirror", Nucl. Engr. & Design, 2 (1985) 239.

[48] J.N. DOGGETT et al., "A Fusion Technology Demonstration Facility (TDF), Nucl. Engr. & Design, 2, (1985) 223.

[49] T. KAWABE, "Preconceptual Design of the Light Ion Beam Fusion Target Development Facility", University of Wisconsin Report UWFDM-664, November 1985.

[50] L.J. WITTENBERG, J.F. SANTARIUS and G.L. KULCINSKI, "Lunar Source of ^{3}He for Commercial Fusion Power", Fusion Technology, 10, September 1986.